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A CLASSIFICATION SYSTEM FOR RADIOACTIVE WASTE DISPOSAL — WHAT WASTE GOES WHERE?

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ABSTRACT

The Nuclear Regulatory Commission is developing a Radioactive Waste Disposal Classification System appropriate for use in the regulation of radioactive waste disposal. The purpose of developing this system is to classify the radioactive wastes according to the type and duration of containment required for their safe disposal. Potential impacts from two reference disposal methods representative of a three-category classification system are estimated using a consistent set of radionuclide release pathways. The estimated impacts are compared to the study dose guidelines in order to determine maximum allowable concentrations of isotopes appropriate for the waste categories. In general, postulated events in which individuals encounter the contaminated wastes provide the concentration limits for the waste categories.

SUBMISSION OF COMMENTS

Comments should be submitted by September 15, 1978, to William P. Bishop, A/D for Waste Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.

TABLE OF CONTENTS

Chapter		Page
	Title Page	i
	Disclaimer	ii
	Abstract and Submission of Comments	iii
	Table of Contents	iv
	List of Figures	x
	List of Tables	xii
	Acknowledgments	xvii
	Preface	xviii
1.	INTRODUCTION	1
2.	BACKGROUND INFORMATION	2
	2.1 The Nature of Radioactivity	2
	2.2 The Nature of Radioactive Wastes	4
	2.3 Dose Guidelines	6
	2.3.1 Exposures to Individuals (Guidelines 1 and 2)	8
	2.3.2 As Low As Reasonably Achievable (Guideline 3)	8
	2.3.2.1 Population Dose Commitments	8
	2.3.2.2 Cost-Benefit Considerations	10
	2.3.3 Positive Net Benefit (Guideline 4)	11
3.	RADIOACTIVE WASTE DISPOSAL CLASSIFICATION METHODOLOGY	13
	3.1 Basis of the System Methodology	15
	3.2 Environmental Exposure Events	15
4.	DATA BASE	18
	4.1 Description of Existing Low Level Waste Disposal Facilities	18
	4.2 Description of Reference Containment Facility	20
	4.3 Characterization of Low Level Wastes	20
	4.4 Inhalation of Dust by a Reclaimer	20
	4.4.1 Calculation for ^{239}Pu	24
	4.4.2 Application to Other Radionuclides	25
	4.4.3 Parametric Variations	25

TABLE OF CONTENTS (Cont)

Chapter		Page
4.5	Well Water Reclamation Event.	27
4.6	Direct Gamma Exposure	31
4.7	Atmospheric Releases of Contamination . . .	31
	4.7.1 Single Container Accident	33
	4.7.2 Continuous Operational Releases . .	34
4.8	Groundwater Migration	35
	4.8.1 Sensitivity Analyses.	38
	4.8.2 Dual Component Plutonium.	46
4.9	Ingestion of Food Produced on the Disposal Site	46
4.10	Surface Erosion	49
4.11	Maximum Allowable Concentrations.	49
	4.11.1 Relative Impacts from the Events.	51
	4.11.2 Maximum Allowable Concentrations for Mixtures.	55
	4.11.3 Variation of Dose Guidelines. . . .	55
	4.11.4 Maximum Allowable Concentrations for Intermediate Ground Burial. . .	55
4.12	Population Doses and Cost-Benefit	57
	4.12.1 Population Dose Rates from Air Transport	57
	4.12.2 Population Dose Rates from Groundwater Migration	57
	4.12.3 Waste Volume Projections.	59
	4.12.4 Cost Benefit Analysis	59
4.13	Parameters Used in Evaluating Releases from Nonradioactive/Low-Level Waste Interface	62
	4.13.1 General Considerations.	64
	4.13.2 Description of Reference Sanitary Landfill Facility.	66
4.14	Environmental Pathways.	66
	4.14.1 Inhalation of Contaminated Dust . .	66
	4.14.2 Water Well Near Landfill.	68
	4.14.3 Direct Gamma Exposures.	74

TABLE OF CONTENTS (Cont)

Chapter Page

4.14.4 Atmospheric Releases of Contamination 74
4.14.5 Groundwater Migration 76
4.14.6 Ingestion of Food Produced on RSLF Site 80

4.15 Nonradioactive/Low-Level Waste Interface. . 80
4.16 Cautions and Conclusions. 86

4.16.1 Impact of Decay Daughters 86

4.16.1.1 Decay Schemes 87
4.16.1.2 Daughter Effects from Reclaimer-Inhalation Events. 87
4.16.1.3 Impacts of Daughters on Groundwater Releases. 97

4.16.2 Additional Considerations 98

5. APPLICATIONS. 100

5.1 Classification of Specific Material 100

5.1.1 Adaptation of RWDCS Methodology . . 101
5.1.2 Description of Specific Materials 102

5.1.2.1 Fuel Element Hulls 102
5.1.2.2 Decommissioned Reactor Components 105
5.1.2.3 Decommissioned Fuel Reprocessing Plants. . . . 112
5.1.2.4 Failed Equipment and Noncombustible Waste . . . 115

5.1.3 Interface Limits for Specific Materials 115

5.1.3.1 Fuel Element Hulls 116
5.1.3.2 Activation of Large Pieces of Material 120
5.1.3.3 Surface Contamination of Large Pieces of Equipment. 122
5.1.3.4 Additional Future Exposure Events. 129

TABLE OF CONTENTS (Cont)

Chapter | Page

5.1.4 Conclusions Concerning Specific Material Limits 130

5.2 Low Specific Activity Materials 130

5.2.1 Characteristics of Uranium Mill Tailings. 131

5.2.1.1 Type of Ore. 131
5.2.1.2 Typical Milling Operation. 134
5.2.1.3 Characteristics of Mill Tailings 135

5.2.2 Environmental Pathways and Exposure Mechanisms 135

5.2.2.1 Description of Reference Mill Tailings Site 136
5.2.2.2 Mill Tailings Management at the RMTS. 136
5.2.2.3 Inhalation of Radon by Reclaimer. 139
5.2.2.4 Well Water Reclamation Event. 140
5.2.2.5 Direct Gamma Exposure. . . 141
5.2.2.6 Atmospheric Releases of Radioactivity. 143
5.2.2.7 Groundwater Migration. . . 144
5.2.2.8 Surface Erosion. 149

5.2.3 Cost-Benefit Considerations 149
5.2.4 Radioactivity in Coal Ash 150
5.2.5 Summary and Conclusions 150

5.3 Radiological Impacts for Maxey Flats Normalized to a Reference Reactor Year. . . 154

5.3.1 Introduction. 154
5.3.2 Facility Description and History of Operations 155

5.3.2.1 General Description of the Area 155
5.3.2.2 History of Operations. . . 158
5.3.2.3 Meteorology. 158
5.3.2.4 Radioactive Nuclide Inventory. 158

TABLE OF CONTENTS (Cont)

Chapter | Page

5.3.3 Environmental Pathways and Exposure Mechanisms 160

5.3.3.1 Inhalation of Dust by a Reclaimer. 161
5.3.3.2 Well Water Scenario. . . . 166
5.3.3.3 Direct Gamma Exposure. . . 166
5.3.3.4 Atmospheric Releases of Contamination. 166
5.3.3.5 Groundwater Migration. . . 169
5.3.3.6 Surface Water Transport. . 171

5.3.4 Population Dose Calculations. . . . 175

5.3.4.1 Population Doses from Groundwater Migration. . . 175
5.3.4.2 Population Doses from Surface Water Transport. . 178

5.3.5 Summary and Conclusions 178

5.4 Latty Avenue Site, Hazelwood, Missouri. . . 181

5.4.1 Introduction. 181
5.4.2 Description of Site and Calculation of Inventories. 182

5.4.2.1 General Description of Latty Avenue Site. 182
5.4.2.2 Calculations of Nuclide Inventory. 183

5.4.3 Environmental Pathways and Exposure Mechanisms 184

5.4.3.1 Inhalation of Radon by Reclaimer. 186
5.4.3.2 Well Water Scenario. . . . 186
5.4.3.3 Direct Gamma Exposure. . . 186
5.4.3.4 Atmospheric Releases of Contaminated Dust and Radon. 188
5.4.3.5 Groundwater Migration. . . 191
5.4.3.6 Surface Water Transport. . 191

5.4.4 Population Doses and Cost-Benefit. 192
5.4.5 Summary and Conclusions 193

TABLE OF CONTENTS (Cont)

Chapter		Page
5.5	Summary of Applications in Chapter 5. . . .	193
REFERENCES.		196
Appendix A.	The Advisory Panel Meeting on Waste Classification Held in Tucson, Arizona, on March 9 and 10, 1978	201
Appendix B.	Basis of Pathway Models.	206
Appendix C.	Impacts of Decay Daughters	215
Appendix D.	Engineering Evaluation of the Latty Avenue Site, Hazelwood, Missouri	240
Glossary.		G-1

LIST OF FIGURES

Number		Page
3.1	Schematic of Assumed Radioactive Waste Classification System	14
3.2	Set of Environmental Exposure Events	16
4.1	Schematic of Reference Containment Facility	21
4.2	Relationship Between Dust Loading and Exposure Time on Dose Rates at Constant Concentration Limit in Wastes	28
4.3	Aquifer Release Rate for ^{239}Pu as a Function of Time	42
4.4	Normalized Aquifer Release Rate for ^{239}Pu as a Function of the Dimensionless Parameter, Ro	44
4.5	Normalized Aquifer Release Rate for ^{129}I as a Function of the Dimensionless Parameter, Ro	45
4.6	Maximum Concentrations for ^{239}Pu for Event and Dose Guideline Variations	53
4.7	Maximum Concentrations for ^{90}Sr for Event and Dose Guideline Variations	54
4.8	Waste Volumes and Concentrations	60
4.9	Cost Benefit Analysis Results	63
4.10	Sanitary Landfill Operation	65
4.11	Radioactive Decay Chain for the 4n Series	88
4.12	Radioactive Decay Chain for the (4n+3) Series	89
4.13	Radioactive Decay Chain for the (4n+2) Series	90
4.14	Radioactive Decay Chain for the (4n+1) Series	91
4.15	Radioactive Decay Chain for ^{90}Sr and ^{135}Cs	92

LIST OF FIGURES (Cont)

Number		Page
4.16	Relative Impact, Compared to Parent, of Members of the ^{244}Cm Decay Chain.	93
4.17	Relative Impact, Compared to Parent, of Members of the ^{235}U Decay Chain	95
5.1	Generalized Potential Exposure Pathways from RMTS	137
5.2	Generalized Geological Cross-Section of the Maxey Flats Region.	156
5.3	Schematic Diagram of Water Pathways at Maxey Flats	162
5.4	Radon Concentration as a Function of Distance from the Site.	190

LIST OF TABLES

Number		Page
4.1	Survey of Existing Low Level Waste Disposal Facilities	19
4.2	Reference Containment Facility Parameters	22
4.3	Initial Inventory and Average Concentrations for Nuclides in RCF	23
4.4	Maximum Concentrations for Nuclides from the Reclaimer Inhalation Event	26
4.5	Maximum Inventories from Well Water Event	30
4.6	Maximum Concentrations of Radionuclides from Direct Exposure Events	32
4.7	Maximum Concentrations for Nuclides from the Single Container Accident	36
4.8	Characteristics of Nuclides in RCF Inventory	37
4.9	Maximum Inventories from Groundwater Migration	39
4.10	Results of Parametric Variations for ^{239}Pu	40
4.11	Results of Parametric Variations for ^{129}I	41
4.12	Maximum Allowable Concentrations for Food Pathway	48
4.13	Maximum Inventories from Sheet Erosion	50
4.14	Maximum Allowable Concentrations for Nuclides	52
4.15	Example Calculation for Mixture of Nuclides	56
4.16	Population Dose from Single Container Event	58
4.17	Relative Waste Concentrations and Volumes Projected per GW(e)yr	61

LIST OF TABLES (Cont)

Number		Page
4.18	Reference Sanitary Landfill Operations Parameters.	67
4.19	Parameters Used in Worker Inhalation Event	69
4.20	Maximum Concentrations in Waste at RSLF to Assure Guidelines Are Not Exceeded for Direct Inhalation	70
4.21	Maximum Inventory in the RSLF Based on Water Well Near Site Boundary	72
4.22	Maximum Concentrations in Wastes at the RSLF Based on Well Water Near Site Boundary.	73
4.23	Maximum Concentrations of Radionuclides from Direct Exposure Events	75
4.24	Isotope Concentrations in Wastes that Result in 500 Mrem/yr Dose to Individual at Site Boundary.	77
4.25	Population Density Distribution as a Function of Plume Distance from Source of Radiation at RSLF	78
4.26	Population Centerline Dose Data	79
4.27	Nuclide Specific Parameters for Groundwater Migration	81
4.28	Nuclide Releases from RSLF Through Groundwater Transport to River.	82
4.29	Interface Concentrations from Food Pathway	83
4.30	Maximum Concentrations for Exposure Pathways.	85
4.31	Maximum Allowable Concentrations Adjusted for Decay Daughters	96
5.1	Transuranic Contamination in Zircaloy Derived from Irradiation.	104

LIST OF TABLES (Cont)

Number		Page
5.2	Estimated Radioactive Waste from Power Reactors.	107
5.3	Nuclide Equilibrium Fraction at Elk River	110
5.4	Distribution of Nuclides in Elk River Waste and Power Reactors.	111
5.5	Comparison of AIF and Elk River Scale-Up Waste Projections	113
5.6	Fuel Reprocessing Plant Decommissioning Waste	114
5.7	Maximum Allowable Volumetric Contamination Concentrations in Hulls	118
5.8	Maximum Allowable Concentration Limit for Hulls	121
5.9	Maximum Allowable Concentrations for Activation of Specific Materials.	123
5.10	Nuclide Concentration in Waste Groups for Elk River Waste	124
5.11	Nuclide Concentration in Waste Groups for 3600 MWth Reactor	125
5.12	Maximum Allowable Surface Concentrations for Large Equipment	128
5.13	Principal Radioactive Decay Products of ^{238}U	132
5.14	^{238}U Decay Series	133
5.15	Parameters of Reference Mill Tailings Site.	138
5.16	Maximum Individual Doses from Well Water.	142
5.17	Population Dose Rates during Operational Phase	145
5.18	Population Dose Rates during Operational Phase	146

LIST OF TABLES (Cont)

Number		Page
5.19	Dose Commitments from Radon during Post-Operational Phase.	147
5.20	Groundwater Releases and Dose Rates	148
5.21	Summary of Uranium Content of Coal and Coal Ash in the U.S..	151
5.22	Radon Fluxes from Uranium Mill Tailings and Coal Ash Piles Required for 1000 MW(e) Power Plant	152
5.23	Impacts from Most Restrictive Events. . . .	153
5.24	Average Wind Speed and Direction Frequencies for Lexington, Kentucky	159
5.25	Normalization of Maxey Flats Inventory to a RRY of Waste	163
5.26	Inhalation Dose Rates/RRY of Waste.	164
5.27	Maximum Individual Dose Rates/RRY of Waste from Well Water	165
5.28	Continuous Operational Release Dose Rates/RRY of Waste.	168
5.29	Dose Rates from Evaporator Plume Releases .	172
5.30	Dose Rates from Groundwater Transport . . .	174
5.31	Dose Rates from Surface Water Transport . .	173
5.32	Surface Waters and Downstream Populations .	176
5.33	Infinite Population Doses for Groundwater Releases.	177
5.34	Population Dose Rates	179
5.35	Comparison of Calculated Dose Rates (mrem/yr/RRY)	180
5.36	Nuclide Average Concentrations and Inventory	185

LIST OF TABLES (Cont)

Number		Page
5.37	Parameters Used in Groundwater Transport Calculations.	187
5.38	Population, Concentrations and Doses vs. Distance from Site.	189
5.39	Summary of Pathways and Doses	194

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PREFACE

Radioactive wastes have different radiological and chemical properties. Available disposal methods provide different capabilities for containing or isolating wastes from mankind. The objective of a waste disposal classification system is to match the disposal method required for any waste with the characteristics of that waste.

In undertaking this study we have adopted that principal and constructed a methodology for its application. We have applied it to several real or model situations, and we have seen where the "fit" occurs.

1. INTRODUCTION

The NRC is developing a radioactive waste disposal classification system that classifies wastes according to requirements for their safe disposal. The ultimate goal is the proposal of a waste classification regulation. The objectives of this classification system are: (1) to classify radioactive wastes according to their requirements for safe disposal, (2) to address the concerns of the public, and (3) to implement the system without undue burden on those directly affected by it.

While developing the classification system, the NRC is conducting a study intended to provide the technical base. The methodology developed in the study must be technically correct; furthermore, for the study to be successful it must address the concerns of the public and those who will be directly affected by the classification system. To help insure technical accuracy many resources such as technical advisory panels and staff reviews are being used. To insure that the concerns of the public and those affected by the classification system are incorporated into a waste classification regulation, it is necessary to solicit their comments.

The study, which is nearing completion, has produced sufficient results to allow for serious comment. So that consideration of comments may be incorporated while the study is in progress, comments should be submitted promptly. While results of the study at its current status are being published at this time, additional refinements have already been identified. Therefore, this status report should be regarded as a serious solicitation of comments while the study is in progress rather than a solicitation for comments on a final report being considered by the NRC staff.

The report has been written for three different readerships: (1) the interested but not technically knowledgeable, (2) the generally technically knowledgeable, and (3) the expert. Although this approach may be burdensome to some readers, it will hopefully allow for a broad base of response to this study.

Chapter 2 contains technical and study background information, followed by a discussion of the classification system methodology in Chapter 3. The data base is given in Chapter 4 and Chapter 5 contains applications to specific wastes.

2. BACKGROUND INFORMATION

The purpose of this chapter is to provide background information about the nature of radioactivity and radioactive waste and the dose guidelines used in the study.

2.1 THE NATURE OF RADIOACTIVITY

The nature of an element is determined by the number of protons contained in each atom of the element. A proton is a positively charged subatomic particle. However, the atoms of a given element can contain differing numbers of an uncharged subatomic particle which is called a neutron. Atoms of a given element with a fixed number of protrons are called isotopes of the element; hence, an element can consist of one or more isotopes. For example, an atom of plutonium contains 94 protons. The isotope ^{239}Pu contains 94 protons and 145 neutrons (the atomic weight 239, is the sum of the protons and neutrons in an atom, 145 + 94 = 239), the isotope ^{240}Pu contains 146 neutrons and the isotope ^{241}Pu contains 147 neutrons.

Many isotopes of the elements spontaneously emit subatomic particles and are therefore radioactive. For example, when strontium-90 (^{90}Sr) undergoes radioactive decay it emits a subatomic particle called a beta particle. After decaying ^{90}Sr becomes ytrium-90 (^{90}Y) known as a decay daughter. Ytrium-90 is also radioactive and subsequently emits a beta particle, thus becoming zirconium-90 (^{90}Zr), a stable isotope.

When ^{239}Pu decays, it emits an alpha particle that consists of two neutrons and two protons. Alpha decay occurs mainly in isotopes with an atomic weight that is usually greater than 210.

While some isotopes decay, they also emit electromagnetic radiation known as gamma radiation. Gamma rays are very similar to x-rays only more penetrating. For example, when cobalt-60 (^{60}Co) decays to nickel-60 (^{60}Ni) by beta emission, two energetic gamma rays are also emitted.

It is also important to know the rate at which radioactive atoms decay. This decay rate, or activity, is expressed in curies (Ci) or microcuries (μCi) where 1 μCi equals 10^{-6} Ci or 3.7×10^4 disintegrations per second.

Another important parameter used in characterizing radioactive decay is known as the "half-life" ($T_{\frac{1}{2}}$). This is the

time that it takes for half of any initial quantity of radioactive atoms to decay to a different isotope. For example, the $T_{\frac{1}{2}}$ for tritium (^{3}H) is only 12.3 years but the $T_{\frac{1}{2}}$ for iodine-129 (^{129}I) is 15.9 x 10^{6} years (15,900,000 yr). The $T_{\frac{1}{2}}$ is also related to a parameter known as the decay constant (λ_d) by the following equation:

$$\lambda_d = \ln 2/T_{\frac{1}{2}} = 0.693/T_{\frac{1}{2}} \qquad (2.1)$$

The activity and the total number of radioactive atoms of a particular type depend upon their creation rates as well as upon their half-life for decay. If left undisturbed a radioactive parent and its decay daughters will all reach the same level of activity, matching that of the longest lived initiating isotope (the parent isotope). This condition is known as secular equilibrium. Thus if undisturbed, all members of the uranium-238 (^{238}U) decay chain have the same activity as the ^{238}U. However, if the uranium is removed, as in the milling of ore, the isotope that is not removed, thorium-230 (^{230}Th), becomes the controlling isotope.

Radioactive exposure occurs when the human body absorbs alpha particles, beta particles, or gamma rays. The range of alpha particles is very short; they affect an individual mainly when the alpha emitter is taken internally. Beta particles have a much lighter mass than alphas and have a longer range, but they can still cause damage to the skin or to internal tissues when taken internally. Gamma rays, however, can interact with all the tissue of an individual who is near gamma-emitting material.

The biological effects of radiation are related to the energy and type of radiation; therefore, for a given type of radiation, the potential biological damage or dose is measured in terms of the energy deposited per unit mass of the material. A dose of one rad corresponds to the absorption of 100 ergs/g of material. A given dose in rads from alpha particles has the potential for causing more damage to a biological system than the same dose from gamma radiation. It may, therefore, be misleading to directly compare doses from different types of radiation.

The "dose equivalent" measured in units of rem is used as a direct measure of the potential damage from different types of radiation. It is obtained by multiplying the radiation dose by the relative biological effectiveness for damage from the type of radiation. For example, a dose of 1 rad of alpha radiation yields a dose equivalent of about 20 rem.

In this report, biological impact will be measured in terms of the dose equivalent, in units of rems, mrem (0.001 rem) mrem/yr, man rem and man rem/yr. In later chapters of this report the term "dose" is intended to mean "dose equivalent," as in the statement "a dose of 0.5 rem."

Some radioactive elements, when taken internally, have a propensity for accumulating in certain organs of the body. If these elements are mainly alpha or beta emitters, then the dose to that particular organ is far greater than the corresponding dose if the same amount of material were distributed uniformly throughout the body. Thus, in determining dose from radioactive materials, parameters relating to the more restrictive dose are considered, whether they are to the whole body, the bone, the lungs or other organs.

2.2 THE NATURE OF RADIOACTIVE WASTE

Radioactive wastes are generated in each stage of the nuclear fuel cycle. Low-Level Wastes (LLW) include those generated in fuel fabrication, reactor and reprocessing operations (including clean-up and decommissioning processes) and non-fuel cycle sources such as hospitals and industrial users of radioactive materials. Uranium mill tailings and radioactivity in coal ash can also be considered radioactive waste.

Included in the low-level wastes are dewatered solids and otherwise solidified non-high-level liquid wastes. Some plant wastes, such as laundry and decontamination solutions, are common to most waste generating facilities. But other wastes are specific to certain plants, such as acid etch solutions from a fuel reprocessing plant. Prior to disposal, these liquid wastes must be immobilized. Absorbents are used widely in both the chemical and nuclear industry to immobilize liquids for transportation and disposal. Some of the typical granular or powdered absorbent materials include vermiculite, silica gels, plaster of paris, and various clays. The absorbent method, properly applied, will entrap the waste liquid so that no free liquid exists within the bulk material.

Incorporation of liquid and ion-exchange resin-type radioactive wastes in cement or concrete has been a common practice for many years. The optimum proportions of cement and waste vary with the type of waste to be solidified. Maximum waste contents for the solidified products are typically 75 wt% for solid waste or 33 wt% for aqueous

solutions or slurries. Besides cement or concrete, technology is available for the immobilization of a wide variety of wastes in asphalt, or bitumen, including neutralized evaporator concentrates, sludges, ion exchange media and incinerator ashes. Systems for immobilizing reactor wastes in ureaformaldehyde have also been marketed.

Solid low-level radioactive wastes also include a large variety of combustible items such as paper, rags, plastic sheeting, protective clothing, gloves, rubber shoes, wood, organic ion exchange resins, filter acids, etc. Much of the waste material is collected as general trash, which usually consists of a mixture of combustible and noncombustible items.

Waste compositions vary depending on the operations involved. General trash, or non-glove box waste, includes mainly cellulosic materials such as paper, wood, cardboard, absorbent cotton, cotton clothing, and rags. Glove box generated wastes, on the other hand, contain chiefly rubber or plastic materials since cellulosic materials are mostly excluded from glove box operations.

The primary constituents of the noncombustible types of low-level waste are contaminated and activated metal, but other noncombustibles such as glass and concrete are also present. Incidental quantities of combustible material such as grease, plastic, and floor sweepings may also be present with the noncombustible waste.

Decontamination and decommissioning of nuclear facilities will produce large quantities of construction and structural materials that have been contaminated and/or activated during use. These wastes are usually considered to be in the low-level category.

In addition to the waste generated in the nuclear fuel cycle, there are a number of other low-level waste sources, such as medical, university and research users of radiation, which send radioactive wastes to burial facilities for disposal. These wastes include animal remains, contaminated glassware and laboratory supplies, failed equipment, trash and small amounts of excess radioactive materials.

Because of the diversity of sources and treatments used in handling the wastes, a detailed description of a truly representative typical physical waste form is not possible. Most disposal facilities do require, however, that certain minimum specifications be met before wastes are

buried. These specifications include restrictions prohibiting free liquids, unreacted pyrophorics or explosives, and unlabeled hazardous biological or chemical waste forms.

2.3 DOSE GUIDELINES

The key objective of the classification system is to classify radioactive wastes according to their requirements for safe disposal. Hence, it is necessary to adopt guidelines for determining what is "safe." If there were nationally accepted criteria, these would have been adopted as the guidelines; however, such criteria for waste management do not currently exist. The Environmental Protection Agency (EPA) has a program to develop such criteria and standards, but these criteria are not available for this study.

Rather than delay this study and the development of the waste disposal classification system until criteria and standards are available, we have postulated a reasonable set of study guidelines. In developing the guidelines, the existing body of regulation governing radiological exposures of the population was considered. In general, the current regulations are based on the achieveable performance of existing facilities such as power plants. These performance levels which are achievable by these facilities may be more restrictive than required for the protection of the public and, therefore, are not appropriate for this study.

Most national and international organizations consider the International Commission on Radiological Protection (ICRP) to be a cognizant and authoritative body dealing with the matter of protection from ionizing radiation. Since the ICRP's recommendations will be considered reasonable, their recommendations have been used, specifically ICRP publication number 26 which was adopted in January of 1977.[1]

The ICRP recommends a system of dose limitations whose main features (or principals) are:

1. The dose equivalent to individuals shall not exceed the limits recommended for the appropriate circumstance by the ICRP.

2. All exposures shall be kept as low as reasonably achievable, economics and social factors being taken into account.

3. No practice shall be adopted unless its introduction produces a positive net benefit.

Stochastic effects are those for which the probability of health effects is related to the total population dose independent of the number of individuals receiving portions of that dose. It is further assumed that no dose, regardless of how small, is completely harmless. An example of stochastic risk is that of using public transportation. From a review of available information related to risks regularly accepted in everyday life, it can be concluded that the level of acceptability for fatal risks to the general public is an order of magnitude lower than for occupational risks. On this basis, a risk in the range of 10^{-6} to 10^{-5} per year would be likely to be acceptable to any individual member of the public. A whole body dose-equivalent limit of 500 mrem in a year, as applied to critical groups has been found to provide this degree of safety.

The ICRP does not propose dose limits for populations. Instead, it emphasizes that each man-made contribution to population exposures should be justified by its benefits, and that the limits for individual members of the public refer to the total dose equivalent received from all sources.

The study guidelines, which are discussed in the following sections, were derived from the ICRP recommendations:

1. 500 millirem/yr to a few individuals (10's of individuals).

2. 100 millirem/yr to many individuals (100's of individuals).

3. Exposures to the general population will be as low as reasonably achievable, economic and social factors being taken into account.

4. Population exposures are to be justified by resulting benefit: <u>One millirem/year/GWeyr</u> for many individuals as the result of the disposal of the waste generated per Gigawatt-yr of electrical energy (8.8×10^{12} kilowatt hours).

These dose limitations apply to either the whole body or critical organs, whichever event is the most restrictive.

2.3.1 Exposures to Individuals (Guidelines 1 and 2)

Guideline 1 was derived directly from the ICRP recommendations. The 100-millirem value in guideline 2 was derived by assigning 20% of the allowable annual dose rate (500 millirem) to waste management. If alpha emissions are included in the estimates of natural background as they are in estimating exposures from waste disposal, then 100-millirem is approximately 20% of natural background. It should be noted that only a very small segment of the total population would be exposed at this level. The average dose rate drops off at least as fast as the inverse of the distance from the source of the radiation, so that the dose rate drops by at least an order of magnitude from the facility boundary to a distance of one mile. The one mrem/yr/GWeyr value of guideline 4 is arbitrary, but it is so conservative that it should be defendable.

These guidelines are applied to estimates of potential exposures and not to estimates of actual exposures. The methodology contained in this study is directed toward estimated exposures if a particular series of hypothetical events took place. The methodology is not used to predict expected exposures. (This is an important distinction when judging these guidelines.)

When waste management criteria are available, and if they differ from the study guidelines, the quantitative results of this study would need to be revised. However, revising the results to reflect changes in the guidelines would be straightforward.

2.3.2 As Low As Reasonably Achievable (Guideline 3)

The first step in determining whether exposures are as low as reasonably achievable (ALARA) is to estimate the population dose commitment.

2.3.2.1 Population Dose Commitments

The usual method for estimating the total potential impact of low levels of ionizing radioactivity in the environment is to calculate the "population dose commitment." The population dose commitment can be considered to be the average exposure rate to the population times the size of the population times the period of time over which the population is exposed. It is the dose integrated over the entire population over all times. In practice, population dose commitments are often calculated over some arbitrary period of time such as 50 or 1,000 years.

The basis of the population dose commitment is the linear nonthreshold theory in which, for low levels of exposures, the number of health effects is proportional to the total exposure to the population. For example, the total number of health effects from the exposure of an individual to one rem would be the same as for the exposure of one thousand people to one millirem, provided that the exposure rates were low. Population dose commitments are usually expressed as man-rem (for the previous example it would be one man-rem).

Provided that there is some upper bound (often referred to as a speed limit) for which dose rates from radiation are not exceeded, dose rates are not presented as part of the results of the man-rem calculations. This failure to consider dose rates can result in misleading conclusions. For example, consider the emissions from uranium mill tailings and coal ash. For the same amount of electricity generated by a coal-fueled and a uranium-fueled power plant, the potential population dose commitment from the coal ash and mill tailings is about the same. Thus, it would appear that if the coal ash can be safely disposed of, the mill tailings can also be safely disposed of using the same disposal method. However, this conclusion does not reflect the potentially higher initial dose rates from mill tailings.

Another example of the difficulty of obtaining useful information from population dose commitment calculations can be drawn from comparing the United States population dose commitment of natural background to that from mill tailings (over the effective life of ^{230}Th, the parent isotope). The population dose commitment from background is about 10^{13} (or 10,000,000,000,000) man-rem as compared to an estimated 3×10^7 to 3×10^9 man-rem (depending on the control of radon emissions) from mill tailings assuming that all known U.S. reserves of uranium ore are processed. The ratio of man-rem from natural background to that from mill tailings is between about 10^3 and 10^5. Do these results mean that natural background is acceptable and therefore mill tailings are also acceptable, or do they mean mill tailings are not acceptable and therefore natural background is a very great hazard? Without considering dose rates this dilemma is difficult to resolve.

Using the value of 3×10^8 man-rem for mill tailings, the annual dose rate to the average member of the United States population would be 14 microrem. Even if there were an unlimited supply of uranium ore and mill tailings were

generated at the current rate indefinitely, the average annual dose rate from the mill tailings would be 22 millirem. This equilibrium value is reached when the activity in the previous mill tailings decays at the same rate as new activity is being added to the environment. That is, all future generations could continue to add radioactivity to the environment at the same rate as we may be doing with mill tailings without increasing the average exposure to an individual by more than 4.5% of background. Further, this equilibrium dose rate would not be reached for hundreds of thousands of years. (For the purpose of this study, exposures from alpha emmiters are included in background which is estimated to be 500 millirem.)

In actual practice, most of the exposure could be potentially experienced by a small segment of the population. Therefore, by restricting the exposures to the individuals receiving maximum exposure and to small segments of the population, the exposures to the majority of the population will be very small. It has been observed in this study as in others,[2] that by protecting the few or many individuals, we also protect the entire population.

2.3.2.2 Cost-Benfit Considerations

The population dose commitments or man-rem calculations do have a useful function as an indicator of the total impact of a given waste management action when considering the cost-benefit ratio. If it is true that the impact is proportional to the man-rem then the acceptable cost in avoiding that impact should be directly related to the impact, i.e. to the man-rem. The usual method of expressing this--the cost-benefit ratio--is the dollar per man-rem. That is, a certain amount of dollars should be spent to avoid each man-rem. The most commonly quoted value is $1000/man-rem.[3] In the proper application of the criterion of $1000/man-rem, all costs and all benefits should be considered. Since this was not always possible in the scope of the present study, this criterion was not always applicable. For example, if the alternative course of action involves significant impacts that are not expressed in dollars, the use of a dollar per man-rem criterion results in an imbalance in "cost" per man-rem ratio. The other impacts to be considered include the transportation risk, use of valuable resource, occupational risks, etc.

As stated previously, the time over which the man-rem are accumulated can be important. Consider a source of one

million man-rem from ^{232}Th and its daughters. A $1000/man-rem criterion would suggest that it would be appropriate to spend as much as one billion dollars to eliminate this potential hazard. However, because of the long half-life of ^{232}Th, the peak annual man-rem rate would be 0.00005. Over the first million years there would be only 50 man-rem accumulated. Although this is an extreme example, it does suggest the need to restrict the time over which the man-rem calculations are performed when applying the dollars per man-rem criterion.

In this study, the dollar per man-rem criterion will be used with caution. When it appears that the criterion can be reasonably applied without qualification it will be applied. When the application of the criterion requires qualification, the man-rem will be discussed but conclusions will be avoided. The ultimate application of the criterion appears to be more appropriate in the licensing process than in generic waste classification.

2.3.3 Positive Net Benefit (Guideline 4)

After it has been determined that the waste can be disposed of safely and that the population exposures are as low as reasonably achievable, there is a further question. Can the population exposures be justified in terms of the benefit received from generating the waste? That is, do the positive impacts (benefits) from generating the waste exceed the negative impacts from disposing of the waste? The guideline used in this study is this: There is a net positive benefit if many individuals each receive a dose rate of less than 1 millirem/yr for each GWe-yr of energy generated.

As previously discussed, the dose rate provides more meaningful information than does the population dose commitment. Therefore, the guidelines for positive net benefit in this study are expressed in dose rates. The practical effect of this study guideline is to assure that the dose rates from small disposal operations are commensurate with the quantity of waste being disposed.

There are many wastes from sources other than the nuclear fuel cycle. These wastes are not directly considered by this study guideline. This is due to the difficulty of expressing the benefits from medical, industrial and research waste generation in quantified terms. However, it seems reasonable to assume that if fuel cycle wastes can be properly managed, then other types of waste with similar characteristics will also be properly managed.

In summary, the primary guidelines used in this study are the protection of the few and many individuals with the secondary consideration given to the further reduction of population exposures when the true cost of such reductions can be reasonably identified. Or more simply, if after the speed limits have been met, are the population exposures as low as can reasonably be achieved or are further reductions justifiable?

3. RADIOACTIVE WASTE DISPOSAL CLASSIFICATION METHODOLOGY

The basic objective of the waste classification methodology is to insure that the various radioactive wastes are to be disposed of by the most appropriate disposal method according to the study guidelines. In a previous part of this study, conducted by Science Applications Incorporated for Lawrence Livermore Laboratory[4], it was determined that the classification system should contain three categories of actions in handling radioactive waste:

1. Discharge directly to the biosphere in a manner similar to the handling of routine trash.

2. Confine the waste for a period of time in a controlled manner with predictably low release rates.

3. Isolate the waste from the biosphere so that biologically significant releases or inadvertent reentry by mankind into the disposal area is highly unlikely.

Although it is possible to consider several subcategories to these three items, the above categorization is adequate for the present time. The need for removing the waste from man's direct environment increases as the potential for exposing individuals or populations to unacceptable levels of radiation from the waste increases. It follows, as shown in Figure 3.1, that wastes appropriate for disposal by direct release (action 1) can be considered as non-radioactive, in a regulatory sense. Furthermore, wastes appropriate for interim confinement (action 2) can be considered as "low-level" wastes (LLW). Wastes that must be isolated from the biosphere (action 3) can be considered as "high-level" wastes (HLW). However, these are not the current legal definitions.

With this system, classification of the wastes according to the nature of the radioactivity (transuranic, fission product, activation product, half-life, etc.) is considered only insofar as these characteristics relate to protection of the public health and safety. Also, alpha-emitting material including transuranics are not classified separately but follow essentially the same disposal criteria as other radioactive waste material. In the classification system, the method governing the disposition of waste is based primarily on the hazard potential and is expressed in terms of radioactivity per unit volume at the time of disposal.

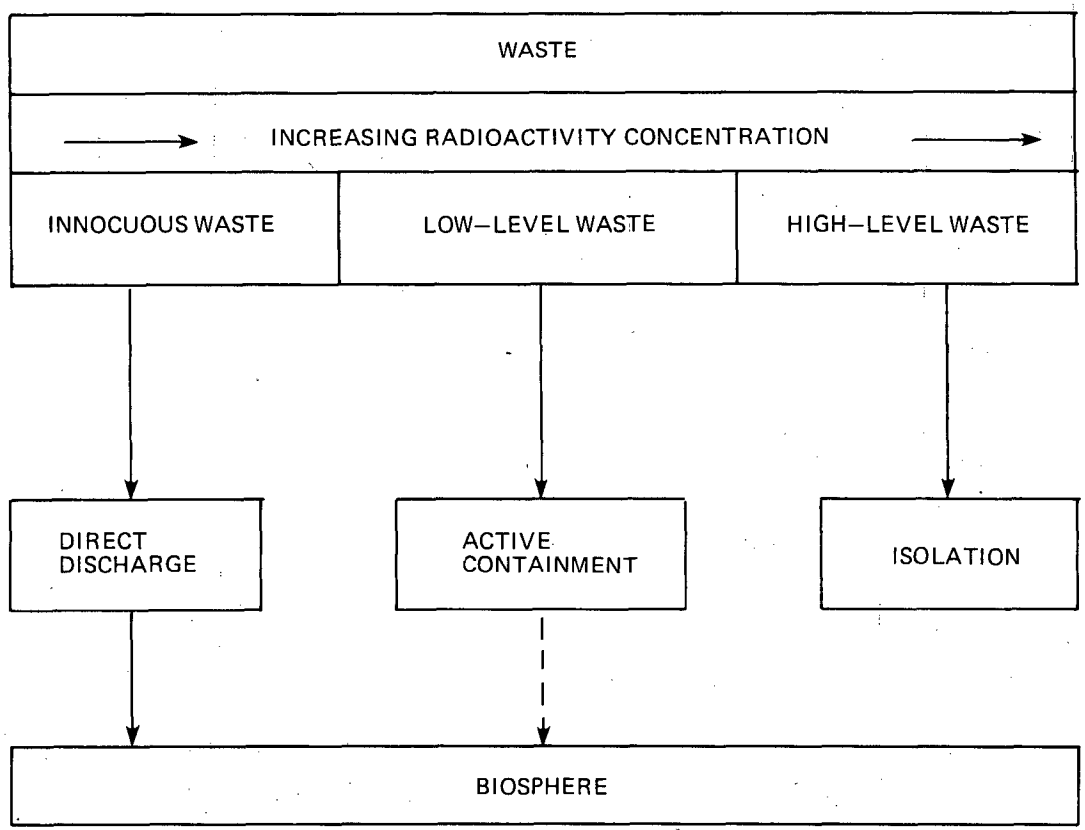


FIGURE 3.1 SCHEMATIC OF ASSUMED RADIOACTIVE WASTE CLASSIFICATION SYSTEM

3.1 BASIS OF THE SYSTEM METHODOLOGY

A desirable method for achieving the objectives is to utilize presently available technology for estimating the potential risks to the public from the disposal of radioactive waste. By comparing the potential risks with the study guidelines, the waste volume concentration or method of disposal can be modified to provide adequate protection to the public. The concept for this methodology was developed early in the program. In using this method, it is necessary to develop the technical basis for a consistent analysis of potential risks from disposed radioactive waste.

The potential exposures from disposed radioactive waste can occur either from the waste migrating from its disposal location into man's environment or it can occur from individuals encountering the waste.

3.2 ENVIRONMENTAL EXPOSURE EVENTS

The analytical procedures for determining the quantitative values of the waste-class interfaces employ three basic steps:

1. Identifying a set of reasonably conservative exposure events.

2. Describing the transport of the radioactivity through the environment to man or man's encounter into the waste.

3. Calculating the concentrations or inventories of radioactivity in the wastes that will assure that the doses to the exposed population groups both from the standpoint of the maximum individual dose and the total population dose do not exceed the dose guidelines.

The set of potential exposure events formulated for the analysis includes events in which individuals may come into contact with the waste in place as well as events in which the waste is transported offsite either by water or air. The events are categorized in Figure 3.2.

No period of administrative control for action 1 was assumed. For action 2 mechanisms in which individuals at the disposal location can be exposed to radiation occur after the site has been released for unrestricted use, postulated to be 150 years. During the 150 years of

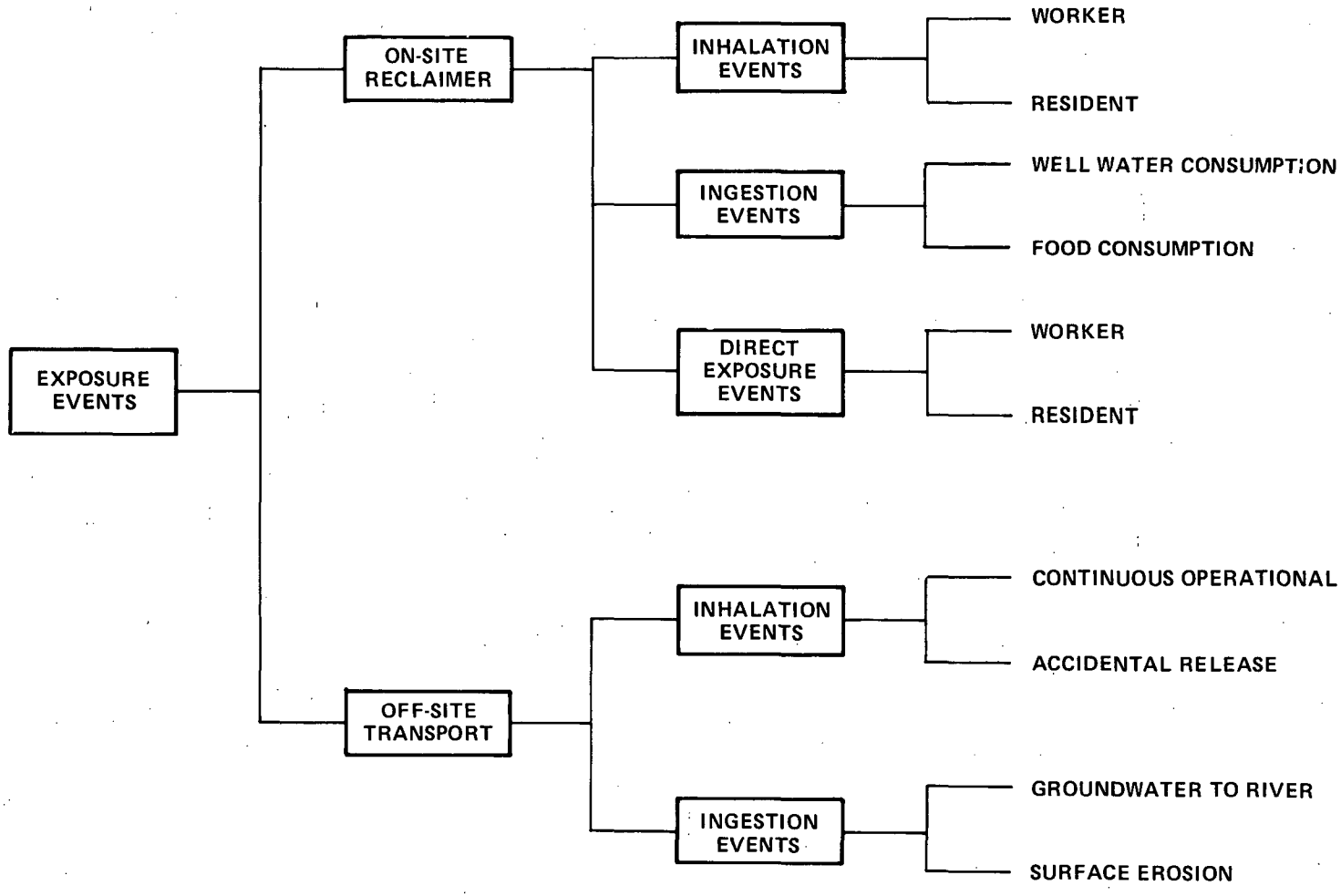


FIGURE 3.2 SET OF ENVIRONMENTAL EXPOSURE EVENTS

administrative control, most of the short-lived isotopes will have decayed. The mechanisms included in the set by which individuals contact the waste are:

1. Inhalation of dust by a reclaimer digging in the waste, or by residents on the reclaimed site.

2. Ingestion of water from a well dug by a reclaimer.

3. Consumption of food grown in a garden containing contaminated soil.

4. Direct exposure to workers or residents from gamma radiation.

Events in which the radioactivity is transported from the site include:

5. Atmospheric transport to individuals via continuous releases and accidental releases.

6. Groundwater migration to a resource waterway.

7. Surface erosion to a resource waterway.

During the conduct of the study, numerous pathways have been and will continue to be considered. However, many of these pathways are either not restricing or are highly improbable. Only those reasonable pathways which are restricting are considered in detail. This does not mean that these events will occur. It is the intent of the methodology to establish consistent sets of events to be analyzed in such a manner as to estimate a range of probable impacts. The guidelines then can be used to determine quantitative interface values of the classification system.

The basic approach, the dose guidelines, release events, calculational basis and results, were examined and discussed by an advisory panel composed of representatives from all aspects of nuclear waste management. Panel members and meeting minutes are given in Appendix A.

Using these interfaces, the NRC will develop Regulatory Guides for the application of the Radioactive Waste Disposal Classification System (RWDCS). (One such Guide will classify waste according to the source of the waste. For example, the waste from the primary cooling system of a power reactor may be classified as low-level waste while the waste from the secondary cooling system, provided there are no significant leaks from the primary, may be non-radioactive waste.)

4. DATA BASE

This chapter examines the bases for the Reference Containment Facility (RCF) and the Reference Sanitary Landfill Facility (RSLF), and describes the results of the application of the RWDCS methodology to the reference base cases.

The procedure for determining the LLW/HLW interface is to examine the environmental impacts from waste disposed in an RCF. Concentration or inventory limitations for low-level wastes are then determined by comparison of the impacts with the guidelines given in Chapter 2.

Similarly, the interface between low-level waste and wastes of sufficiently small concentrations of radioactivity to be considered as nonradioactive from the regulatory point of view is evaluated.

4.1 DESCRIPTION OF EXISTING LOW-LEVEL WASTE DISPOSAL FACILITIES

For perspective on current LLW handling and to provide background for judging the reasonablility of the RCF, a review of existing low-level waste burial facilities was performed.[5-11] Current practice in the United States is to dispose of solid low-level waste by shallow land burial. There are presently six commercial low-level radioactive waste burial facilities and five major active facilities operated by the Department of Energy (DOE) in this country.

The characteristics of these facilities were reviewed to establish reasonably conservative yet realistic parameters as input to the computational model based upon a RCF. Table 4.1 gives a summary of the existing low-level waste disposal facilities which were included in the review.

Although the source of the wastes may differ from site to site, the general operational characteristics of the facilities and the compositions of the actual wastes are generally similar.

Basically, low-level wastes received at the burial facility are placed directly in pits or trenches excavated into the native soil or till at the site. The overburden removed during excavation is then used to cover the wastes. The pits and trenches are sloped and the cover is applied for control of ground water and surface runoff from precipitation.

TABLE 4.1 SURVEY OF EXISTING LOW-LEVEL WASTE DISPOSAL FACILITIES

Location	Commercial Capacity (m^3)	Climate	Nearby Rivers	Cover Depth (m)	Observed Radionclide Migration
HANFORD, WA *		Semi-Arid	10 km to Columbia	2.5	Through Uptake by Deep Rooted Plants
RICHLAND, WA	9×10^5	Semi-Arid	10 km to Columbia	2	Not Observed
BEATTY, NV	7×10^5	Arid	3 km to Amorgosa	2	Not Observed
INEL, ID*		Semi-Arid	3 km to Big Lost	1	Possibly by On-Site Ground Water
LOS ALAMOS, NM*		Semi-Arid	8 km to Rio Grande	1.5	On Site Vadose Water Zone
SHEFFIELD, IL	2×10^5	Humid	Site Boundary	1	Not Observed
MOREHEAD, KY	3×10^6	Humid	500 m	1	On and Off-Site Ground and Surface Water
OAK RIDGE, TN *		Humid	On-Site	1	On-Site Ground Water, Off-Site Surface Water
SAVANNAH RIVER, SC*		Humid	On-Site Savannah	1.2	On-Site Ground Water
BARNWELL, SC	2×10^6	Humid	Site Boundary	3	Not Observed
WEST VALLEY, NY	2×10^5	Humid	On-Site	3	On-Site Ground Water, Off-Site Surface Water

*DOE Sites

4.2 DESCRIPTION OF REFERENCE CONTAINMENT FACILITY

The RCF is basically a model shallow land burial facility whose parameters were determined using "good engineering judgement". Therefore, the RCF parameters are not the average of the existing shallow land burial facility parameters even though data from existing sites were considered. A schematic drawing of the RCF is shown in Figure 4.1. An aquifer is assumed to lie 10 m below the bottom of the burial trenches and the water in this aquifer flows at a rate of 100 m/yr toward a large river located 1 km away. The total disposal capacity of the RCF is 6 x 10^5 m^3 of waste, which is sufficient to contain the volume of low-level radioactive wastes generated by 1000 reference reactor years (RRY), or about 800 GWeyr of nuclear power production. Table 4.2 contains the key parameters relating to the RCF. Based on present practices, it is assumed that the wastes will have a minimum of 1 m of earth covering. The advantages of increasing the earth covering to at least 10 m by intermediate depth burial are also examined.

4.3 CHARACTERIZATION OF LOW-LEVEL WASTES

The diversity of sources and treatments used in handling wastes precludes a detailed description of volumes, concentrations and physical forms of low-level waste. However, most existing disposal facilities prohibit liquids, pyrophorics, explosives and hazardous biological or chemical waste forms. Only solid waste will be considered for disposal at the RCF. The average concentrations in wastes for disposal at the RCF, given in Table 4.3, have been determined by NRC as representative of those expected to be disposed of in the future.[12] Key isotopes from this total list of expected concentration are used throughout the balance of this report for analyzing the waste classification system.

4.4 INHALATION OF DUST BY A RECLAIMER

The first occurrence considered is the reclamation of the RCF site after 150 years. For this event people are exposed to contaminated dust while moving earth at the site. The equation relating the dose rate of the few individuals to the maximum concentration of the contaminant in the waste is:

$$D = C_m K U_a T_x f(DF)_m \exp(-150\lambda_m)/\rho \qquad (4.1)$$

where

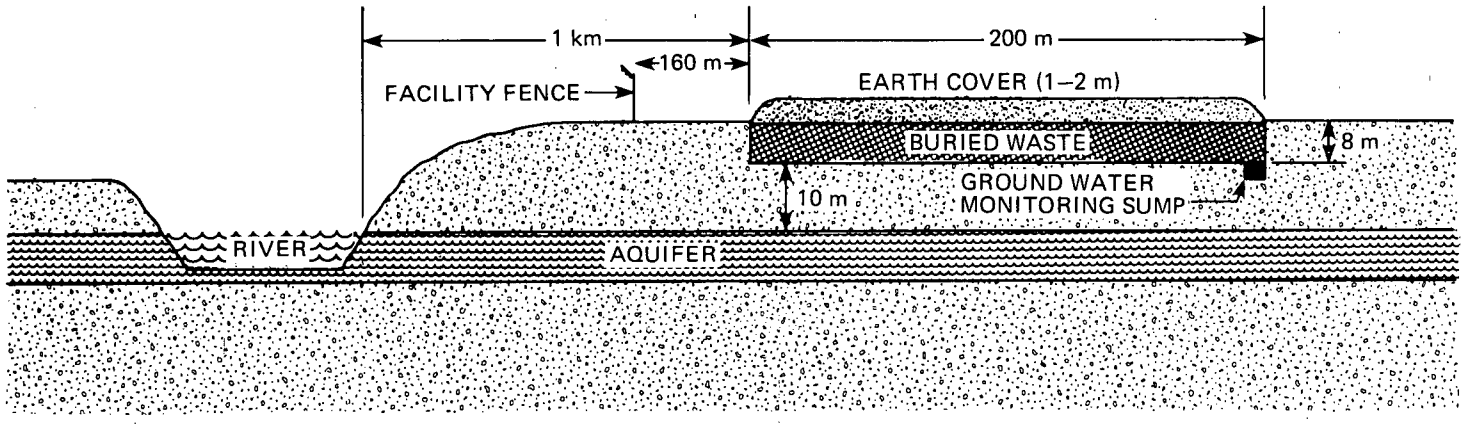


FIGURE 4.1 SCHEMATIC OF REFERENCE CONTAINMENT FACILITY

TABLE 4.2

REFERENCE CONTAINMENT FACILITY PARAMETERS

Parameter	Value
Site Plan Area	2.0 +6 m^2 [a]
Site Capacity for Waste	6.3 +5 m^3
Site Capacity for Waste	1.0 +3 RRY of nuclear power production
Distance to Site Boundary	1.6 +2 m
Distance-Pit to Aquifer	1.0 +1 m
Water Velocity-Pit to Aquifer	1.0 +1 m/yr
Annual Precipitation	1.1 m/yr
Distance-Site to Surface Water	1.0 +3 m
Water Velocity-Aquifer	1.0 +2 m/yr
Dispersion Coefficient	1.0 +1 m^2/yr
Minimum Earth Cover Over Waste	1.0 m
Maximum-to-Average Concentration of Waste	1.0 +1
Fraction of Pit Volume Occupied by Waste	0.5
River Flow Rate	5.0 +2 m^3/s

[a] 2.0 +6 is equivalent to 2.0×10^6. This notation is used in most tables.

TABLE 4.3

INITIAL INVENTORY[a] AND AVERAGE CONCENTRATIONS[b] FOR NUCLIDES IN THE RCF

Nuclide	Total Inventory (Ci)	Average Concentration (Ci/m^3)
^{3}H	2.7 +6	4.3
^{14}C	9.4 +4	1.5 -1
^{55}Fe	1.6 +5	2.6 -1
^{60}Co	4.8 +5	7.7 -1
^{90}Sr	1.8 +3	2.9 -3
^{99}Tc	1.2 +1	1.9 -5
^{129}I	2.4	3.8 -6
^{135}Cs	1.2 +1	1.9 -5
^{137}Cs	3.2 +5	5.1 -1
^{235}U	2.0 +1	3.2 -5
^{238}U	4.5 +2	7.1 -4
^{237}Np	1.1 -2	1.8 -8
^{238}Pu	8.2 +1	1.3 -4
^{239}Pu	1.1 +1	1.7 -5
^{240}Pu	1.7 +1	2.7 -5
^{241}Pu	4.2 +3	6.7 -3
^{242}Pu	6.0 -2	9.5 -8
^{241}Am	8.2	1.3 -5
^{243}Am	5.2 -1	8.3 -7
^{243}Cm	1.6 -1	2.5 -7
^{244}Cm	4.8 +1	7.6 -5

[a]Inventory at closing, prior to release

[b]from Table 2, reference

D = allowed dose rate (500 mrem/yr)

C_m = maximum concentration of isotope m in the waste at the time of burial, ($\mu Ci/cm^3$)

K = dust loading in the air (5×10^{-4} g/m^3)

U_a = breathing rate of exposed individuals (0.91 m^3/hr)

T_x = time period of exposure (500 hrs)

f = average to maximum concentrations of isotope m in site soil

$(DF)_m$ = dose rate conversion factor for isotope m (mrem/yr/μCi inhaled)

λ_m = radioactive decay constant for isotope m$(yr)^{-1}$

$\exp(-100\lambda_m)$ = correction for decay during 150-yr control period

ρ = density of waste material (1.6 g/cm^3)

Solving eq (4.1) for C_m, the maximum concentration at the burial, yields:

$$C_m = \frac{D\ \rho\ \exp\ (150\lambda_m)}{K\ U_a\ T_x\ f\ (DF)_m} \qquad (4.2)$$

4.4.1 Calculation for ^{239}Pu

For the base case event, the exposure is to a few individuals who work in dusty air loaded with 5×10^{-4} grams of dust per m^3 of air for about one-quarter year (500 hrs).

The factor f is the product of the average-to-peak concentration of the nuclide in the waste (taken to be 0.1 based on data from INEL[10] and elsewhere[11] with the fraction of waste in the trenches (0.5); thus f is equal to 0.05 for this case.

Using the 50 yr dose commitment factor of 3.05 mrem/pCi from reference 2 to obtain $(DF)_m$ for ^{239}Pu, and the other values as given, eq (4.2) gives a maximum concentration of

^{239}Pu in waste of 1.2 $\mu Ci/cm^3$. Concentrations in waste up to this value would result in doses to workers under the stated conditions of less than 500 mrem/yr from the exposure.

These example calculations are based on ^{239}Pu because it is one of the isotopes of main interest. However, any isotope could have been used in the analysis.

4.4.2 Application to Other Radionuclides

The above approach can be applied to all nuclides in the inventory for which dose rate conversion factors are available. The list of maximum concentrations, given in Table 4.4, are obtained using eq (4.2) and dose conversion factors from reference 2. The dose rate conversion factors $(DF)_m$ obtained from reference 2 are similar to the implicit conversion factors that can be obtained from the recommended concentration guides listed in reference 13. Therefore dose factors can be used from that source,[13] if not available for certain nuclides in reference 2.

4.4.3 Parametric Variations

The inadvertent or unknowing exposure to disposed wastes could occur as a result of several future actions. Assuming that disposal is by some near-surface method such as shallow land burial, possible courses of exposure to the wastes include efforts to reclaim the disposal site for productive use, such as housing, farming, or resource exploration. Archeological activities or salvage of apparently useful disposed items could also occur. Both the duration of the resultant exposures, and the amounts of buried waste involved can vary over large ranges. Some engineering judgement is required to select the most reasonable values to be used in any analysis of the effects of the potential reclamation events. Factors that are varied in this parametric analysis are the dust loading and the exposure time period.

Typical dust loadings around the country average about 40 $\mu g/m^3$ in rural areas and about 150 $\mu g/m^3$ in urban areas. Over 90% of all measurements are less than 300 $\mu g/m^3$. Plowing fields raises dust loadings up to 30 times the average values for farms.[14] Obviously, the wind speed and duration, orientation of the excavation and composition of the disposed wastes all influence the dust loading. There is an obvious correlation between dust loading and the probable exposure time because the higher the dust

TABLE 4.4

MAXIMUM CONCENTRATIONS FOR NUCLIDES FROM THE RECLAIMER INHALATION EVENT

Nuclide	Maximum Concentration ($\mu Ci/cm^3$)
^{3}H	10^{9} [a]
^{14}C	1.4 +6
^{55}Fe	10^{9}
^{60}Co	10^{9}
^{90}Sr	9.5 +3
^{99}Tc	5.9 +6
^{129}I	5.5 +2
^{135}Cs	2.1 +5
^{137}Cs	1.6 +6
^{235}U	6.2 +1
^{238}U	6.7 +1
^{237}Np	1.8
^{238}Pu	3.7
^{239}Pu	1.0
^{240}Pu	1.0
^{241}Pu	6.6 +4
^{242}Pu	1.1
^{241}Am	3.6
^{243}Am	3.1
^{242}Cm	10^{9}
^{244}Cm	1.9 +3

[a] A value of 10^9 is inserted whenever the maximum concentration is greater than 10^9 $\mu Ci/cm^3$. This is used consistently in all Tables of the report.

loading, the less time a person stays exposed because of physical discomfort, while a lower dust loading can be tolerated for a much longer period. The relationship between dust loadings, exposure times and resultant dose rates to the exposed individuals are shown in Figure 4.2.

Besides the base exposure of a few individuals involved in construction to an elevated dust loading, other exposure events can be postulated. If 200 homes are built on the site, with five occupants each, this 1000 person population may be exposed to dust from the wastes carried to the surface by construction activities. Presuming that significant stabilization of the exposed waste mixed with soil does not occur until one year has elapsed after initial occupation of the homes (typical time to get lawns in place), and that the waste in the dust is further diluted (by a factor of ten) with the clean surface soil and that the annual average dust loading from the wastes is 50 $\mu g/m^3$, a person who resided there full time for one year, breathing at a moderate activity rate would receive a dose rate of about 50 mrem/yr from buried wastes containing 1 $\mu Ci/cm^3$ of ^{239}Pu. Most of the 1000 people, of course, will not spend their full time outside in that area, and average breathing rates are lower than the value used. Therefore, the 50 mrem/yr is a very conservative value with the actual average value expected to be much less.

4.5 WELL WATER RECLAMATION EVENT

Another event which deserves attention is the use of contaminated groundwater from the aquifer immediately below the site. The maximum radionuclide concentrations in the aquifer would occur on the downgradient edge of the site shortly after the time of disposal, and are a function of the radionuclide leach rate constants and the nuclide inventory in the waste. The maximum radionuclide inventories in the waste are related to the dose guidelines by the following equation:

$$I_m = \frac{m_t D}{\lambda_L f_O U_a (DF)_m} \qquad (4.3)$$

where

m_t = total water flow in the aquifer = initial upgradient aquifer flow plus addition to flow from rainfall on the site (ℓ/yr)

D = dose rate guidelines (500 mrem/yr)

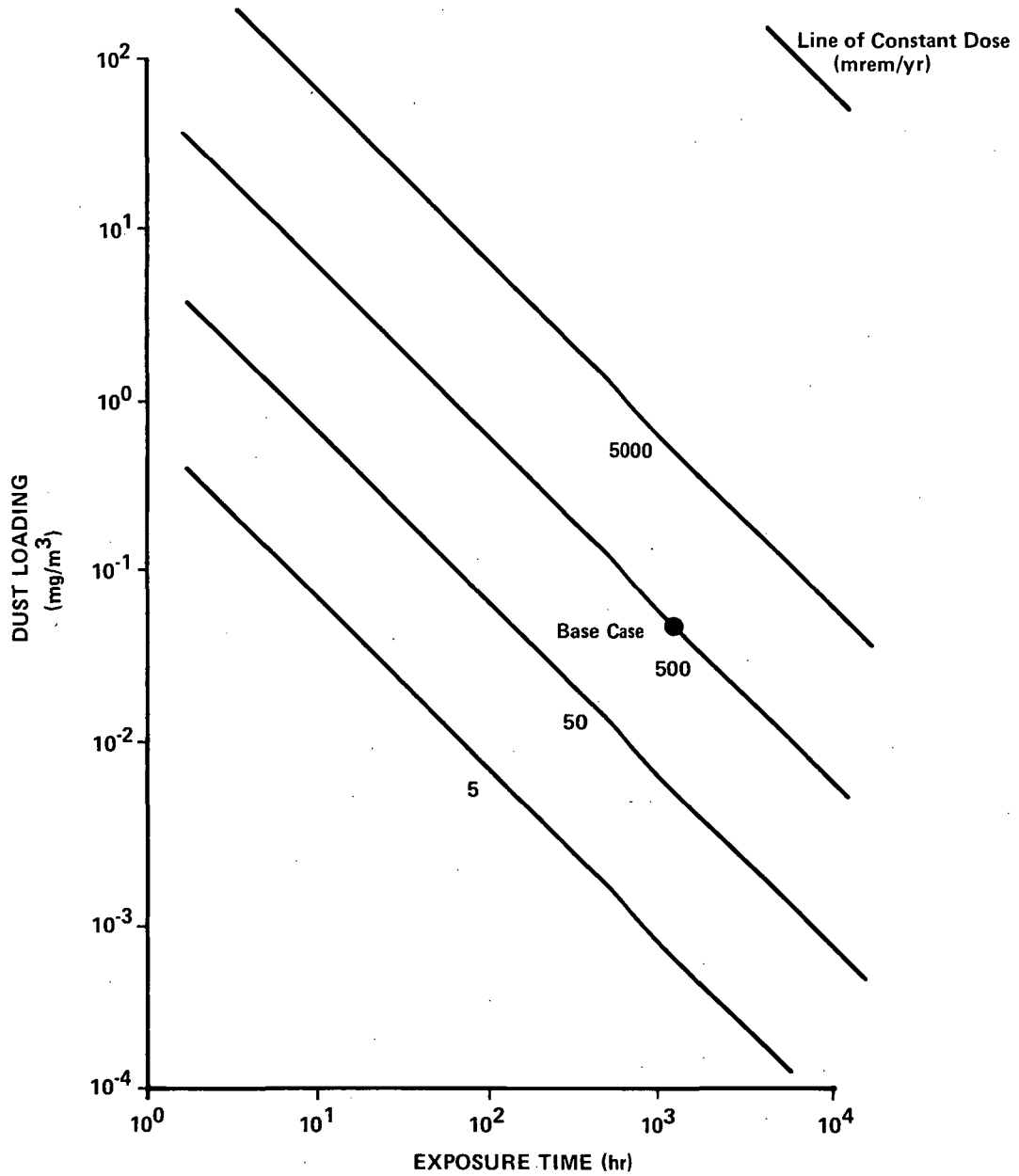


FIGURE 4.2

RELATIONSHIP BETWEEN DUST LOADING AND EXPOSURE TIME ON DOSE RATES AT CONSTANT CONCENTRATION LIMIT IN WASTES

I_m = initial inventory limitation (Ci)

λ_L = nuclide leach constant (yr^{-1})

f_0 = peak ratio of quantity of nuclide arriving in groundwater at well to that leaving wastes in first year

U_a = water consumption factor[2] (730 ℓ/yr)

$(DF)_m$ = dose conversion factor[2] (mrem/Ci)

The factor f_0 is obtained from groundwater migration model calculations. Details of this factor are discussed in Appendix B.1.

The concentrations in the well water, C_w, are given by:

$$C_w = \frac{\lambda_L I_o f_o}{m_t} \qquad (4.4)$$

The maximum concentrations that are allowed by the dose guidelines are given by:

$$C_w = \frac{D}{U_a (DF)_m} \qquad (4.5)$$

The well water concentrations and maximum nuclide inventories in the waste are listed in columns 2 and 3 of Table 4.5.

Maximum concentrations of nuclides from this scenario can be obtained for the RCF site parameters by dividing the site inventory by the waste volume (6.3×10^5 m^3) and multiplying by 10, the peak-to-average concentration in the waste. The resulting maximum concentrations are given in the last column of Table 4.5. Although this event actually limits the site inventory, the maximum concentrations listed, applicable specifically to the RCF, can be used to facilitate comparisons with limitations from the other exposure events.

The above approach results in conservative estimates of the concentrations in the aquifer at the well. For this event the time and spatial dependencies of waste disposal should be investigated in more detail to arrive at more precise estimates of the concentrations expected.

TABLE 4.5

MAXIMUM INVENTORIES FROM
WELL WATER EVENT

Nuclide	Concentration in Well (pCi/l)	Maximum Inventory (Ci)	Fraction of Maximum Inventory in 10^3 RRY of Waste	Maximum Concentration in Wastes ($\mu Ci/cm^3$)
^{3}H	6.5 +6	9.5 +5	4.8 -1	15
^{14}C	2.4 +5	9.0 +6	1.6 -4	140
^{55}Fe		1 +13		10^9
^{57}Co		1 +13		10^9
^{60}Co		1 +13		10^9
^{90}Sr	9.0 +1	8.2 +4	3.1 -2	1.3
^{99}Tc	1.1 +5	4.1 +6	2.0 -3	64
^{129}I	9.5 +1/7.4 +1	6.2 +3	6.5 -1/8.3 -4	9.8 -5/7.7 -2
^{135}Cs	3.5 +4	1.2 +6	3.7 -6	20
^{137}Cs		1 +13		10^9
^{235}U	8.5 +2	7.0 +5	5.7 -6	11
^{238}U	8.9 +2	7.3 +5	5.5 -5	12
^{237}Np	5.0 +2	1.8 +5	9.4 -6	2.9
^{238}Pu		1 +13		10^9
^{239}Pu	9.0 +2	5.7 +6	4.0 -6	90
^{240}Pu	9.0 +2	5.1 +7	2.3 -6	8.1 +2
^{241}Pu		1 +13		10^9
^{242}Pu	9.5 +2	8.1 +5	3.0 -7	13
^{241}Am		1 +13		10^9
^{243}Am	8.4 +2	3.8 +7	4.5 -6	6.0 +2
^{242}Cm		1 +13		10^9
^{244}Cm		1 +13		10^9

4.6 DIRECT GAMMA EXPOSURE

For some gamma-emitting radionuclides, the limiting concentration is associated with the reclaimer digging into the waste or living on the waste and receiving an external gamma dose. As shown in Appendix B.2, the equation relating the limiting concentration in the waste, (C_m) to the resulting gamma exposure is:

$$C_m = \frac{2\mu D}{(0.0575)G(\mu/\rho)_t E_m T_x f} \exp(150\lambda_m) \qquad (4.6)$$

where

μ = effective gamma ray attentuation coefficient for soil (cm^{-1})

G = gamma emission rate per μCi of radionuclide (γ/sec/μCi)

$(\mu/\rho)_t$ = mass absorption coefficient for tissue (cm^2/g)

E_m = average energy of the emitted gamma rays (MeV)

and the other terms as defined previously.

Table 4.6 contains maximum concentrations of the major gamma-emitting radionuclides in the waste for the two cases. In the first case, the worker is directly exposed to the waste for one-sixth of the time he is working on site (170 hours). In the second case the exposed individuals are continually on site, but are shielded by at least one meter of earth cover. This earth cover has a significant effect on the exposures and maximum concentrations, so that these maximum concentrations are not limiting. Comparison of these limiting maximum concentrations with the average radionuclide mix in the RCF reveals that ^{137}Cs is the dominant contributor to the total gamma exposure for times less than about 500 years, then several transuranics become dominant.

4.7 ATMOSPHERIC RELEASES OF CONTAMINATION

The atmospheric release events occur during the handling of the waste. The events include off-site exposures to the public from a single container accident and continuous spillage during operation.

TABLE 4.6

MAXIMUM CONCENTRATIONS OF RADIONUCLIDES
FROM DIRECT EXPOSURE EVENTS

Nuclide	Max Concentration for Worker ($\mu Ci/cm^3$)	Max Concentration for Resident ($\mu Ci/cm^3$)
^{60}Co	5.5 +4	5.9 +7
^{90}Sr	1.5 +3	1.6 +6
^{129}I	35	10^9
^{137}Cs	8.3	3.7 +6
^{237}Np	9.1	10^9
^{235}U	4.1	10^9
^{238}U	59	10^9
^{238}Pu	3.1 +3	10^9
^{240}Pu	1 +2	10^9
^{241}Am	23	10^9
^{243}Am	5.1	10^9
^{244}Cm	7.7 +3	10^9

4.7.1 Single Container Accident

The single container accident is defined as the instantaneous release of a fraction of the contents of a container of waste during the burial operation. It is assumed that the wastes are packaged in 55-gallon drums which are presently the most common containers. The major pathway of concern for dose to non-occupational personnel is the airborne transport of the material released from the drum to the site boundary. The equation relating the maximum concentration of waste in the drum to the dose rate to a maximum individual is:

$$C_m = \frac{D}{V_c f_r (\chi_t/Q_t) U_a (DF)_m} \qquad (4.7)$$

where

D = allowed dose rate (500 mrem/yr)

V_c = volume of container (2.1×10^5 cm^3)

f_r = fractional release from the barrel (10^{-3})

χ_t/Q_t = normalized concentration-time exposure (sec/m^3)

U_a = breathing rate (3.3×10^{-4} m^3/sec)

$(DF)_m$ = dose rate conversion factor for nuclide (mrem/yr/μCi)

The term (χ_t/Q_t) is discussed in Appendix B.3 and is obtained from particulate transport calculations for an instantaneous point source. For a transport distance of 160 m to the fenceline, a windspeed of 1.56 m/sec (3.5 mph), a Pasquill F stability factor, (χ_t/Q_t) is equal to 2.5×10^{-3} sec/m^3. Inserting this value into eq (4.7) gives, for ^{239}Pu:

$$C_m = 100 \ \mu Ci/Cm^3 \qquad (4.8)$$

Therefore, this event gives a maximum ^{239}Pu concentration that is about two orders of magnitude less restrictive than

the reclamation inhalation scenerio as calculated by eq (4.2). However, this event is more restrictive than the reclamation inhalation event for isotopes with short half-lives such that $\exp(100\ \lambda_m)$ is greater than 100. That is, the accident takes place before decay reduces the concentrations of the short half-life nuclides.

The application to all nuclides in the inventory is accomplished in the same manner described in Chapter 4.4.2.

4.7.2 Continuous Operational Releases

In this event it is postulated that drums have small leaks, surface contamination or are ruptured occasionally and that the ensuing minor release is then transported to the site boundary.

The following assumptions have been used:

1. About 10^{-7} of the waste becomes airborne.
2. The exposure occurs at the site boundary (160 m).
3. A Pasquill "D" stability level is used with an isotropic windrose and an average windspeed of 1.56 m/sec.
4. The site is operated for 40 years.

The equation relating the maximum concentrations in the waste with other relevant parameters is:

$$C_m = \frac{DYP}{f_{rc}(\chi/Q)VU_a(DF)_m} \qquad (4.9)$$

where

P = peak-to-average nuclide concentrations in the waste (10)

Y = years of site operation (40 yrs)

f_{rc} = fractional waste release (10^{-7})

V = waste volume (6.3×10^{11} cm^3)

U_a = breathing rate (8×10^6 ℓ/yr)

$(DF)_m$ = dose conversion factors from reference 2

D = allowed dose rate (500 mrem/yr)

The term (χ/Q) is the normalized transport concentration. Details of its determination are given in Appendix B.4.

Factors for the dose commitment are used in eq (4.9) because the exposure is continual over an extended period of time.

Substituting the value of DF for ^{239}Pu into eq (4.9) yields a C_m equal to 250 $\mu Ci/cm^3$. This is 2.5 times larger than the maximum concentrations from the single container accident event. Because the calculation for other isotopes proceeds the same as in the single container accident event, all maximum concentrations calculated for the continuous release event will be 2.5 times larger than those given in Table 4.7.

4.8 GROUNDWATER MIGRATION

In this section the migration of radionuclides through underground aquifers is investigated. In the groundwater events, the nuclides are assumed to be leached by rainwater and then migrate by saturated flow to a confined aquifer 10 m below the RCF, where they then migrate 1000 m to a surface river where the aquifer discharges. The river has a postulated flow of 500 m^3/sec. The dose rate to maximum and average individuals are related to the concentration in the river, and hence, to the nuclide inventory and leach rate. The limiting parameter for this event is the nuclide inventory. This, however, can also be expressed as a maximum concentration using the volume of the RCF, as described in Chapter 4.5. The maximum inventory from this event is given by eq (4.3), where all parameters retain their same values except m_t and f_0. The m_t parameters are now equal to the annual river flow (1.6×10^{13} ℓ/yr), and f_0 is generally smaller because the aquifer distance is longer. Since m_t is many orders of magnitude greater than the total aquifer flow rate, and since f_0 is reduced or the same and all other parameters remain the same, this groundwater event is not limiting for any nuclides. However, this event does have the potential for causing the largest population dose. Therefore, it will be considered in more detail. The last column in Table 4.8 lists the leach constants used. Very little information is available on leach constants for several of the nuclides. To obtain leach constants for a few reference nuclides it was necessary in this investigation to evaluate soil samples taken below actual waste burial pits at Idaho National Engineering Lab INEL[15] and to assume that nuclide contamination resulted from the migration of nuclides leached according to the following expression:

TABLE 4.7

MAXIMUM CONCENTRATIONS FOR NUCLIDES FROM THE SINGLE CONTAINER ACCIDENT

Nuclide	Maximum Concentration ($\mu Ci/cm^3$)
^{3}H	10^{9}
^{14}C	1.4 + 8
^{55}Fe	3.3 + 7
^{60}Co	4.0 + 5
^{90}Sr	2.5 + 4
^{99}Tc	5.8 + 8
^{129}I	5.5 + 4
^{135}Cs	2.0 + 7
^{137}Cs	4.0 + 6
^{235}U	6.2 + 3
^{238}U	6.7 + 3
^{237}Np	1.8 + 2
^{238}Pu	1.1 + 2
^{239}Pu	1.0 + 2
^{240}Pu	1.0 + 2
^{241}Pu	5.0 + 3
^{242}Pu	1.1 + 2
^{241}Am	3.0 + 2
^{243}Am	3.0 + 2
^{242}Cm	8.2 + 3
^{244}Cm	5.2 + 2

TABLE 4.8

CHARACTERISTICS OF NUCLIDES IN RCF INVENTORY

Nuclide	Sorption Coefficient (K)	Half-Life (yr)	Decay Constant, λ_m (yr^{-1})	Leach Constant, λ_L (yr^{-1})
^{3}H	1	1.23 +1	5.62 -2	1 -1
^{14}C	1 +1	5.73 +3	1.21 -4	1 -4
^{55}Fe	3.3 +3	2.70	2.57 -1	1 -1
^{60}Co	3.3 +3	5.3	1.32 -1	1 -1
^{90}Sr	1 +2	2.9 +1	2.43 -2	1 -2
^{99}Tc	1	2.13 +5	3.25 -6	1 -4
^{129}I	1	1.59 +7	4.36 -8	1 -1
^{135}Cs	1 +3	2.3 +6	3.01 -7	1 -3
^{137}Cs	1 +3	3.01 +1	2.3 -2	1 -3
^{235}U	1.4 +4	7.04 +8	9.85 -10	1 -5
^{238}U	1.4 +4	4.47 +9	1.55 -10	1 -5
^{237}Np	1 +2	2.14 +6	3.24 -7	1 -5
^{238}Pu	1 +4	8.78 +1	7.89 -3	1 -5
^{239}Pu	1 +4	2.44 +4	2.84 -5	1 -5
^{240}Pu	1 +4	6.54 +3	1.06 -4	1 -5
^{241}Pu	1 +4	1.5 +1	4.62 -2	1 -5
^{242}Pu	1 +4	3.87 +5	1.79 -6	1 -5
^{241}Am	1 +4	4.33 +2	1.60 -3	1 -5
^{243}Am	1 +4	7.37 +3	9.40 -5	1 -5
^{242}Cm	3.3 +3	4.5 -1	1.55	1 -5
^{244}Cm	3.3 +3	1.79 +1	3.87 -2	1 -5

$$L_r = \lambda_L I_m \exp(-\lambda_E t) \quad (4.10)$$

where

L_r = leach rate (Ci/yr)

$\lambda_E = \lambda_L + \lambda_m$

Table 4.9 contains maximum release rates to the surface river and maximum nuclide inventories obtained from eq (4.5). Also given in the table is the ratio of the reference radionuclide inventory to the maximum inventory. Nuclides not listed in the table have negligible releases to the river. The dose conversion factors used in these calculations are obtained from reference 2.

As stated previously the limiting parameter is the site inventory. Maximum concentrations can be determined from the maximum site inventory. These maximum concentrations also are given in the table.

4.8.1 Sensitivity Analysis

There is considerable uncertainty in the accuracy of the parameters used in this analysis. A sensitivity analysis of the main parameters yields important information about the effect these uncertainties have on the release rate. Because ^{239}Pu received major attention previously, it was selected as one of the nuclides for the sensitivity study. The other nuclide, ^{129}I, was selected because it is also of potential concern.

The parameters that are varied in the analyses are the leach constant, the dispersion coefficient, the sorption coefficient, the pit-to-aquifer and aquifer lengths and the pit-to-aquifer and aquifer water velocities. The reference inventory was used in all these parametric studies. The ranges of the parameters varied are shown in Tables 4.10 and 4.11. For ^{239}Pu those parameters which have a major effect on the release rate are: λ_L, K, V_{aq}, and x_{aq}. The effect of varying K on the magnitude, shape and time dependence on the concentration release is shown in Figure 4.3. The magnitude of the pulse decreases significantly when the transit time becomes much larger than the half-life.

For ^{129}I, the only parameter having a major effect on the release rate is λ_L. This is because the transit time through the aquifer is significantly shorter than the half-life of ^{129}I.

TABLE 4.9

MAXIMUM INVENTORIES FROM GROUNDWATER MIGRATION

Nuclide	Time For Max. Value at Release Point (yr)	Maximum Site Inventory (Ci)	Fraction of Maximum Inventory To Ref. Inventory	Maximum Concentration In The Waste ($\mu Ci/cm^3$)
^{3}H	1.1 +1	1.1 +10	4 -5	1.7 +5
^{14}C	1.2 +2	5.0 +7	2.8 -5	7.9 +2
^{99}Tc	1.2 +1	9.5 +6	8.4 -4	1.5 +2
^{129}I	1.2 +1	1.9 +4	2.1 -4	0.3
^{135}Cs	1.2 +4	7.6 +5	6.1 -6	12
^{235}U	1.6 +5	7.6 +7	5.3 -8	1.2 +3
^{238}U	1.6 +5	8.8 +7	4.5 -8	1.4 +3
^{237}Np	1.6 +3	4.0 +7	4.3 -8	6.4 +2
^{239}Pu	1.2 +5	2.2 +9	1.0 -9	3.5 +4
^{240}Pu	1.2 +5	1.8 +13	6.7 -12	2.8 +8
^{242}Pu	1.2 +5	1.2 +8	2.0 -9	1.9 +3
^{243}Am	1.2 +5	2.9 +12	5.9 -11	4.6 +7

TABLE 4.10

RESULTS OF PARAMETRIC VARIATIONS FOR ^{239}Pu

Description	Leach Constant (yr^{-1})	Dispersion Coefficient (m^2/yr)	Sorption Coefficient	Pit-to-Aquifer Distance (m)	Pit-to-Aquifer Velocity (m/yr)	Aquifer Length (m)	Aquifer Velocity (m/yr)	Aquifer Release Rate (Ci/yr)	Normalized Aquifer Release Rate, (F)
Base Case	1 -5	1 +1	1 +4	1 +1	1 +1	1 +3	1 +2	5.0 -3	3.8 -2
λ_L = 1 -4	1 -4							1.8 -3	1.4 -2
λ_L = 1 -6	1 -6							4.8 -3	3.7 -2
D = 1	1 -5	1 +0						6.6 -3	5.1 -2
D = 5		5 +0						5.6 -3	4.3 -2
D = 2 =1		2 +1						4.4 -3	3.4 -2
K = 1 +2		1 +1	1 +2					1.3 -1	1
K = 1 +3			1 +3					9.2 -2	7.1 -1
x_{pa} = 2 +1			1 +4	2 +1				2.4 -3	1.8 -2
x_{pa} = 5 +1				5 +1				8.6 -4	6.6 -3
V_{pa} = 5 +1				1 +1	5 +1			6.0 -3	4.6 -2
V_{pa} = 1 +2					1 +2			6.4 -3	4.9 -2
x_{ar} = 1 +2					1 +1	1 +2		7.4 -2	6.7 -1
x_{ar} = 2 +3						2 +3		1.8 -4	1.4 -3
V_{ar} = 1 +1						1 +3	1 +1	2.0 -13	1 -12
V_{ar} = 5 +2							5 +2	4.8 -2	3.7 -3

TABLE 4.11

RESULTS OF PARAMETRIC VARIATIONS FOR ^{129}I

Description	Leach Constant (yr^{-1})	Dispersion Coefficient (m^2/yr)	Sorption Coefficient	Pit-to-Aquifer Distance (m)	Pit-to-Aquifer Velocity (m/yr)	Aquifer Length (m)	Aquifer Velocity (m/yr)	Aquifer Release Rate (Ci/yr)	Normalized Aquifer Release Rate
Base Case	1 -1	1 +1	1	1 +1	1 +1	1 +3	1 +2	5.6 -4	0.82
λ_L = 1	1							2.1 -3	0.29
λ_L = 1 -2	1 -2							7.0 -4	1.00
D = 2	1 -1	2						6.3 -4	0.89
D = 2 +1		2 +1						5.2 -4	0.76
K = 1 +1		1 +1	1 +1					3.1 -4	0.44
x_{pa} = 2 +1			1	2 +1				5.6 -4	0.78
x_{pa} = 5 +1				5 +1				5.2 -4	0.74
V_{pa} = 5 +1				1 +1	5 +1			6.6 -4	0.94
V_{pa} = 1 +2					1 +2			6.6 -4	0.94
x_{ar} = 1 +2					1 +1	1 +2		5.9 -4	0.85
x_{ar} = 2+3						2 +3		3.5 -4	0.52
V_{ar} = 1 +1						1 +3	1 +1	3.8 -4	0.57
V_{ar} = 5 +1							5 +2	5.9 -4	0.86

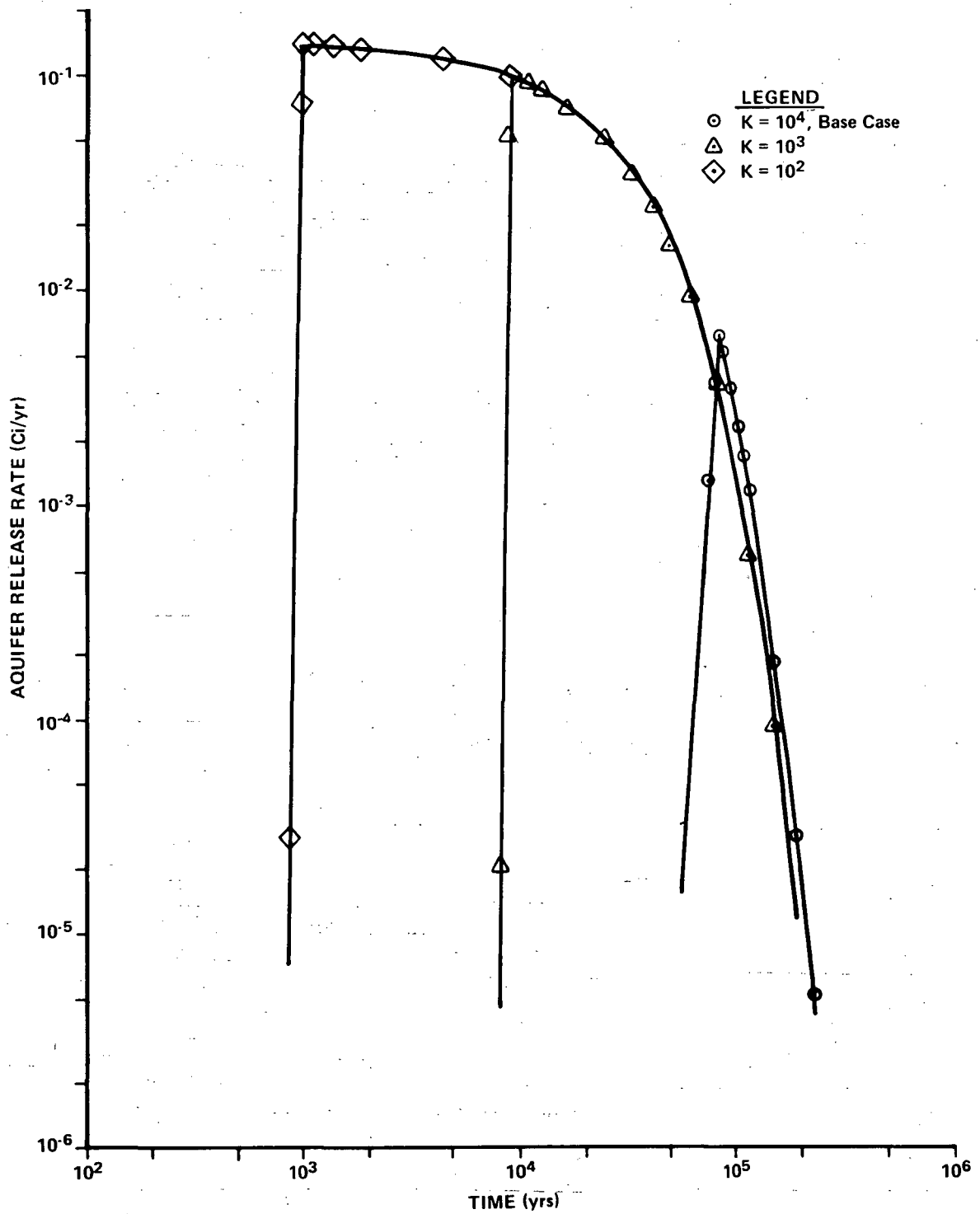


FIGURE 4.3

AQUIFER RELEASE RATE FOR ^{239}Pu AS A FUNCTION OF TIME

The sensitivity analysis can be generalized by utilizing nondimensional parameters. It has been shown[16] that the maximum value of the release rate (neglecting dispersion) can be expressed in the following form:

$$F = \exp(-Ro) \tag{4.11}$$

where:

$$Rp = \text{peak aquifer release rate (Ci/yr)} \tag{4.12}$$

$$F = Rp/\lambda_m I_m$$

$$Ro = \lambda_m Kx/V = \text{nondimensional time for the peak release} \tag{4.13}$$

The results of the parametric variation for ^{239}Pu are shown in Figure 4.4, in nondimensional form. The base case calculation is identified by the square, the diamonds represent the parametric variation computations using $\lambda_L = 10^{-5}\ yr^{-1}$; the circles represent the variations for $\lambda_L = 10^{-4}\ yr^{-1}$; and the curve is from eq (4.11). The value for $\lambda_L = 10^{-6}\ yr^{-1}$ is also shown in the graph. Even though there is still a small dependence of F upon the leach constant, the dominant behavior is described by the variation with Ro. The magnitude of F changes from unity very little for Ro << 1; however, as Ro increases beyond unity, F decreases rapidly. This behavior applies to other nuclides and other basic parameters. For example, the parametric variation of ^{129}I is shown in Figure 4.5. Since all values of Ro investigated are significantly less than one, F changes very little over the two orders of magnitude for Ro. One point not lying close to the curve from eq (4.11) is a value of F = 0.29 at Ro = 4.8×10^{-7}. This point represents the case for $\lambda_L = 1$, so that the inventory is severely depleted in the first year, and the $\lambda_L I_m$ normalization overestimates the effect of the leaching constant on the release rate, resulting in a low value for F.

Equation (4.12) can also be applied to a multicomponent or heterogeneous aquifer system such as the present groundwater pathway, by using the appropriate average values for K, x and V. For an "n" component system the parameters are:[17]

$$\bar{x} = \sum_{i=1}^{n} x_i \tag{4.14}$$

$$f_i = x_i/\bar{x} \tag{4.15}$$

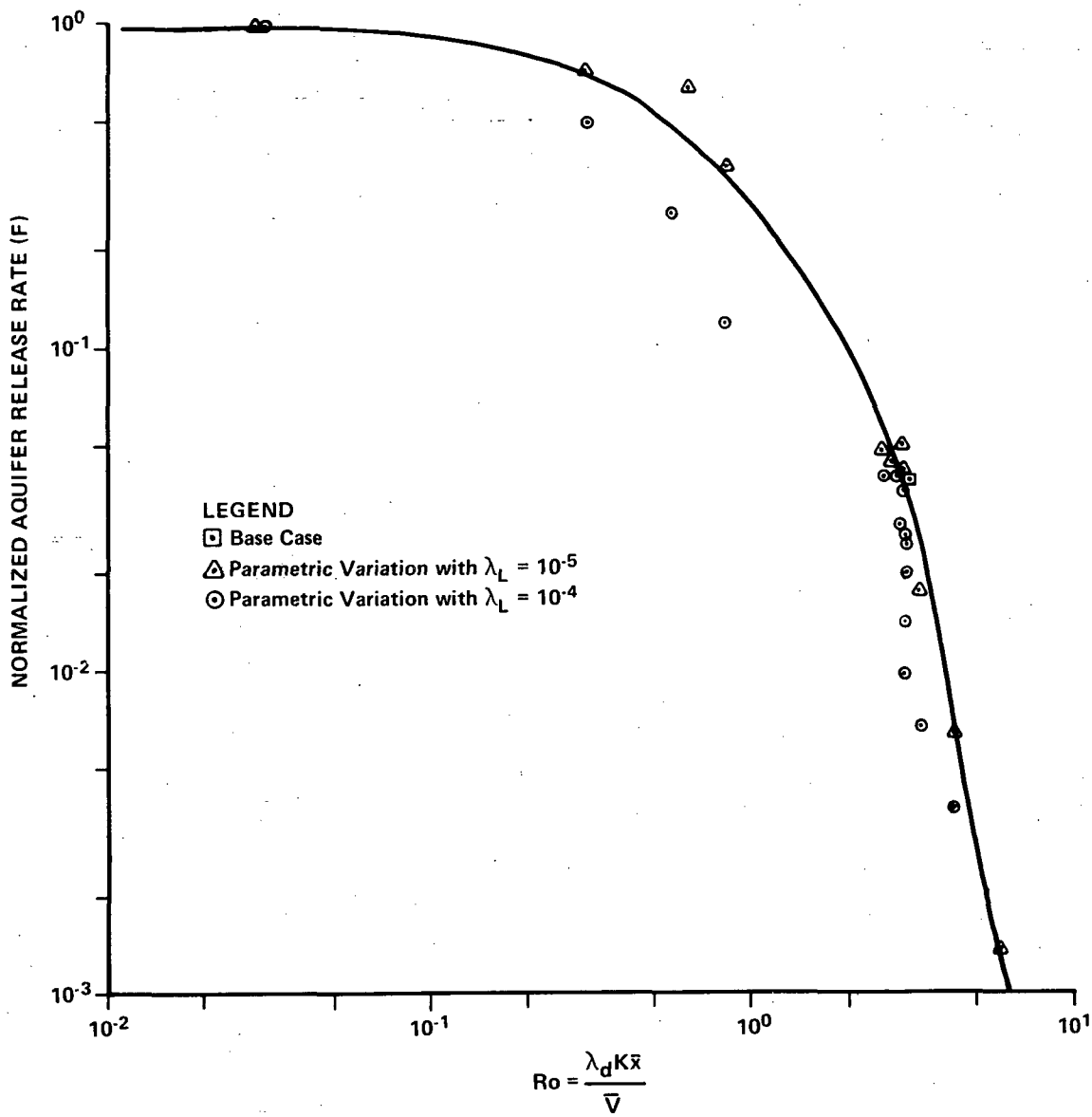


FIGURE 4.4

NORMALIZED AQUIFER RELEASE RATE FOR ^{239}Pu AS A FUNCTION OF THE DIMENSIONLESS PARAMETER, Ro

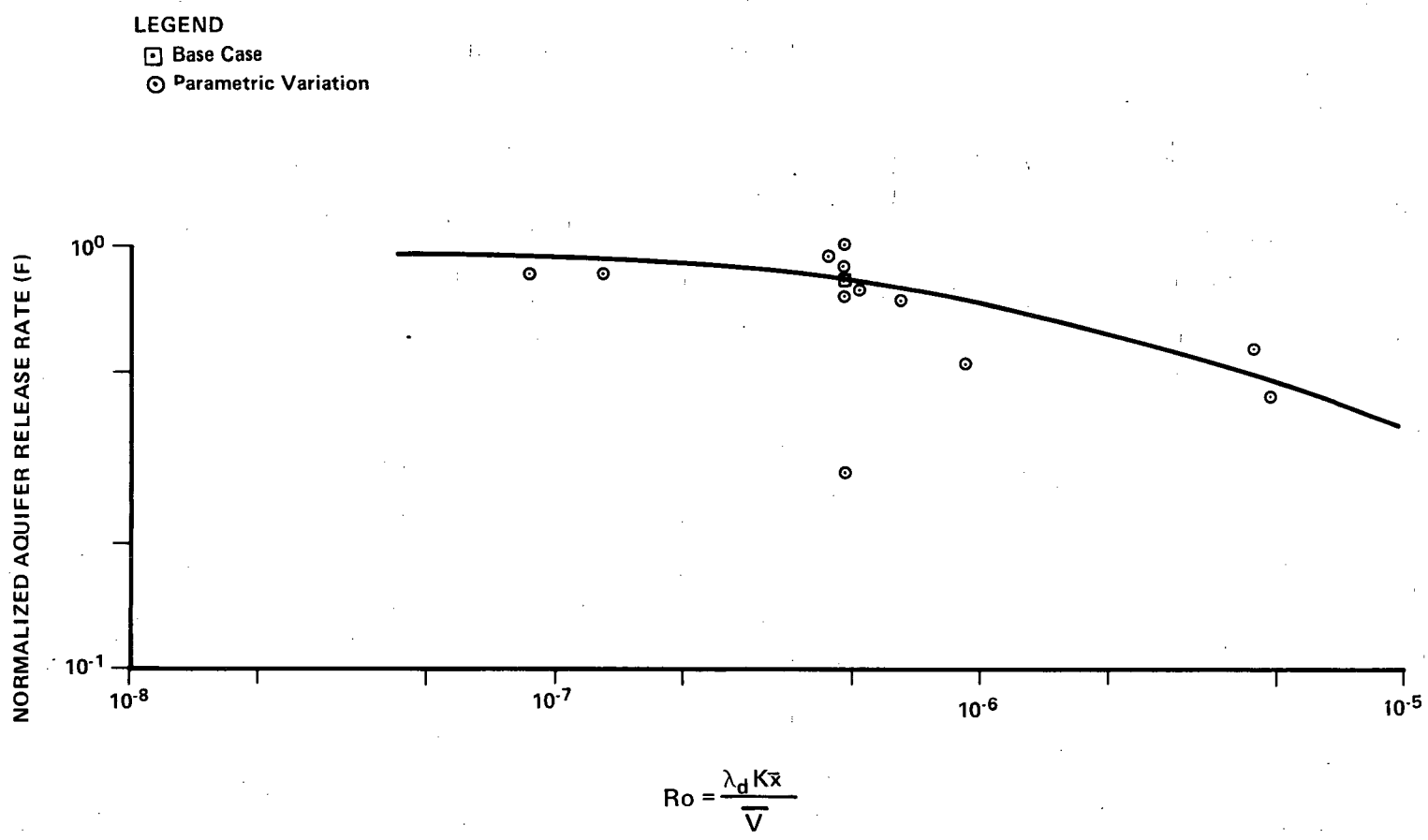


FIGURE 4.5

NORMALIZED AQUIFER RELEASE RATE FOR ^{129}I AS A FUNCTION OF THE DIMENSIONLESS PARAMETER, Ro

$$\bar{V} = \left(\sum_{i=1}^{n} \frac{f_i}{V_i} \right)^{-1} \tag{4.16}$$

and

$$\bar{K} = \bar{V} \sum_{i=1}^{n} \left(\frac{f_i K_i}{V_i} \right) \tag{4.17}$$

Then eq (4.11) becomes:

$$Ro = \lambda_m \bar{K}\bar{x}/\bar{V} \tag{4.18}$$

Equation (4.18) was used on the present two-component system to formulate the values in Figures 4.4 and 4.5.

It should be noted that the above equations and the dimensionless representation in Figures 4.4 and 4.5 are very general and can be applied to any atom.

4.8.2 Dual Component Plutonium

The parametric analysis provides the data to estimate the effects from a dual component of plutonium. For example, if 10% of the ^{239}Pu in an inventory can migrate with a K = 100, instead of 10,000, then the maximum release rate for the K = 100 case is 0.065 Ci/yr in about 1,200 years. The K = 10,000 component will have a maximum release rate of about 0.023 Ci/yr in about 120,000 years. In this particular example, the total peak release rate is just 0.065 Ci/yr because the release of the two components is greatly separated in time. In the lower leach rates, however, the ^{239}Pu is present in the groundwater for a longer period of time and the magnitude of the peak release rate could be larger than the peak rate of either component.

4.9 INGESTION OF FOOD PRODUCED ON THE DISPOSAL SITE

Some contamination of the surface soil could eventually result from reclamation activities at the RCF. Vegetables possibly could be grown in the contaminated soil, or milk cows or beef raised on contaminated grass. Consumption of these foodstuffs would then result in exposures to the few individuals involved in eating the produce. Equation (4.19) gives the relationship between the maximum allowable concentration of radioactivity in the wastes and the dose guideline, dose conversion factor, and consumption and uptake factors for each nuclide:

$$C_m = \frac{D\ f_1\ f_2\ e^{+\lambda m T}}{(DF)_m\ B_{mv}(U_{ap}^{meat}\ F_f Q_a + U_{ap}^{milk}\ F_m Q_a + U_{ap}^{veg})\ f_3} \tag{4.19}$$

where

C_m = the maximum allowable concentration in wastes for the mth nuclide ($\mu Ci/cm^3$)

D = the dose guideline value (500 mrem/yr)

λ_m = the radioactive decay constant for the mth nuclide (yr^{-1})

T = the period of institutional control (150 yr)

f_1 = effective dilution factor including maximum to average concentrations in waste and dilution during burial (20)

f_2 = mixing factor for buried materials transferred to surface and intermingled with clean soil at surface (10)

$(DF)_m$ = dose conversion factor from reference 2 for m^{th} nuclide (mrem/pCi)

B_{mv} = vegetative bioaccumulation and uptake factor for m^{th} nuclide by v^{th} plant from reference 2 (concentration in vegetable/concentration in soil)

U^i_{ap} = usage factors from reference 2 (U_{ap}^{milk} = 310 ℓ/yr; U_{ap}^{meat} = 110 kg/yr; $U_{ap}^{vegetables}$ = 520 kg/yr)

Q_a = animal consumption rate from reference 2 (50 kg/day)

F_f = stable element transfer coefficient relating animal consumption rate to concentration in edible meat, from reference 2 (day/kg)

F_m = stable element transfer coefficient relating animal consumption rate to concentration in milk from reference 2 (day/ℓ)

f_3 = fraction of annual food consumption produced on site (0.5)

Table 4.12 contains a list of the maximum allowable concentrations for nuclides in waste based on this pathway. For those nuclides with half-lives small compared to 150 years, this pathway again is not limiting. The maximum allowable concentrations for ^{129}I were calculated for both thyroid and whole body doses (see chapter 4.11).

TABLE 4.12

MAXIMUM ALLOWABLE CONCENTRATIONS
FOR FOOD PATHWAY

Nuclide	MAC ($\mu Ci/cm^3$)
^{3}H	4.6 +3
^{14}C	2.4 -2
^{55}Fe	10^9
^{60}Co	5.2 +8
^{57}Co	10^9
^{90}Sr	0.17
^{99}Tc	0.96
^{129}I	3.6 -3/2.8
^{135}Cs	1.9
^{137}Cs	12.7
^{235}U	0.30
^{238}U	0.31
^{237}Np	0.28
^{238}Pu	10.4
^{239}Pu	3.2
^{240}Pu	3.2
^{241}Pu	1.55 +5
^{242}Pu	3.4
^{241}Am	3.9
^{243}Am	3.0
^{242}Cm	10^9
^{244}Cm	1.66 +3

4.10 SURFACE EROSION

It is reasonably conservative to assume that future containment facilities will not be sited in areas where substantial erosion is likely to occur. However, if the RCF were located in an area where wind or water erosion were occurring, some contamination could ultimately be released to surface waters or dispersed into the atmosphere. Design features, such as covering the filled burial area with pebbles, through which vegetation could be established, would tend to minimize erosion processes.

For the sake of determining whether erosion may be a limiting event, a straightforward, conservative calculation based on a representative erosion rate was performed. There are a number of site specific parameters influencing erosion rates. Some of these are surface slope, amount of precipitation, distances to watercourses, distances from peaks, amount and type of vegetation, and soil properties. However, six tons of soil per acre per year is a typical sheet erosion rate.[18] Using this rate and soil density of 1.6 gm/cm^3, it will require 1200 years for one meter surface cover to be eroded away before the buried wastes begin to erode.

The maximum inventory in the waste for sheet erosion of the wastes into the river is given by eq (4.20). A factor of 10 dilution of the waste with clean soil is assumed:

$$I_m = \frac{D\,\dot{m}_t\,d}{F_e\,Yo\,(DF)_m} \exp\,(1200\lambda_m) \qquad (4.20)$$

where

d = dilution with clean dirt (10)

F_e = fraction of waste eroded from the site per year (2.7×10^{-3})

and other parameters are as defined earlier. The maximum inventories are given in Table 4.13. This event is not limiting for any isotope.

4.11 MAXIMUM ALLOWABLE CONCENTRATIONS

Three of the seven events in the set given in Chapter 3.2 yield limitations on the maximum concentrations, and three of the events yield inventory limitations that were

TABLE 4.13

MAXIMUM INVENTORIES FROM SHEET EROSION

Nuclide	Decay Constant (yr^{-1})	Maximum Inventory (Ci)	Maximum Concentration ($\mu Ci/cm^3$)
^{14}C	1.21 -4	1.6 +9	2.6 +4
^{99}Tc	3.26 -6	6.3 +8	1 +4
^{129}I	4.36 -8	5.7 +5/4.4 +8	9 / 7 +3
^{135}Cs	3.01 -7	2.1 +8	3.3 +3
^{235}U	9.85 -10	5 +6	80
^{238}U	1.55 -10	5.4 +6	85
^{237}Np	3.24 -7	3.2 +6	50
^{238}Pu	7.89 -3	8.2 +10	1.3 +6
^{239}Pu	2.84 -5	5.7 +6	90
^{240}Pu	1.06 -4	6.3 +6	1 +2
^{242}Pu	1.79 -6	5.7 +6	90
^{241}Am	1.60 -3	3.5 +7	5.5 +2
^{243}Am	9.40 -5	5.7 +6	90

related to maximum concentrations specific to the RCF. The Maximum Allowable Concentration (MAC) is defined as the most restrictive concentration given by the set of events. The MAC's are listed in Table 4.14. Those MAC's in parenthesis are the maximum concentrations associated with the inventory limitations and, as such, are specific to the RCF. They are, however, a conservative application of the inventory limits and provide a very consistent, practical basis for waste classification.

As previously noted, thyroid and whole body doses were calculated for ^{129}I. The MAC associated with the whole body dose is listed in the table because it takes six kg of ^{129}I to make one curie. Therefore, as calculated by Rodger[19], if the ratio of ^{129}I to ^{127}I in the thyroid is 2% or less, it is not possible to exceed the annual permissible thyroid dose. In any real situation, an individual's iodine consumption will be from many sources, making the contribution from a LLW disposal site a small percentage of his bodily intake. It appears, therefore, that the thyroid dose limitation is probably not dominant, so the whole body dose limitation was used.

Therefore, except for ^{55}Fe, ^{241}Pu and ^{243}Cm, the limiting events are those in which exposures were received at the site by reclaimers.

It should be noted that, although the transuranic isotopes are generally considered the most hazardous, the MAC's for the transuiranics are less restrictive than for the MAC's of the majority of non-transuranic isotopes considered.

4.11.1 Relative Impacts from the Events

An indication of the relative impact of each event is obtained by plotting the potential dose rate as a function of the nuclide concentration for each event. The result for ^{239}Pu is shown in Figure 4.6. The reclaimer-inhalation event is limiting, followed closely by the food pathway event. Next in magnitude are the single container, well water and erosion events, followed by continuous operational releases and then groundwater events. The impact from direct gamma occurs only from the atomic x-ray emission, which was not considered in this study.

The relative impacts for ^{90}Sr are shown in Figure 4.7. For this nuclide, food pathway is the most restrictive, followed by the well water and reclaimer-inhalation. Finally, the single container event and the continuous operational release events are shown. The impacts from groundwater and erosion are so small they were not included in this graph.

TABLE 4.14

MAXIMUM ALLOWABLE CONCENTRATIONS FOR NUCLIDES ($\mu Ci/cm^3$)

Nuclide	MAC From Reclaimer Inhalation	MAC From Well Water	MAC From Direct Exposure	MAC From Single Container Accident	MAC From Ground Water	MAC From Food Pathway	MAC From Sheet Erosion	Maximum Allowable Concentration
^{3}H	10^{9}	1.5 +1	--	10^{9}	1.7 +5	4.6 +3	--	(15)
^{14}C	1.4 +6	1.4 +2	--	1.4 +8	7.9 +2	2.4 -2	2.6 +4	2.4 -2
^{55}Fe	10^{9}	10^{9}	--	3.3 +7	--	10^{9}	--	3.3 +7
^{60}Co	10^{9}	10^{9}	5.5 +4	4.0 +5	--	5.2 +8	--	5.5 +4
^{90}Sr	9.5 +3	1.3	1.5 +3	2.5 +4	--	1.7 -1	--	1.7 -1
^{99}Tc	5.9 +6	6.4 +1	--	5.8 +8	1.5 +2	9.6 -1	1.0 +4	9.6 -1
^{129}I	5.5 +2	9.8 -5/7.7 -2	3.5 +1	5.5 +4	3.0 -1	3.6 -3/2.8	9.0/7.0 +3	(9.8 -5/7.7 -2)
^{135}Cs	2.1 +5	2.0 +1	--	2.0 +7	1.2 +1	1.9	3.3 +3	1.9
^{137}Cs	1.6 +6	10^{9}	8.3	4.0 +6	--	1.3 +1	--	8.3
^{235}U	6.2 +1	1.1 +1	4.1	6.2 +3	1.2 +3	3.0 -1	8.0 +1	3.0 -1
^{238}U	6.7 +1	1.2 +1	5.9 +1	6.7 +3	1.4 +3	3.1 -1	8.5 +1	3.1 -1
^{237}Np	1.8	2.9	9.1	1.8 +2	6.4 +2	2.8 -1	5.0 +1	2.8 -1
^{238}Pu	3.7	10^{9}	3.1 +3	1.1 +2	--	1.0 +1	1.3 +6	3.7
^{239}Pu	1.0	9.0 +1	--	1.0 +2	3.5 +4	3.2	9.0 +1	1.0
^{240}Pu	1.0	8.1 +2	1.0 +2	1.0 +2	2.8 +8	3.2	1.0 +2	1.0
^{241}Pu	6.6 +4	10^{9}	--	5.0 +3	--	1.6 +5	--	5.0 +3
^{242}Pu	1.1	1.3 +1	--	1.1 +2	1.9 +3	3.4	9.0 +1	1.1
^{241}Am	3.6	10^{9}	2.3 +1	3.0 +2	--	3.9	5.5 +2	3.6
^{243}Am	3.1	6.0 +2	5.1	3.0 +2	4.6 +7	3.0	9.0 +1	3.0
^{242}Cm	10^{9}	10^{9}	--	8.2 +3	--	10^{9}	--	8.2 +3
^{244}Cm	1.9 +3	10^{9}	7.7 +3	5.2 +2	--	1.7 +3	--	1.7 +3

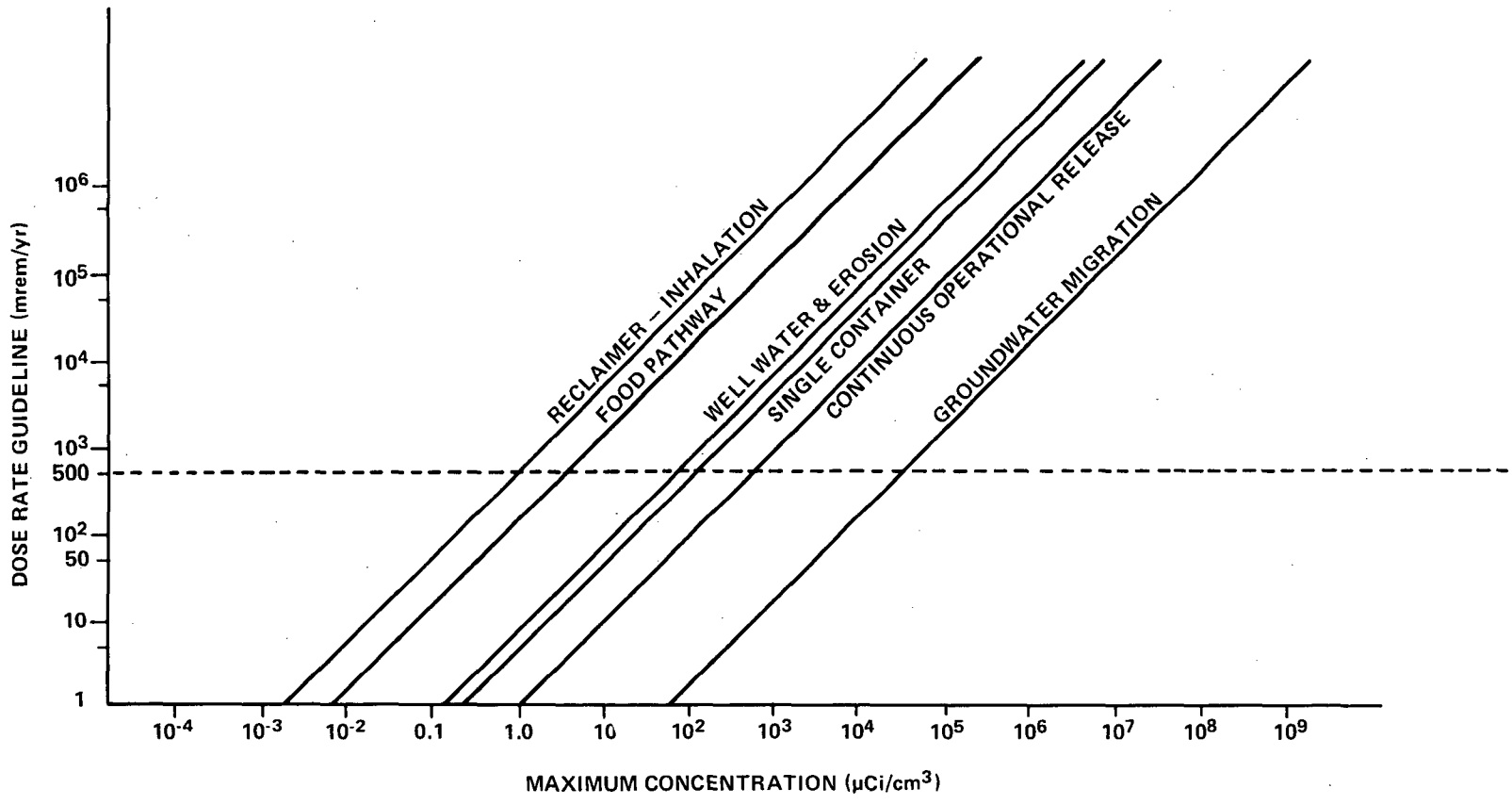


FIGURE 4.6 MAXIMUM CONCENTRATIONS FOR ^{239}Pu FOR EVENT AND DOSE GUIDELINE VARIATIONS

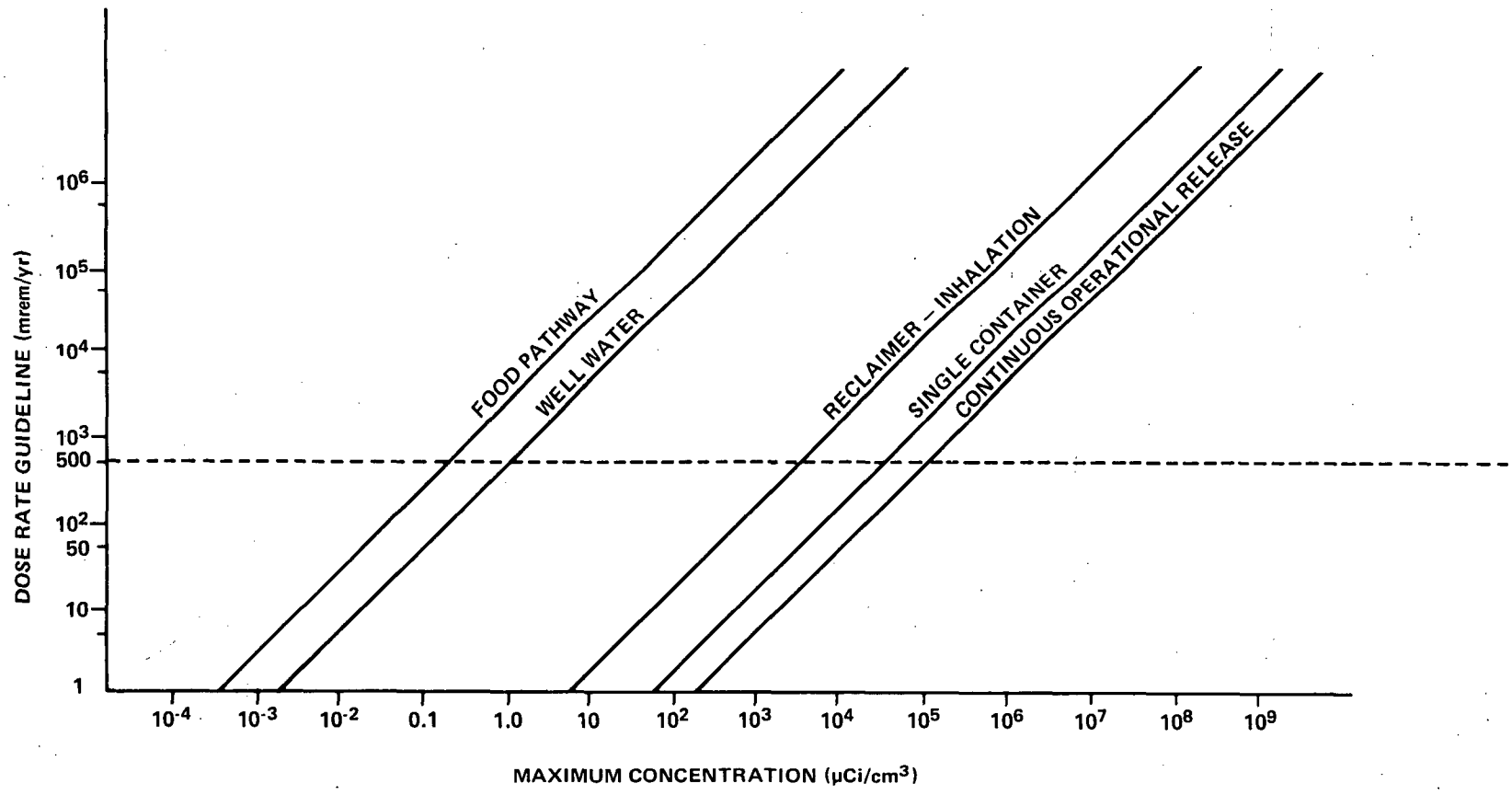


FIGURE 4.7 MAXIMUM CONCENTRATIONS FOR ^{90}Sr FOR EVENT AND DOSE GUIDELINE VARIATIONS

4.11.2 Maximum Allowable Concentration for Mixtures

The concentrations presented in Table 4.14 are based upon each isotope resulting in a dose rate equal to the dose guidelines. For mixtures of nuclides in the waste, the sum of the actual concentrations of each nuclide, C_i, to the MAC_i for each nuclide must be less than or equal to unity as shown in eq (4.21):

$$\sum_{i=1}^{n} \frac{C_i}{MAC_i} \leq 1 \qquad 4.21$$

where

C_i = concentration of isotope "i" in the mixture (Ci/cm^3)

MAC_i = MAC for isotope "i"

As an example, consider the mixture of nuclides in a specific amount of waste as shown in Table 4.15. Because the sum for all nuclides of ratios of the concentrations to MAC's is less than one, the waste can be considered to be low-level.

4.11.3 Variation of Dose Guidelines

The dose rate guidelines given in Chapter 2, have a range of two orders of magnitude associated with them. In all of the equations used in the analysis the maximum concentrations vary linearly with the dose rate guideline. The MAC's, therefore, will change in direct relation to the change in the dose rate guideline. This is illustrated in Figures 4.5 and 4.6 for ^{239}Pu and ^{90}Sr. The MAC's will increase or decrease by an order of magnitude with a corresponding order of magnitude increase or decrease in the guideline.

4.11.4 Maximum Allowable Concentrations for Intermediate Ground Burial

It is of considerable interest to investigate the changes that occur in the MAC's for low-level waste disposal when the depth of burial is increased from one meter to at least

TABLE 4.15

EXAMPLE CALCULATION FOR MIXTURE OF NUCLIDES

Nuclide	Concentration C_i ($\mu Ci/cm^3$)	MAC_i ($\mu Ci/cm^3$)	C_i/MAC_i
^{60}Co	230.00	55,000.0	0.0042
^{99}Tc	0.11	0.96	0.1150
^{137}Cs	0.88	8.3	0.1060
^{239}Pu	0.42	1.0	0.4200
Sum	231.41	---	0.6452

ten meters. Several events in the set then are no longer applicable. For example, the unwary reclaimer is not likely to dig deep enough to expose the wastes and direct gamma exposures are negligible due to the extra shielding. In addition, the food scenario is no longer applicable. The MAC's resulting from intermediate depth burial are given in Table 4.15. In most cases the MAC's for the transurunic nuclides are increased by two orders of magnitude, and the MAC for ^{137}Cs is increased by about six orders of magnitude. However, the MAC's for most of the other fission products are unchanged.

Intermediate depth burial also offers several operational and reclamational advantages at what may be a very little additional cost.

4.12 POPULATION DOSES AND COST-BENEFIT

Population doses have been calculated for both groundwater migration and airborne transport. Based on projections of waste concentrations and volumes and on costs for disposal in both HLW and LLW facilities, cost-benefit ratios have been formulated.

4.12.1 Population Dose Rates from Air Transport

Population doses for the single container accident event were estimated based on the reference population distribution shown in Table 4.16, and the same atmospheric conditions used for the calculation in Chapter 4.9. The results of these calculations for ^{239}Pu are also given in Table 4.18. The population dose rate is 1.2×10^{-4} manrem/yr. This yields a 50 year dose commitment of less than 6×10^{-3} manrem.

4.12.2 Population Dose Rates from Groundwater Migration

Population doses from groundwater migration were determined for the reference inventory by correcting the maximum individual dose rate for the "average individual" usage factor and multiplying by the downstream population at risk and by a factor of 50 to account for the dose commitment. This value is corrected for the site inventory by dividing by 800 GWeyr (1000 RRY of waste). The resulting value, for 1 $\mu Ci/cm^3$ of ^{239}Pu, is 10^{-4} manrem/GWeyr. The downstream population at risk is assumed to be 8×10^5 persons.

TABLE 4.16

POPULATION DOSE FROM SINGLE CONTAINER EVENT

Distance From Source	Population Along Centerline[a]	Dose (manrem/yr)
800	8	8 -5
1,000	8	2 -5
3,200	50	2 -6
8,000	50	1 -6
16,000	600	2 -6
24,000	400	2 -7
32,000	1,000	2 -7
40,000	1,000	7 -8
48,000	1,000	3 -7
64,000	3,200	2 -8
80,000	6,400	1 -8
TOTALS	13,716	1.2 -4

This gives 0.006 manrem 50-year dose commitment for the reference ^{239}Pu concentration per accident.

[a] Assumes 1/4 of the reference population is along centerline of plume

The isotope ^{239}Pu is a small contributor to the total population doses from groundwater migration. The reference inventory mixture of isotopes yields a normalized population dose of 5 manrem/GWeyr.

4.12.3 Waste Volume Projections

Projections of waste generation,[5] as shown in Figure 4.8 and Table 4.17, allow the determination of relative changes in volume, radioactivity concentrations and costs depending on the acceptance of different categories for disposal. The average concentration of the low-level waste in these projections[5] is about 10 $\mu Ci/cm^3$.

Therefore, changing the location of the HLW/LLW interface affects only a small fraction of the total projected wastes. The bulk of the waste volume will be below any probable threshold.

4.12.4 Cost-Benefit Analysis

The differential (marginal) doses calculated for changes in the LLW/HLW concentration interface can be combined with differential costs to provide a marginal cost-expanded to dose-avoided comparison. Guideline 3 and 4 apply to this analysis.

First, guideline 4 is less restrictive than guideline 1 or 2 for the RCF situation. Guideline 3, $1000/manrem, indicates a possible type of approach, but the actual quantitative value may not be applicable.

Cost figures used in this analysis are $3500/$m^3$ burial in a high-level waste repository and $100/$m^3$ for burial in a low-level waste disposal facility. This results in a cost savings of $3400/$m^3$ for each additional m^3 of waste that can be placed in the RCF, rather than at the waste repository.

Given the projected waste volumes and concentrations in Table 4.17 and the marginal costs of $3400/$m^3$ for disposal in the repository, the cost-benefit ratio can be determined. By increasing the MAC, additional volumes of waste will be acceptable for RCF disposal. For instance, the first projected category of waste higher than routine low-level waste would contain a maximum of 1000 Ci/m^3. The production of this waste is estimated to be 75 m^3/GWeyr. For a marginal cost of $3400/$m^3$, $255,000/GWeyr can be saved by containing this waste in the RCF.

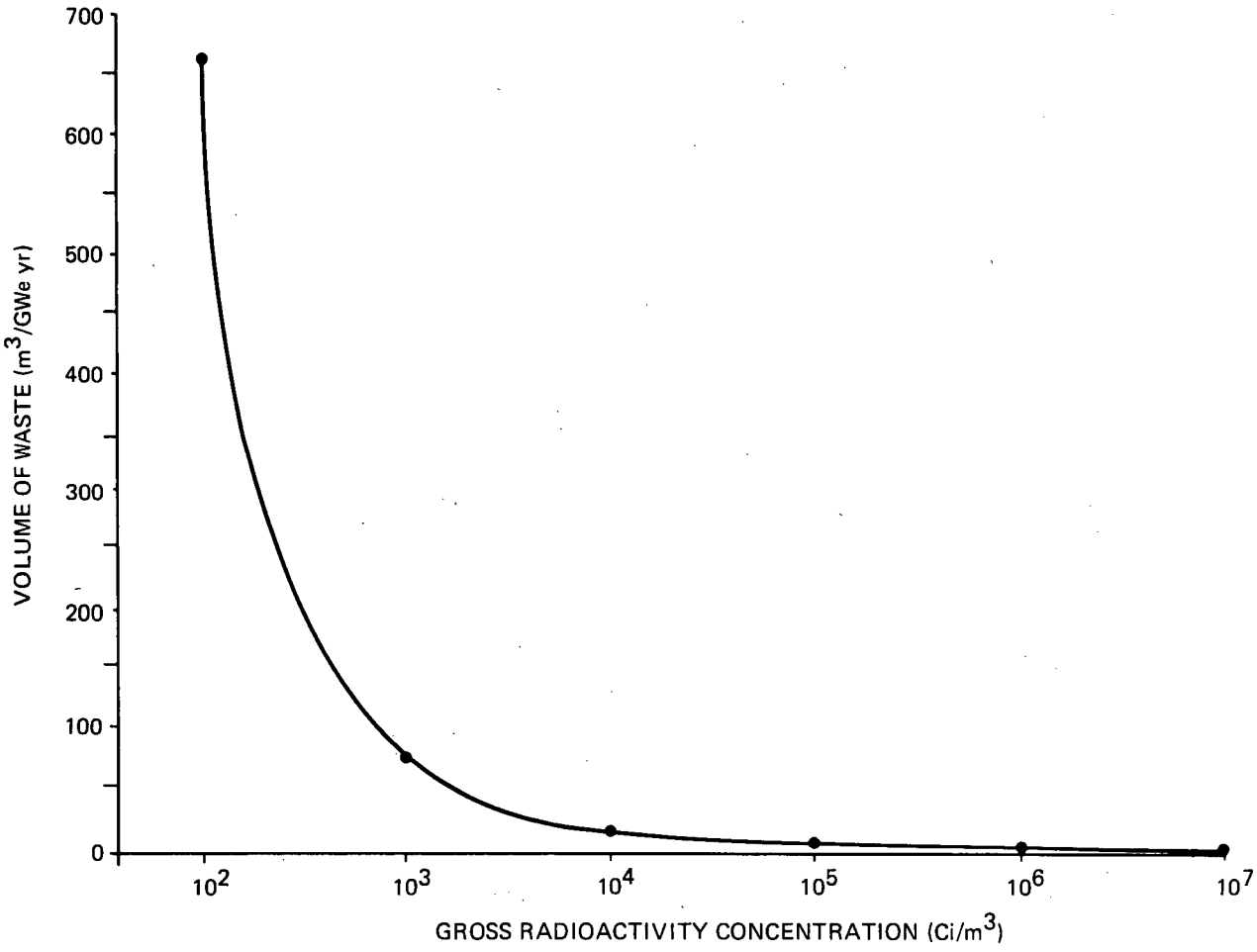


FIGURE 4.8 WASTE VOLUMES AND CONCENTRATIONS

TABLE 4.17

RELATIVE WASTE CONCENTRATIONS AND VOLUMES PROJECTED PER GW(e) YR[a]

Waste Categories As Generated	Gross Radioactivity Concentration (Ci/m^3) Upper Limit	Volume Generated Per GW(e) Yr (m^3)
Routine Low-Level	100	660
Intermediate Level 1	1,000	75
Intermediate Level 2	10,000	20
High-Level	10,000,000	3

(a) Information from ERDA -76-43

This would give a population dose of about 1×10^{-3} manrem/GWeyr for ^{239}Pu in the groundwater pathway, resulting in a cost-benefit ratio of $\$2 \times 10^{8}$/man-rem. The costs clearly exceed the guideline by several orders of magnitude. Figure 4.9 contains graphs of the costs, population doses and resulting cost-benefit ratios for the above ^{239}Pu example.

If all nuclides are considered, the population dose increases to 10 manrem/GWeyr, and the cost-benefit ratio for this level of wastes is $\$2.6 \times 10^{4}$/manrem.

4.13 PARAMETERS USED IN EVALUATING RELEASES FROM NONRADIOACTIVE-LOW LEVEL WASTE INTERFACE

The analysis of the interface between LLW and waste that need not be considered radioactive (from a regulatory waste disposal point of view), and hence would not require special controls or handling, is similar to the approach used for the low-level/high-level interface. This phase of the study involves the analysis of the set of events for radioactive release not from a controlled radioactive waste facility, but from a typical municipal waste disposal site (i.e. sanitary landfill operation).

Burial of very low levels of solid radioactive waste (i.e. less than 1,000 times the activity specified in 10CFR20 Appendix C) is presently permitted under rather restrictive conditions. The analysis in this study is based upon generalization of these restrictions but does not imply complete abandonment of all controls. This waste is still considered as municipal waste and should be treated accordingly with disposal made in an approved sanitary landfill facility. It is generally recommended that the wastes not be used as fill dirt nor otherwise discarded at random. However, burial in a municipal sanitary landfill facility under local or regional government control is recommended as a sound waste management practice.

Because disposal in a typical municipal sanitary landfill operation neither entails any particular long-range controls, nor implies any stringent burial or site standards, the analysis performed was based on reasonably conservative estimates of conditions that would likely exist for a typical sanitary landfill operation.

Both water and air pathways to man are examined. Finally, nonradioactive low-level waste interface values for this suggested waste classification system, based on selected dose guidelines, are presented and evaluated.

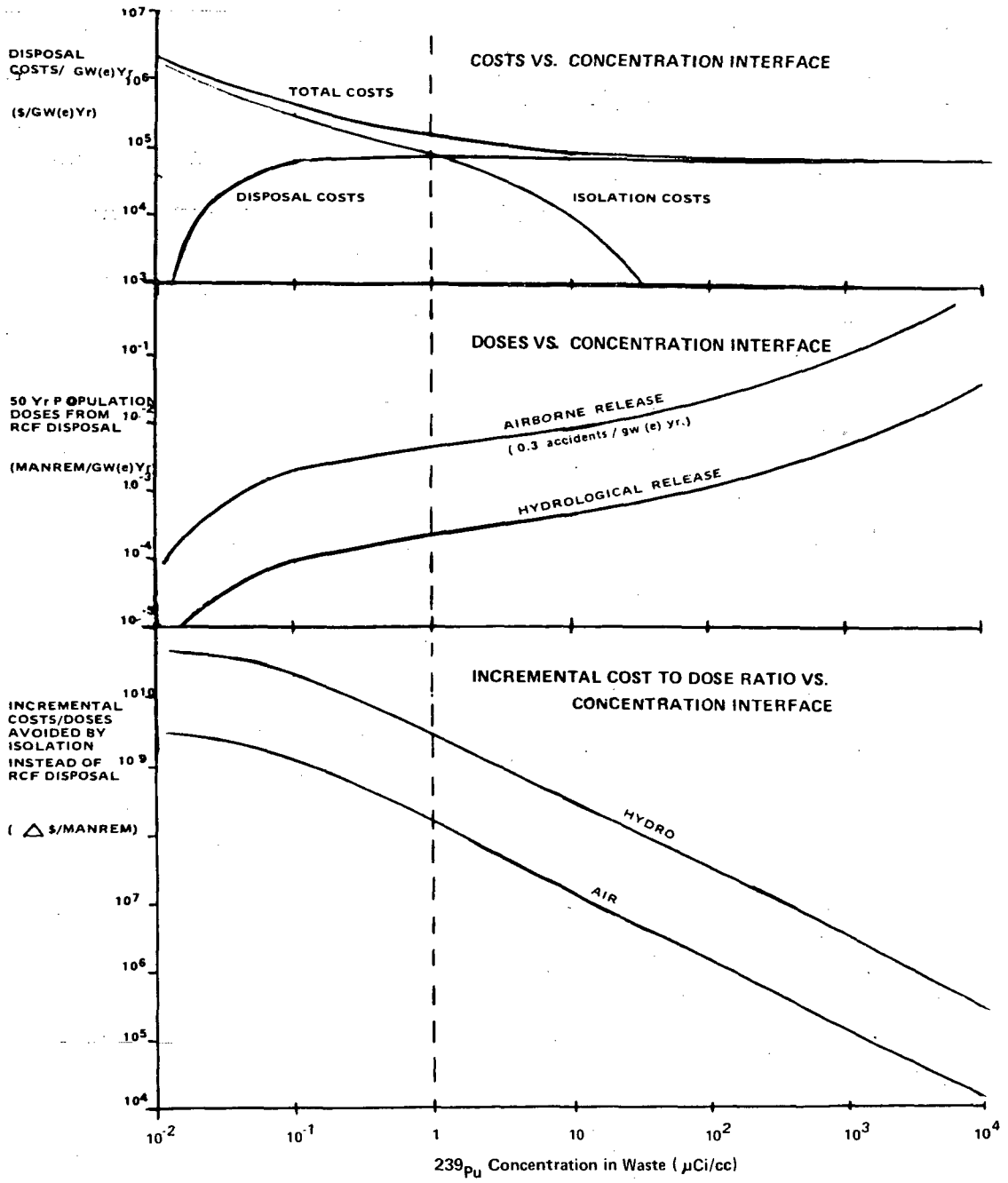


FIGURE 4.9 COST BENEFIT ANALYSIS RESULTS

The discussion of the parameters used to evaluate the threshold is divided into two parts. First, a general discussion of landfill operations is given, and this is then followed by a description of the reference sanitary landfill facility.

4.13.1 General Considerations

Landfill operations are typically located reasonably close to large population centers and usually on land that has very little value for any other use. Sanitary landfill operations are frequently used as a means of converting blighted land, quarries and swamps into usable land. Frequently the water table is very close to the waste and the soil may be a very porous gravel or sand.

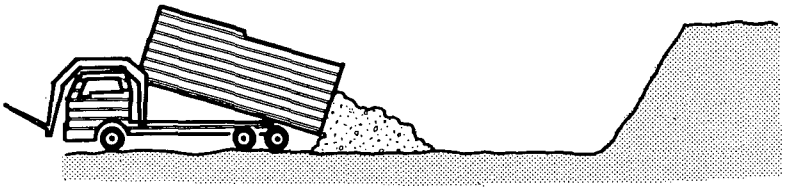
In most areas of the United States the legal authority for administering the management of soild waste rests within the local municipal governments. Municipalities are generally required to adopt codes based upon state and EPA standards to provide for enforcing acceptable practice in the storage, collection, processing and disosal of solid waste. However, many communities are notoriously lax in enforcing those standards.

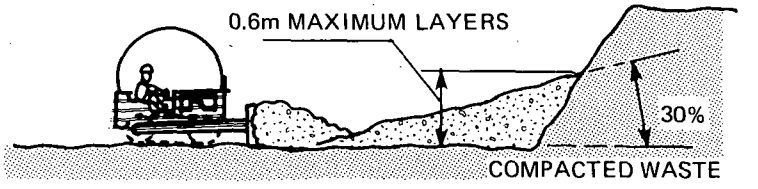
The total volume of waste generated by a community varies considerably with the affluence and location of the community. Across the nation, however, the average U.S. resident generates 6.3 m^3 of solid waste per year.[20]

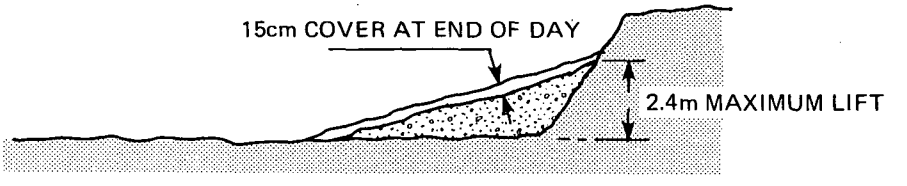
The general operation of a sanitary landfill facility is performed in the following manner. The waste is delivered to the site and deposited in a preplanned, prepared area. The waste is spread and compacted in thin layers by heavy equipment. At the end of each day the collection of waste is covered by a 15 cm layer of compacted earth. This creates a waste cell which is closed to insects and rodents, prevents wind spreading, and is odor-free and clean in apprearance. The typical sequence of a sanitary landfill operation is illustrated in Figure 4.10.

Successive cells are prepared in the same manner until the site is filled. The entire site is finally covered with 0.6 m of compacted earth, graded for proper drainage and planted with native ground cover.

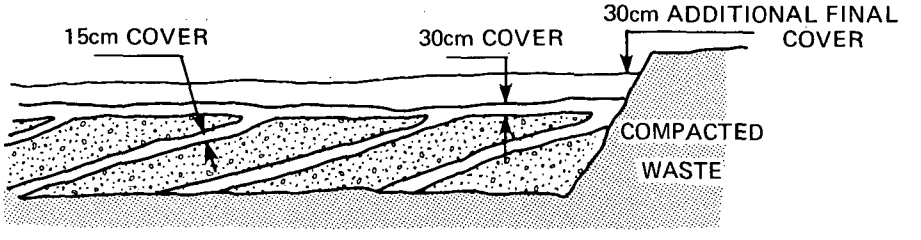
SOLID WASTE SHALL BE SPREAD AND COMPACTED IN SHALLOW LAYERS NOT EXCEEDING A DEPTH OF 0.6m OF COMPACTED SOLID WASTE. THE FACE OF THE OPERATING SANITARY LAND FILL SHALL BE KEPT AT OR NEAR 30% SLOPE SO AS TO INSURE MAXIMUM COMPACTION BY EQUIPMENT.







TYPICAL WASTE COMPACTION AND COVER CYCLE 1st LIFT



SECOND AND CONSECUTIVE CELLS

FIGURE 4.10 SANITARY LAND FILL OPERATION

4.13.2 Description of Reference Sanitary Landfill Facility

For the purposes of this analysis, the Reference Sanitary Landfill Facility (RSLF) has been defined as having the parameters shown in Table 4.18. The rationale for setting these parameters is as follows:

1. The RSLF will serve a community with a population of 250,000 for a five-year period.

2. Waste capacity at 6.3 m^3/individual/yr is 8 x 10^6 m^3.

3. For worst case, the water table is very close to the waste, i.e. within 1 m.

4. A moderately heavy annual rainfall occurs on the RSLF typical of the eastern U.S. (About 1.1 m/yr.)

5. Area has a 1 m/sec average windspeed in the direction of maximum population.

4.14 ENVIRONMENTAL PATHWAYS

Waste materials containing very low quantities of radioactive contamination, if disposed of in a municipal landfill operation, must be considered as potentially available to spread contamination immediately or within a very few years to the general public. Radioactive waste that is intermixed with municipal waste would undoubtedly become mixed with many toxic materials included in municipal waste that would be more hazardous than the radioactive materials (such as: septic materials, corrosive agents, insecticides, pesticides, mercury, and other heavy metals and chemical poisons). Therefore, in considering pathways to the environment, some scenarios that have been proposed previously,[19] such as ingestion of the wastes, have not been included. It is believed that if such an event were to occur, the person involved would have many other problems much more serious than the danger from the radioactive waste.

4.14.1 Inhalation of Contaminated Dust

It was postulated that a worker at the landfill facility may attempt to salvage some item from the waste, thereby stirring up a cloud of dust, or he may be standing close to the truck when the load is dumped. It is also assumed that

TABLE 4.18

REFERENCE SANITARY LANDFILL OPERATIONS PARAMETERS

Parameter	Value
Site Plan Area	2.0 + 6 m^3
Site Capacity for Waste	8.0 + 6 m^3
Distance to Site Boundary	10 m
Distance-Pit Bottom to Aquifer	1.0 m
Water Velocity-Pit Aquifer	1.0 m/yr
Annual Precipitation	1.1 m/yr
Distance-Center of Site to Surface Water	1,000 m
Water Velocity-Aquifer	100 m/yr
Dispersion Coefficient	10 m^2/yr
Minimum Earth Cover Over Waste	0.6 m
Fraction of Pit Volume Occupied by Radioactive Waste	0.05
River Flow Rate	5.0 + 2 m^3/s
Mean Wind Velocity	1.0 m/s
Radioactive Material Suspension Rate (Normalized to 1 pCi/g Soil Concentration)	14 pCi/s

this is a recurring event, and not just a random occurrence. If 5% of the waste at the landfill site is radioactive waste, then this event could occur 5% of the time that the worker is at the site (conservatively assumed to be the same individual at each release occurrence). The maximum concentration allowable in waste from such an event is calculated from eq (4.2), with the exponential term omitted because the event occurs during waste handling.

The values of the parameters used in eq (4.2) for the RSLF are presented in Table 4.19. The resultant maxium concentrations for wastes at the RSLF are given in Table 4.20.

4.14.2 Water Well Near Landfill

Another possible exposure event is the drilling of a water well into the aquifer near the boundary of the landfill. For a site the size of the RSLF, contaminants leached from the buried wastes must travel an average of 800 m to arrive at the site boundary. In that distance, substantial dilution in the aquifer will take place. It is assumed that the well is located downgradient from the RSLF on the centerline of the released plume in the aquifer. If the aquifer is only 50 m thick, approximately 1.6×10^6 m^3/yr of water flows under the RSLF.

The contaminants are leached by the 2.2×10^6 m^3/yr of rainwater entering the groundwater and are diluted into the volume flowing under the RSLF. Lateral and vertical dispersion cause additional dilution as the contaminants are transported downgradient. Assuming no further dilution by dispersion has occurred at the site boundary where a well may be drilled gives about 3.6×10^6 m^3/yr as the dilution volume. The results of the calculations using eq (4.3) with this value for m_t and a dose guideline of 500 mrem/yr are presented in Table 4.21.

The values of the initial inventory which result in 500 mrem/yr maximum doses are also given in Table 4.22. It is likely that only a few people would drink from this well. Chemical and other contaminants leached from the municiple wastes would probably be the controlling factors in establishing the suitability of the water for human consumption. These calculations, therefore, are not expected to be the actual limiting case, because use of the well water for drinking is very unlikely. Many of the chemical pollutants that make the water unpotable do not undergo radioactive decay, and will remain at high concentrations longer than the radioactive contaminants.

TABLE 4.19

PARAMETERS USED IN WORKER INHALATION EVENT

D = 500 mrem/yr

ρ = 1.6

K = 5×10^{-4} g/m^3

U_a = 0.91 m^3/hr

T_x = 1920 hr/yr

f = fracture of waste that is radioactive 0.05

$(DF)_m$ = appropriate dose conversion factor from Reg. Guide 1.109

TABLE 4.20

MAXIMUM CONCENTRATIONS IN WASTE AT RSLF TO

ASSURE GUIDELINES ARE NOT EXCEEDED FOR DIRECT INHALATION

Nuclide	Critical Organ	C_m ($\mu Ci/cm^3$)
^{3}H	Total Body	1.1 +5
^{14}C	Bone	7.9 +3
^{54}Mn	Lung	1.0 +2
^{55}Fe	Lung	2.0 +3
^{57}Co	Lung	3.9 +2
^{60}Co	Lung	2.4 +1
^{63}Ni	Bone	3.3 +2
^{90}Sr	Bone	1.5
^{99}Tc	Total Body	13
^{125}Sb	Lung	65
^{129}I	Thyroid/ Total Body	3.2 2.6 +3
^{135}Cs	Bone	1.2 +3
^{137}Cs	Liver	2.3 +2
^{232}Th	Bone	9.0 -3
^{235}U	Lung	3.7 -1
^{238}U	Lung	3.9 -1
^{237}Np	Bone	1.1 -2
^{238}Pu	Bone	6.7 -3
^{239}Pu	Bone	5.9 -3

TABLE 4.20 (Cont)

Nuclide	Critical Organ	C_m ($\mu Ci/cm^3$)
^{240}Pu	Bone	5.9 -3
^{241}Pu	Bone	3.0 -1
^{242}Pu	Bone	6.2 -3
^{241}Am	Bone	1.8 -2
^{243}Am	Bone	1.8 -2
^{242}Cm	Lung	4.8 -1
^{244}Cm	Bone	3.1 -2

TABLE 4.21

MAXIMUM INVENTORY IN THE RSLF
BASED ON WATER WELL NEAR SITE BOUNDARY

Nuclide	Concentration in Well (pCi/ℓ)	Maximum Inventory (Ci)
^{3}H	6.5 +6	9.5 +5
^{14}C	2.4 +5	8.9 +6
^{90}Sr	9.0 +1	1.1 +5
^{99}Tc	1.1 +5	4.0 +6
^{129}I	9.5 +1/ 7.4 +4	6.4/ 5.0 +3
^{135}Cs	3.5 +4	1.6 +6
^{235}U	8.5 +2	7.0 +5
^{238}U	8.9 +2	7.4 +5
^{239}Pu	9.0 +2	5.7 +6
^{240}Pu	9.0 +2	4.7 +7
^{242}Pu	9.5 +2	8.3 +5
^{243}Am	8.4 +2	3.3 +7

TABLE 4.22

MAXIMUM CONCENTRATIONS IN WASTES AT THE RSLF
BASED ON WATER WELL NEAR SITE BOUNDARY

Nuclide	Concentration in Well (pCi/ℓ)	Maximum Concentration in Wastes ($\mu Ci/cm^3$)
^{3}H	6.5 +6	2.4
^{14}C	2.4 +5	22
^{90}Sr	9.0 +1	0.27
^{99}Tc	1.1 +5	9.9
^{129}I	9.5 +1/ 7.4 +4	1.6 -5/ 1.3 -2
^{135}Cs	3.5 +4	4.0
^{235}U	8.5 +2	1.8
^{238}U	8.9 +2	1.8
^{239}Pu	9.0 +2	14
^{240}Pu	9.0 +2	1.2 +2
^{242}Pu	9.5 +2	2.1
^{243}Am	8.4 +2	83

4.14.3 Direct Gamma Exposures

For gamma emitting radionuclides, direct exposure of workers handling the wastes at the RSLF is of concern. Using eq (4.6), but omitting the exponential term because the exposure takes place at the time of emplacement of the waste, gives the maximum concentrations listed in Table 4.23. It is assumed that workers would be directly exposed to the wastes approximately half the time they are on the job (960 hrs/yr).

4.14.4 Atmospheric Releases of Contamination

It was assumed that 5% of all waste delivered to the site is radioactive. Given the characteristics and the active lifetime of the site, the average rate of delivery of radioactive material to the site was found to be 2.5×10^{-3} m^3/sec. It was further assumed that 10^{-5} of the volume of the radioactive material became suspended in air and was transported by the wind toward the population center. With an assumed waste density of 1.6 gm/cm^3, the rate of suspension of radioactive material (source) is Q = 0.066 pCi/sec per pCi/m^3 of activity in the waste.

It is assumed that the newly delivered refuse is covered with soil each day, giving a geometric source that is a right circular cone whose volume is approximately 35 m^3 (one day's worth of radioactive material). The radius of this cone is about 4.0 m and the height about 3.5 m. The area is about 50 m^2.

It is assumed that the prevailing winds blow towards the population center and the frequency of this wind is 55%. The winds are assumed to blow isotropically in the other 15 compass directions for the remaining 45% of the time. The average wind speed is 1 m/sec and stability class E is assumed.

Equation (4.9) relates the maximum concentrations in the waste to the waste source and site parameters. The values of the parameters used in the RSLF calculation are as follows:

$$D = 500 \text{ mrem/yr}$$

$$Y = 5 \text{ yrs}$$

$$P = 20$$

$$f_{rc} = 10^{-5}$$

TABLE 4.23

MAXIMUM CONCENTRATIONS OF RADIONUCLIDES

FROM DIRECT EXPOSURE EVENTS

Nuclide	Max Concentration ($\mu Ci/cm^3$)
^{60}Co	1.0 -2
^{90}Sr	13
^{129}I	3.4
^{137}Cs	8.4 -2
^{237}Np	0.9
^{235}U	0.4
^{238}U	6.0
^{238}Pu	1.4 +2
^{240}Pu	9.8
^{241}Am	2.0
^{243}Am	0.5
^{244}Cm	16

$$(\chi/Q) = 1.6 \times 10^{-15}\ yr/m^3$$

$$V = 8 \times 10^{12}\ cm^3$$

$$U_a = 8 \times 10^6 \ell/yr$$

$$(DF)_m = \text{dose conversion factor (mrem/}\mu\text{Ci)}$$

Table 4.24 lists the maximum concentrations in the waste that will assure that the dose guidelines are not exceeded for an individual at the nearest site boundary (800 m from the source).

The sector population distribution used in calculating population doses and average individual doses in each sector in the direction of the prevailing wind are presented in Table 4.25.

Parameters which significantly influence the airborne particulate concentrations at a given radial distance are tabulated in Table 4.26. The lateral and vertical dispersion coefficients are represented by σ_y and σ_z, respectively, and the depletion fraction is represented by Q_{eff}/A_o. The apparent or effective source is given by Q_{eff}. Population doses from wastes containing the maximum concentration of ^{239}Pu are also tabulated.

4.14.5 Groundwater Migration

Radionuclides in the materials disposed in the landfill can be released to the environment through leaching and transported through groundwater systems to a point where the groundwater emerges as springs or through slow seepage into a waterway. Because site specific values for the parameters influencing waste migration are not specified, reasonably conservative values corresponding to relatively poor sites have been used in this evaluation. Humans using the contaminated water will receive a radiation dose. The dose clearly depends on the type of nuclide and the time and manner in which the water is consumed. Direct ingestion of the water is the manner which yields the greatest dose. To estimate the dose to humans, based on the amount of a particular nuclide handled in the RSLF, calculation of the release rates from the landfill is performed. At the release point into surface water, dilution will occur, and humans may use the water containing small amounts of these nuclides.

The sorption coefficient and leach constant for each of the nuclides considered to be representative are listed in

TABLE 4.24

ISOTOPE CONCENTRATIONS IN WASTES THAT RESULT IN 500 mrem/yr DOSE TO INDIVIDUAL AT SITE BOUNDARY

Isotope	Waste Concentration ($\mu Ci/cm^3$)
^{3}H	3.0 +5
^{14}C	2.1 +3
^{55}Fe	5.3 +3
^{57}Co	1.0 +3
^{60}Co	6.4 +1
^{90}Sr	3.8
^{99}Tc	3.5 +2
^{125}Sb	17
^{129}I	8.6/ 6.9 +3
^{135}Cs	3.3 +3
^{137}Cs	6.1 +2
^{235}U	9.7 -1
^{238}U	1.0
^{238}Pu	1.8 -2
^{239}Pu	1.8 -2
^{240}Pu	1.6 -2
^{241}Pu	7.9 -1
^{242}Pu	1.6 -2
^{241}Am	4.8 -2
^{243}Am	4.8 -2
^{242}Cm	1.3
^{244}Cm	8.1 -2
^{237}Np	2.8 -2

TABLE 4.25

POPULATION DENSITY DISTRIBUTION AS A FUNCTION OF PLUME DISTANCE FROM SOURCE OF RADIATION AT RSLF

Radial Distance (m)	Population (Sector)
1205	2000
2415	6000
5635	6000
10025	15000
14050	15000
18565	44000
22590	44000
28175	118000

TABLE 4.26

POPULATION CENTERLINE DOSE DATA

Distance from Source (meter)	Population Along Centerline[a]	Q_{eff}/Q_o (pCi/sec) 1cm/sec deposition velocity	Population for MAC of ^{239}Pu (manrem/yr)
800	1	1.00	1.4 -1
1205	500	0.80	3.2 +1
2415	1500	0.40	1.7 +1
5635	1500	0.17	2.6
10025	3750	0.12	1.7
14050	3750	0.08	8.6 -1
18565	11000	0.07	1.4
22590	11000	0.05	1.1
28175	29500	0.04	1.7

[a]Assumes 1/4 of the reference population is along centerline of plume.

Table 4.27. A discussion of the mathematical model is given in Appendix B.2. The calculated releases for each nuclide are given in Table 4.28. The major nuclides which are released from the aquifer are ^{3}H, ^{90}Sr, ^{99}Tc, ^{129}I, and ^{239}Pu. The plutonium release rate is 14 orders of magnitude less than at the source, and the time of maximum release rate (1,000,000 years) minimizes the impact of plutonium releases on the human population. Strontium-90 releases from the aquifer are negligible. Doses are calculated using the method described in Section 4.8 of this report. Tritium, iodine and technetium must be considered in dose calculations because their fractional releases are relatively large.

Radionuclides whose half-lives and sorption coefficient are such that the time required for transport to the surface water is greater than 30 half-lives can generally be ignored.

The calculations are based on a maximally exposed individual consuming 100% of his drinking water requirements from the river without the benefit of filtration, sedimentation or other treatment which would reduce the concentrations of contamination, and provide an upper estimate of the doses received from groundwater transport to surface waters. Direct ingestion by drinking has been shown generally to be the most significant of all possible ingestion pathways for human exposure from contaminated water, and has been used in this study as an indication of the magnitudes of the expected doses from waste disposal.

4.14.6 Ingestion of Food Produced on RSLF Site

The RSLF could eventually be put to beneficial use for housing or farming. If the wastes were uncovered and mixed with the surface soil, contamination of plants grown there would occur. Using the same procedure and values for parameters given in Chapter 4.9, except the time period of institutional control is zero years, gives the interface concentration values listed in Table 4.29. This pathway is seen to be particularly important for ^{3}H, ^{14}C, ^{57}Co, ^{90}So and ^{99}Tc.

4.15 NONRADIOACTIVE/LOW-LEVEL WASTE INTERFACE

Based on the results of the pathways analyses presented in Chapter 4.14, nonradioactive/low-level waste interface concentrations from the most limiting concentrations for each event can be determined to assure that the dose guidelines

TABLE 4.27

NUCLIDE SPECIFIC PARAMETERS FOR GROUNDWATER MIGRATION

Nuclide	Half-life (yr)	Decay Constant (yr^{-1})	Leach Constant (yr^{-1})	Sorption Coefficient
^{3}H	12.3	5.6×10^{-2}	0.1	1
^{55}Fe	2.7	0.257	0.1	3300
^{60}Co	5.3	0.132	0.1	3300
^{90}Sr	29	2.4×10^{-2}	10^{-2}	100
^{99}Tc	2.1×10^{5}	3.2×10^{-6}	10^{-4}	1
^{129}I	1.6×10^{7}	4.4×10^{-8}	0.1	1
^{137}Cs	30.1	2.3×10^{-2}	10^{-3}	1000
^{239}Pu	2.4×10^{4}	2.8×10^{-5}	10^{-5}	10^{4}

TABLE 4.28

NUCLIDE RELEASES FROM RSLF THROUGH GROUNDWATER TRANSPORT TO RIVER

Nuclide	Time of Peak (yr)	Initial Inventory (Ci)	Maximum Nuclide Concentration in RSLF ($\mu Ci/cm^3$)	Population Dose (manrem/yr)
^{3}H	4	1.9 +9	4.8 +3	1.6 -4
^{14}C	2.5 +2	3.8 +10	9.5 +4	8.0 -3
^{99}Tc	11	2.6 +11	6.6 +5	1.2 -3
^{129}I	11	1.6 +4	4.0 -2	20
^{135}Cs	14	1.1 +9	2.8 +3	2.9 -2
^{137}Cs				
^{235}U	+5	2.1 +9	5.3 +3	1.5
^{238}U	+5	2.2 +9	5.5 +3	1.4
^{239}Pu	+5	2.9 +12	7.2 +4	0.1
^{240}Pu	+5	2.0 +12	5.0 +8	1.5 -5
^{242}Pu	+5	2.4 +9	6.0 +3	1.3
^{243}Am	+5	5.2 +11	1.3 +8	5.9 -5

TABLE 4.29

INTERFACE CONCENTRATIONS FROM FOOD PATHWAY

Nuclide	MAC ($\mu Ci/cm^3$)
^{3}H	1
^{14}C	0.024
^{55}Fe	240
^{57}Co	1.4
^{60}Co	128
^{90}Sr	4.6 -3
^{99}Tc	0.96
^{129}I	3.6 - 3/2.8
^{135}Cs	1.9
^{137}Cs	0.4
^{235}U	0.30
^{238}U	0.31
^{237}Np	0.28
^{238}Pu	3.2
^{239}Pu	3.2
^{240}Pu	3.2
^{241}Pu	152
^{242}Pu	3.4
^{241}Am	3.0
^{243}Am	3.0
^{242}Cm	3.76
^{244}Cm	5.0

are not exceeded. For the groundwater pathways, the total site inventory contributes to the concentrations of nuclides in the water consumed. The inventory limit can be reduced to a concentration limit by using the total volume of the radioactive waste (4×10^5 m^3) and the waste density (1.6 g/cm^3). As such, the interface concentrations in parentheses in Table 4.30 are specific to the RSLF, but are a conservative application of the inventory limitation and provide a consistent basis for the RWDCS.

For the airborne cases, the concentrations in the waste are limiting, and were calculated directly. Table 4.30 summarizes the pathways and interface concentrations for the RSLF. Direct inhalation of dust from the waste, direct gamma, food and well water are seen to be the most restrictive cases.

The interface concentrations for the nuclides derived from ingestion of water from the well at the site boundary may be unrealistic, however, because of the low probability of using water so contaminated with chemical pollutants from municipal wastes. If the wastes are contained or immobilized so that the effective leach rates assumed in this analysis are reduced, the well event also becomes less important. Tritium wastes are typically solidified, yielding leach constants lower than the 0.1 used. For iodine and technetium wastes, the physical and chemical forms may be such that leach rates lower than those used may be appropriate.

The limits on concentration were derived for each nuclide individually. In actual wastes, there is expected to be a known mixture of nuclides. To determine whether the mixture falls below the threshold limit, the sum of the ratios of the individual concentrations in the waste for each nuclide with the limit for that nuclide must be less than one, as explained in Chapter 4.11 of this report. This is the same methodology as is now used in applying the limits found in reference 13 to mixtures of liquids or airborne contaminants.

It is instructive to determine whether as low as reasonbly achievable (ALARA) guidelines are met by this approach for selecting the nonradioactive/low-level waste interface concentration limits. As an illustration of an ALARA calculation, consider the cost and dose implications of sending the RSLF radioactive waste to a reference containment facility (RCF) for low-level radioactive wastes. To minimize the cost per unit dose avoided, assume that the costs for the RCF are \$100/m^3, and for the RSLF \$0/m^3,

TABLE 4.30

MAXIMUM CONCENTRATIONS FOR EXPOSURE PATHWAYS
($\mu Ci/cm^3$)

Nuclide	Worker Inhalation	Water Well	Direct Gamma	Off-Site Inhalation	Groundwater Migration	Food Pathway	MAC
^{3}H	1.1 +5	2.4		3.0 +5	4.8 +3	1	1
^{14}C	7.9 +3	2.2 +1		2.1 +3	9.5 +4	2.4 -2	2.4 -2
^{55}Fe	2.0 +3	10^9		5.3 +3	10^9	2.4 +2	2.4 +2
^{57}Co	3.9 +2	10^9		1.0 +3	10^9	1.3 +2	1.3 +2
^{60}Co	2.4 +1	10^9	1.0 -2	6.4 +1	10^9	1.4	1.0 -2
^{90}Sr	1.5	2.7 -1	1.3 +1	3.8	10^9	4.6 -3	4.6 -3
^{99}Tc	1.3 +2	9.9		3.5 +2	6.6 +5	9.6 -1	9.6 -1
^{129}I T	3.2	1.6 -5		8.6	4.0 -2	3.6 -3	(1.6 -5)
WB	2.6 +3	1.3 -2	3.4	6.9 +3	3.1 +1	2.8	(1.3 -2)
^{135}Cs	1.2 +3	4.0		3.3 +3	2.8 +3	1.9	1.9
^{137}Cs	2.3 +2	10^9	8.4 -2	6.1 +2		4.0 -1	8.4 -2
^{235}U	3.7 -1	1.8	4.2 -1	9.7 -1	5.3 +3	3.0 -1	3.0 -1
^{238}U	3.9 -1	1.8	6.0	1.0	5.5 +3	3.1 -1	3.1 -1
^{237}Np	1.1 -2	10^9	9.0 -1	2.8 -2	10^9	2.8 -1	1.1 -2
^{238}Pu	6.7 -3	10^9	1.4 +2	1.8 -2	10^9	3.2	6.7 -3
^{239}Pu	5.9 -3	1.4 +1		1.6 -2	7.2 +4	3.2	5.9 -3
^{240}Pu	5.9 -3	1.2 +2	9.8	1.6 -2	5.0 +8	3.2	5.9 -3
^{241}Pu	3.0 -1	10^9		7.9 -1	10^9	1.5 +2	3.0 -1
^{242}Pu	6.2 -3	2.1		1.6 -2	6.0 +3	3.4	6.2 -3
^{241}Am	1.8 -2	10^9	2.0	4.8 -2	10^9	3.0	1.8 -2
^{243}Am	1.8 -2	8.3 +1	5.0 -1	4.8 -2	1.3 +8	3.0	1.8 -2
^{242}Cm	4.8 -1	10^9		1.3	10^9	3.8	4.8 -1
^{244}Cm	3.1 -2	10^9	1.6 +1	8.1 -2	10^9	5.0	3.1 -2

and that the doses/m^3 from the RCF are zero (they are expected to be several orders of magnitude less that those from the RSLF).

From the limiting case for ^{239}Pu, direct inhalation of dust by workers at the RSLF, assume that four workers receive 500 mrem/yr each for 5 years from waste operations. The total population dose from the RSLF will thus be 10 manrem from 400,000 m^3 of the incremental costs for sending the wastes to the RCF ($100/$m^3$), and the incremental dose avoided at the RSLF (2.5×10^{-5} manrem/m^3) gives $4 x 10^6/manrem. It is obvious that ALARA guidance is not the most restrictive or limiting consideration in selecting maximum concentrations for the wastes suitable for disposal at the RSLF.

This analysis has resulted in selection of conservative limits by which public health and safety will be protected. Selection of less conservative parameters for use in the calculations may be justifiable. Additional efforts in selecting and using the various parameters are recommended. The dust loading factor of 500 $\mu g/m^3$ may be an order of magnitude higher than would be typical for essentially full time conditions. Detailed considerations of these type of factors would tend to raise the nonradioactive/low-level wastes interface concentrations.

4.16 CAUTIONS AND CONCLUSIONS

This classification system is intended to provide radioactive waste generators and handlers guidance only on the final disposition of the wastes. The methodology has provided a basis for quantifying the interfaces between the waste classes. However, some care must be exercised in the application of the RWDCS methodology.

One area that was also investigated is the potential impact that the radioactive decay daughters have on the MAC's.

4.16.1 Impacts of Decay Daughters

The impacts of the radioactive waste are mitigated mainly by decay or dilution. However, as a radionuclide decays the decay daughters generated can also cause negative environmental impacts, the sum total of which can be larger than the maximum impact of the parent.

The main objective of this section is to determine the effect the decay daughters have on the MAC and on the total

inventory of the radionuclides. In accomplishing this the impact of the daughters is compared to the impact of the decay chain parent.

4.16.1.1 Decay Schemes

There are 15 radionuclides whose environmental impact may be affected by the ingrowth of daughters. The parents of two of these chains are the fission products ^{90}Sr and ^{135}Cs. The remaining 13, thorium, uranium and the transuranics, all belong to the four major radioactive decay series characterized by (4n+i), with i = 0, 1, 2 and 3. These decay series are shown in Figures 4.11-4.14. In the figures, a diagonal transition corresponds to the alpha decay and a horizontal transition corresponds to beta decay daughter nuclide concentrations. The ^{90}Sr and ^{135}Cs decay schemes are shown in Figure 4.15.

4.16.1.2 Daughter Effects from Reclaimer-Inhalation Events

The reclamation events do not involve a transport of the radionuclide before the specific event occurs. Therefore, the ingrowth of daughters is described by the Bateman Equations[21] and the activity of each daughter is limited by the activity of the parent.

The nuclear characteristics and concentrations at 100 years for all members of the decay chains are given in Appendix C. Each table in this appendix gives information for a different chain. The daughter information that is given pertains to that chain only, so that the total concentration of a particular daughter is the sum of the concentrations of that nuclide from all chains in which it appears.

The MPC's are used to determine the impact of the radionuclides relative to the parent, then the relative impact of nuclide i, RI_i, is defined by:

$$RI_i = \left(\frac{\text{Concentration (i)}}{\text{MPC (i)}}\right) \Big/ \left(\frac{\text{MAC (parent)}}{\text{MCP (parent)}}\right) \quad (4.22)$$

The relative impact varies significantly with time, so that the effect of the daughters could become more important and at times significantly greater than 100 years. The time dependence of the relative impact is shown in Figures 4.16

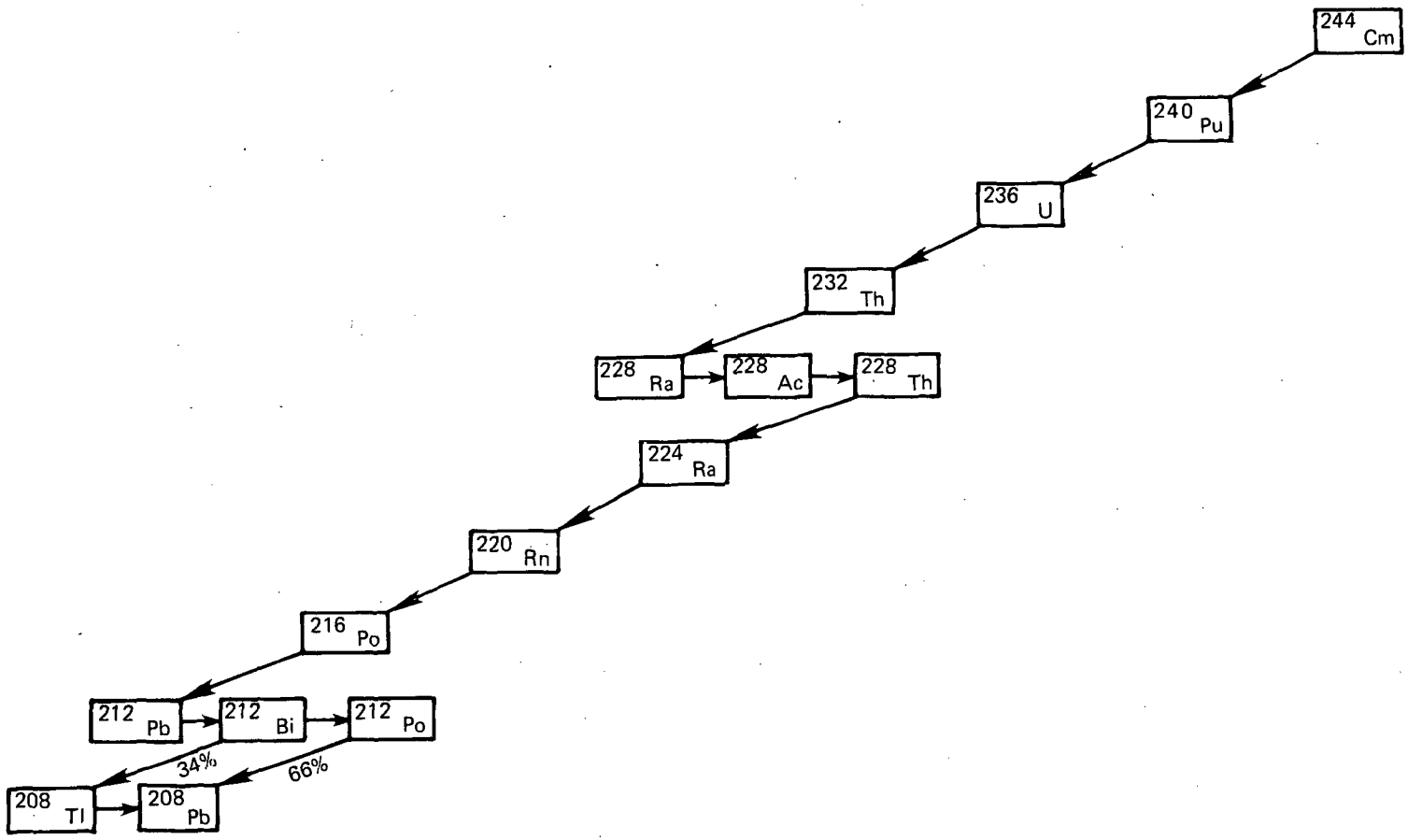


FIGURE 4.11 RADIOACTIVE DECAY CHAIN FOR THE 4n SERIES. ONLY THE MAJOR BRANCHING IS SHOWN. HORIZONTAL TRANSITIONS ARE BETA DECAYS AND PROCEED TO THE RIGHT. DIAGONAL TRANSITIONS ARE ALPHA DECAYS.

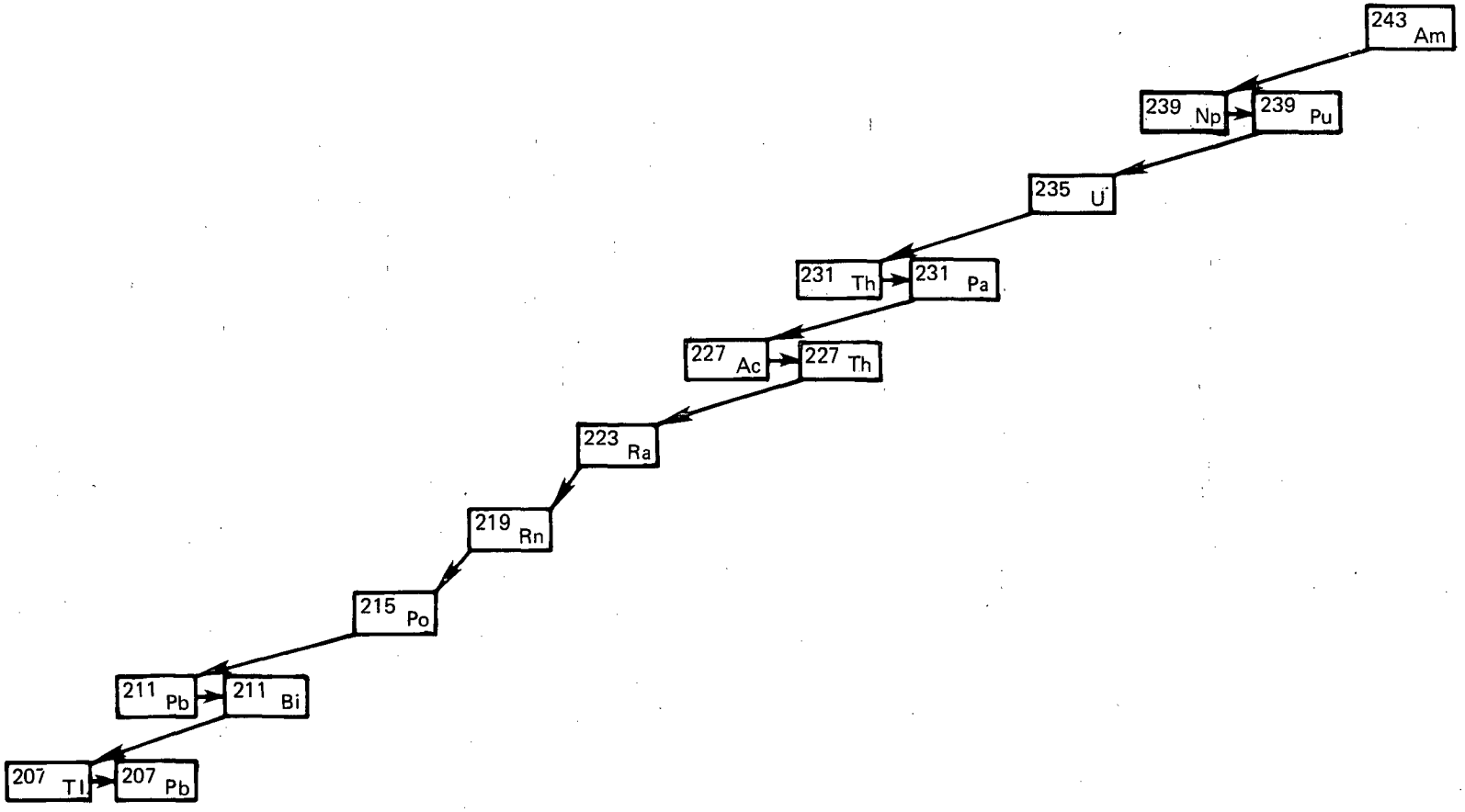


FIGURE 4.12 RADIOACTIVE DECAY CHAIN FOR THE (4n+3) SERIES
ONLY THE MAJOR BRANCHING IS SHOWN.

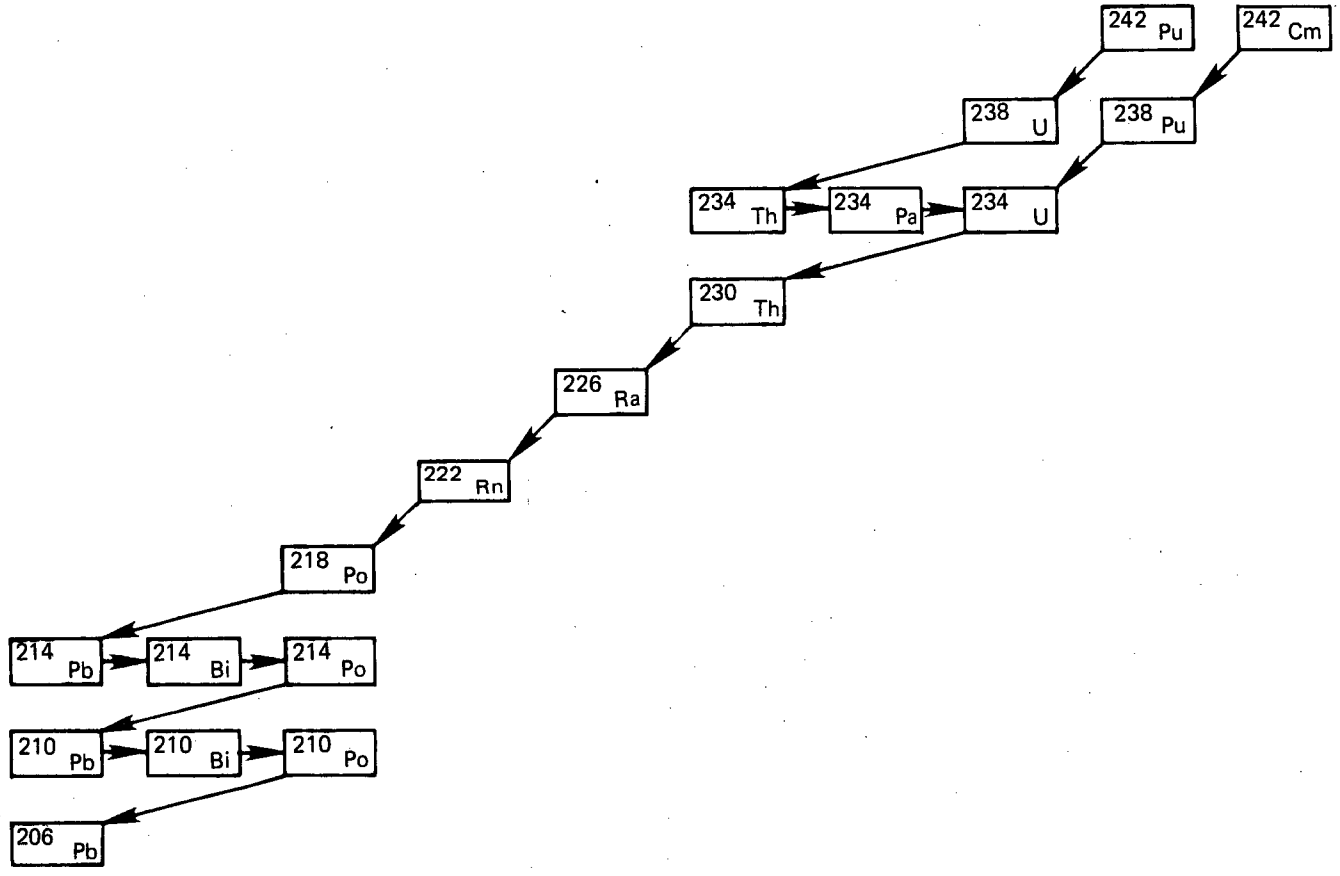


FIGURE 4.13 RADIOACTIVE DECAY CHAIN FOR THE (4n+2) SERIES. MINOR BRANCHING IS NOT SHOWN.

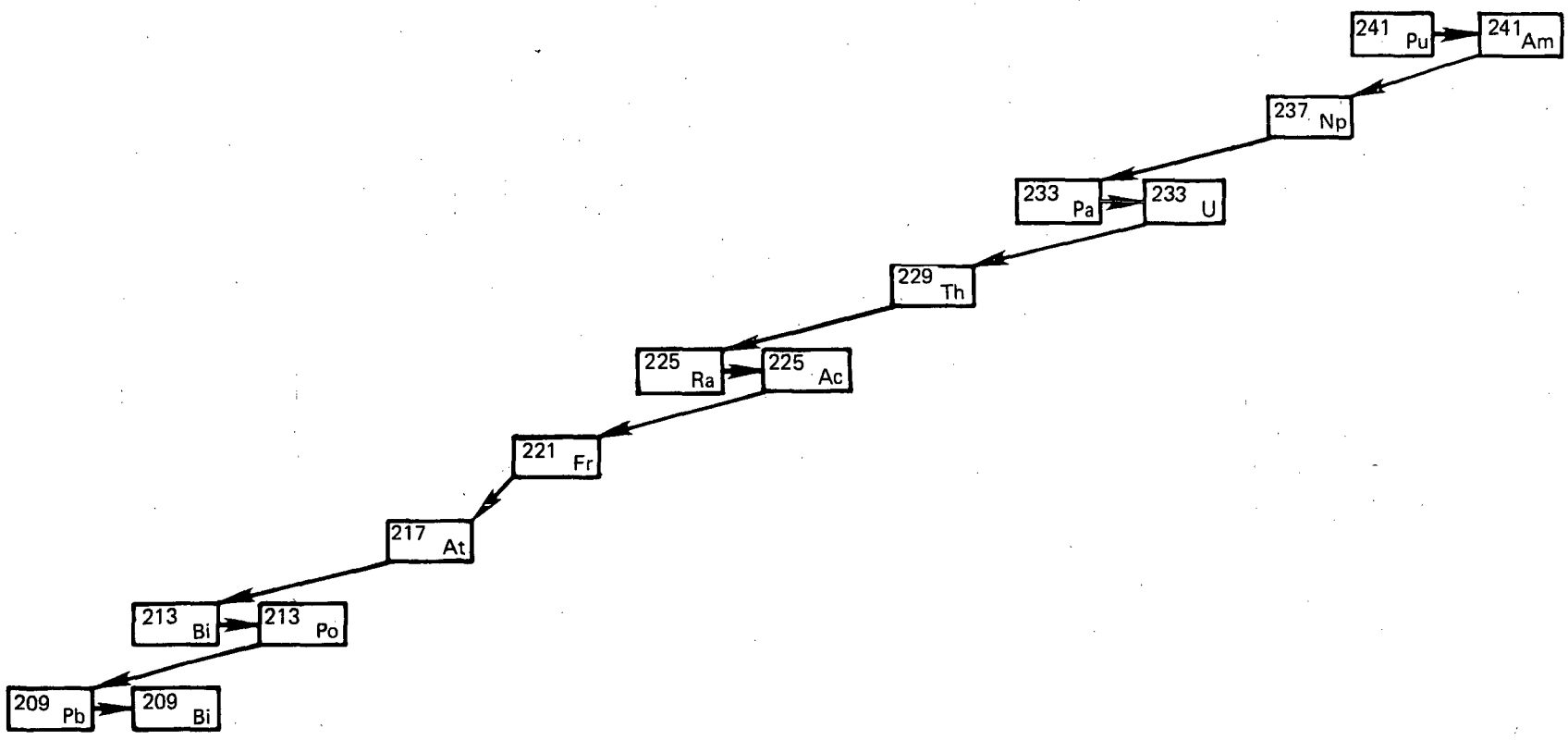


FIGURE 4.14 RADIOACTIVE DECAY CHAIN FOR THE (4n+1) SERIES. MINOR BRANCHING IS NOT SHOWN.

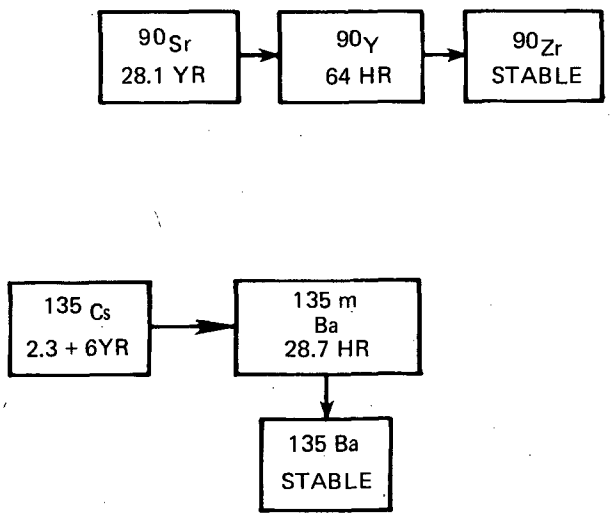


FIGURE 4.15 RADIOACTIVE DECAY CHAINS FOR ^{90}Sr AND ^{135}Cs. HORIZONTAL TRANSITIONS ARE BETA DECAYS.

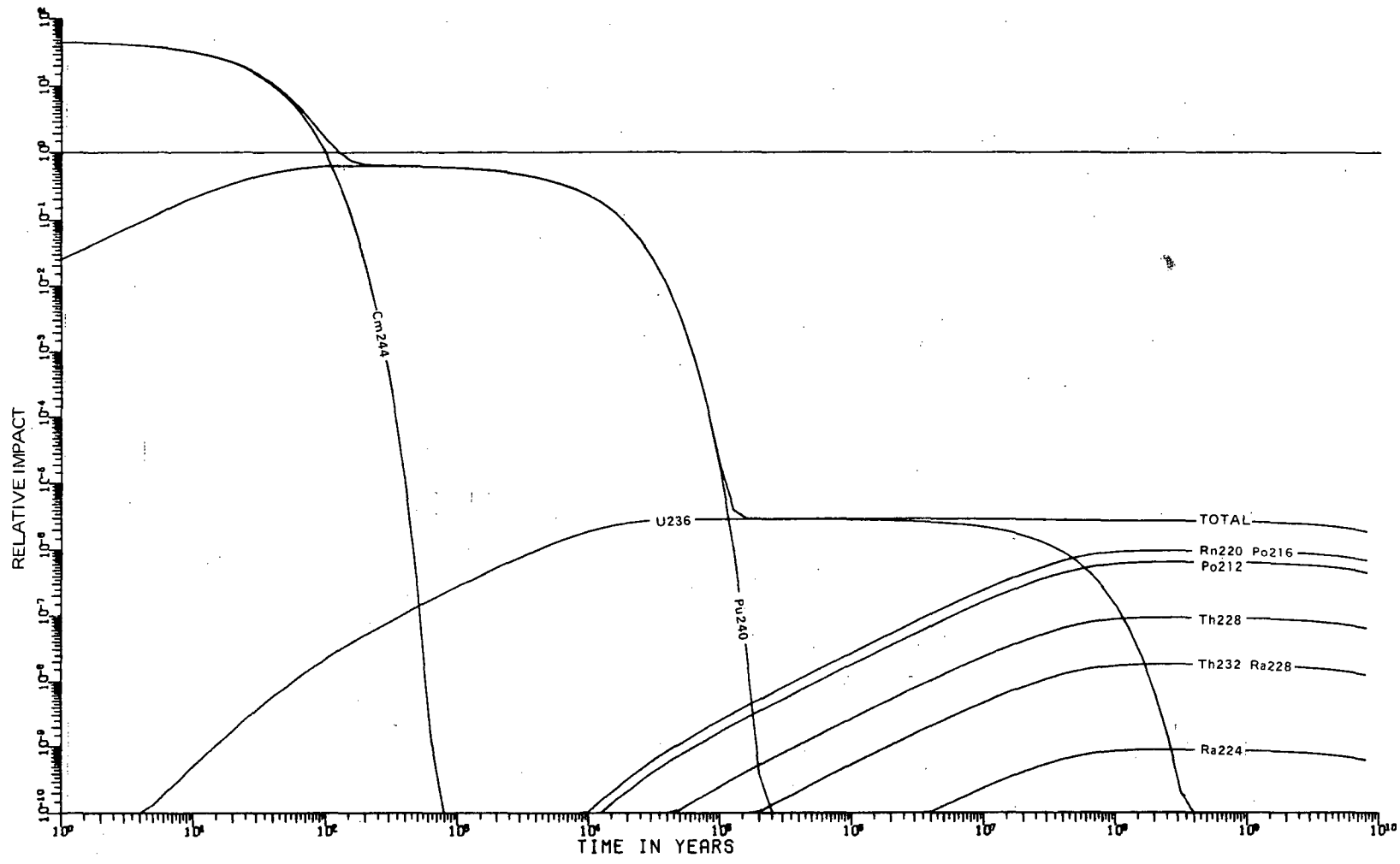


FIGURE 4.16 RELATIVE IMPACT, COMPARED TO PARENT, OF MEMBERS OF THE ^{244}Cm DECAY CHAIN

and 4.17 for the ^{244}Cm and ^{235}U decay chains. The total relative impact is also shown in the figures. The total relative impacts of the decay chain daughters are generally negligible except for the following chains: ^{244}Cm, ^{232}Th, ^{235}U, ^{242}Cm, ^{238}U, ^{241}Pu and ^{237}Np. However, the impacts for ^{235}U, ^{238}U and ^{237}Np do not exceed unity until after 10^4 years. It is not clear that the daughter effects beyond 10^4 years would influence the MAC's.

The MAC's for the parent nuclides are adjusted for daughter effects by the daughter adjustment factor (DAF).

$$MAC_{adj} = (MAC)(DAF) \tag{4.23}$$

$$DAF = \frac{1}{\sum_i RI_i} \tag{4.24}$$

where

MAC_{adj} = maximum allowable concentration adjusted for ingrowth of daughters

DAF = daughter adjustment factor

RI_i is defined in eq (4.22).

The MAC's, DAF, and MAC_{adj} are given in Table 4.31 for each of the parent nuclides considered. Except as noted, the DAF values are for 150 years following burial. Values are also given for ingestion events. These values, however, apply only to those events in which no nuclide transport occurs.

The ^{135}Cs and ^{90}Sr decay schemes are very simple. The daughter effects for the ^{90}Sr chain are negligible, but the daughter of ^{135}Cs does have a significant effect. Therefore, the ingrowth of daughters does limit the maximum allowable concentrations for ^{244}Cm, ^{135}Cs, ^{235}U, ^{242}Cm and ^{241}Pu.

One contributing factor to the DAF for those nuclides requiring adjustment is the fact that the default value of the MPC was used for some of the alpha emitting daughters. This default value for alpha emitters is extremely low, resulting in a considerable overestimation of the RI for those daughters. Most of the daughters in question have short half-lives compared to the parent. If the daughter MPC's are examined in more detail, the environmental impact of these daughters should be investigated and the DAF's adjusted accordingly.

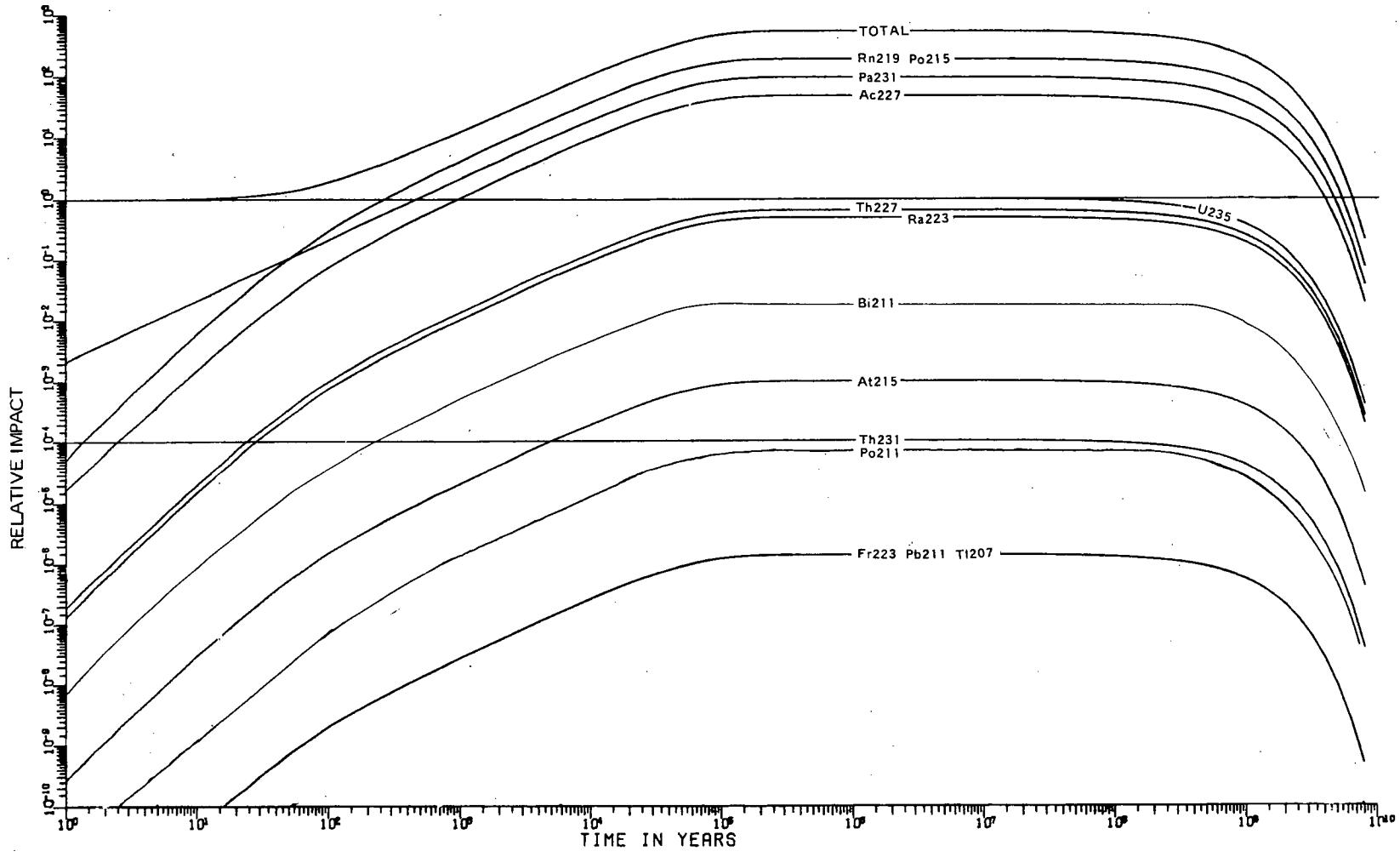


FIGURE 4.17 RELATIVE IMPACT, COMPARED TO PARENT, OF THE MEMBERS OF THE ^{235}U DECAY CHAIN

TABLE 4.31

MAXIMUM ALLOWABLE CONCENTRATIONS
ADJUSTED FOR DECAY DAUGHTERS

Nuclide	Parent MAC (No DAF) (Ci/m^3)	Daughter Adjustment Factor for Air	Adjusted MAC for Inhalation Events (Ci/m^3)	Daughter Adjustment Factor for Water	Adjusted MAC for Ingestion Events (Ci/m^3)
^{90}Sr	0.17	1.0	0.17	1.0	0.17
^{135}Cs	1.9	5.0 -3	9.5 -3	1.0	1.9
^{244}Cm	1.7 +3	6.2 -1	1.1 +3	9.1 -1	1.6 +3
^{240}Pu	1.0	1.0	1.0	1.0	1.0
^{243}Am	3.0	1.0	3.0	7.7 -1	2.3
^{239}Pu	1.0	1.0	1.0	1.0	1.0
^{235}U	0.30	7.7 -2[a]	2.3 -2	1.8 -1	5.4 -2
^{238}U	0.31	1.0	---	6.2 -2	1.9 -2
^{242}Cm	8.2 +3	1.25 -1	1.0 +3	---	---
^{238}Pu	3.7	1.0	3.7	1.0	3.7
^{242}Pu	1.1	1.0	1.1	1.0	1.1
^{241}Pu	5.0 +3	3.6 -2	1.8 +2	1.0 -2	5.0 +1
^{241}Am	3.6	1.0	3.6	1.0	3.6
^{237}Np	0.28	1.0[a]	0.28	7.7 -1	0.22

[a]Value of DAF at 1,000 years. The DAF decreases to a value of 1.7 -3 at about 10^6 years for ^{238}U and to a value of 1.9 -3 at about 10^5 years for ^{235}U and to a value of 3.3 -2 at about 10^6 years for ^{237}Np.

4.16.1.3 Impacts of Daughters on Groundwater Releases

In the groundwater events, the nuclides are assumed to be leached by rainwater, then migrate by saturated flow to a confined aquifer 10 m below the RCF, where they migrate 1,000 m to a surface river with a flow of 500 m^3/sec.

Analytical expressions were used in scoping calculations for the maximum value of the daughter concentrations, and detailed computer calculations were performed on potentially significant daughters. The expression for the ratio of the maximum release rate of the first daughter to the initial leach rate of the parent is given by C_m,[6]

$$C_m = \frac{1}{b} \left[\exp(bu) - 1\right] \exp(-au) \qquad (4.25)$$

where

$a = \lambda_1/\lambda_2$

$b = a - K_2/K_1$

$u = \lambda_2 K_1 x/V$

λ_i = decay constant for parent (i=1) or daughter (i=2) (yr^{-1})

K_i = sorption coefficient (from reference 2)

x = aquifer path length (1,000 m)

V = aquifer velocity (100 m/yr)

Equation (4.25) gives the C_m for any specific position u. Although the RCF parameters fix u for a given nuclide, C_m could be larger at a different location. Therefore, in estimating daughter effects from groundwater releases, the maximum value of C_m was used. This value is given by:

$$C_{max}^{(2)} = c\,\beta\,\alpha \qquad (4.26)$$

where,

$$C = K_1/K_2$$

$$\beta = (\lambda_2 K_2)/(\lambda_1 K_1)$$

$$\alpha = \frac{1}{1-\beta}$$

Although eq. (4.26) is derived for a two-member decay chain, it gives an upper bound for the ratio between any two adjacent members of a longer chain. However, the position u, at which the maximum could occur is not predicted for decay chain members three and larger.

The impact of the daughters through the various decay chains is generally negligible except for ^{226}Ra, and ^{222}Rn and ^{218}Po in the ^{238}Pu chain. Radon is an inert gas, but at the calculated concentrations it is very soluble. It is therefore assumed that all of the radon travels with the groundwater. One of the reasons the daughter impact for ^{222}Rn is so large is because the conservative default MPC value was used.

Preliminary results of detailed computer calculations of the parent and daughter release rates, as a function of space and time, indicate that the ^{226}Ra and ^{222}Rn impacts are also negligible compared to the parent impacts.

4.16.2 Additional Considerations

Although the final interfaces are quantified with MAC's, some of the actual limitations apply to the nuclide inventory. The MAC's presented in parentheses are specific to the RCF and RSLF site parameters and are a conservative limitation.

The application of the methodology to the nonradioactive/low-level interface is still in its initial stages. Consideration was given only to wastes that could be disposed of in a controlled sanitary landfill site. Additional attention should be directed toward quantification of this.

It is also noted that the high level waste repository was assumed to have negligible release and population exposures. Analysis of deep geologic disposal could modify some of the interface values.

The radioactive waste disposal classification methodology

is intended to be a pragmatic, generic approach to establish quantitative interfaces in a three-component classification system. It classifies wastes for disposal according to potential environmental impacts. The approach uses a given set of scenarios and is consistent and straightforward, but must be used with care. While some site-specific applications have been investigated (Chapter 5 of this report), this system is not intended to replace detailed site-specific environmental impact studies, but to be a tool to allow the classification of wastes for various disposal methods.

5. APPLICATIONS

This chapter contains the application or adaptation of the RWDCS methodology to some specific waste types and situations. These applications demonstrate the versatility and usefulness of the radioactive waste classification system methodology.

In Chapter 4, it was shown that limitations on the waste inventory could be related to concentration limits using the site specific parameters of the RCF. Upon that basis, the determination of Maximum Allowable Concentrations (MAC) is appropriate. Selection of the MAC for nuclides in the wastes implies that there are wastes with a range of concentrations with corresponding volumes that will be isolated if above, or contained if below the MAC.

The concentration of radioactivity in some waste forms or types does not range over large values. In some cases, the activity is determined by virtue of its natural composition or irradiation exposure history. Examples of this type of material are fuel element hulls, irradiated reactor components, uranium mill tailings, coal ash, and to a certain extent, already existing facilities containing or contaminated with radioactivity.

To demonstrate the applicability of the waste classification methodology, adaptation of the approach described in Chapter 4 to these types of waste materials is presented in this chapter. The objectives are basically to show how certain materials of given activity fit into the waste classification methodology, and to demonstrate the applicability of the general methodology.

The methodology is first applied to fuel element hulls and decommissioning wastes, and then to large-volume, low-specific activity wastes, i.e. uranium mill tailings and coal ash. An existing low-level waste shallow land burial facility, Maxey Flats, is the third example analyzed using the classification methodology. The final example to which the methodology is applied is a specific site that was not originally intended for the long-term disposal of radioactive wastes, the Latty Avenue site in Hazelwood, Missouri, where uranium ore residues were handled.

5.1 CLASSIFICATION OF SPECIFIC MATERIAL

In this chapter, the impact from the disposal of certain specific materials generated in the Light Water Reactor

(LWR) fuel cycle has been examined and appropriate classifications have been determined. The materials examined are fuel element "hulls," wastes generated during the decontamination and decommissioning of nuclear fuel cycle facilities and failed contaminated equipment.

By the end of 1977, there were 67 power reactors in operation in the United States, with another 138 scheduled to become operable by 1986.[22] The combined power rating for all of these reactors (205) would be about 200,000 MWe or 667,000 MWth. If a design life of 40 years is assumed for each of these reactors, it can be seen that by the year 2026 they will all be ready for decommissioning and decontamination. Additionally, approximately 700 m^3 of cladding material will be generated from each reactor over its 40 year lifetime.

It is estimated that decontamination of these 205 reactors would produce approximately 17,000,000 metric tons of radioactive waste containing nearly 7.0×10^7 Ci of activity from the reactor components and the biological shields alone. Much of the radioactivity in this waste will be induced activity which will decay over a relatively short period of time, leaving the residue relatively safe for contact with the environment. The decision to place this waste into either permanent geological isolation, shallow land burial or intermediate ground burial can have an important impact on the nuclear industry.

The RWDCS methodology developed in Chapter 4 has been adapted to analyze the foregoing types of materials which are generated with specific, predictable activity levels, based on their histories of exposure. The most restrictive pathways for human exposure identified in Chapter 4 were analyzed for these specific materials to determine their classification characteristics.

5.1.1 Adaptation of RWDCS Methodology

Based on the RWDCS methodology presented in Chapter 4, the specific materials are analyzed to determine potential exposure mechanisms. For these special materials, the activity is either tightly bound to the surfaces or is induced throughout the volume of the materials, making it perhaps less susceptible to potential migration pathways than are routine wastes.

Direct corrosion of the surfaces of the specific materials will, however, lead to a possible mobilization of both surface contamination and the volumetric activity in the

volume corroded. In the case of the reclaimer type events, the individuals involved will be exposed to relatively small volumes of waste at any given time. Therefore, the concentration of activity in the wastes to which they are exposed are the basis for interface limits. For some other pathways, however, the total inventory of radioactivity at a given disposal site is the basis for interface limits. In these cases, the concentration limits are derived from the volume of waste disposed at the specific disposal facility.

Because of the nature of the specific materials analyzed in this section, the maximum allowable concentrations of activity in the waste matrix and the maximum allowable surface contamination levels are of primary concern. Surface contamination levels are related to volumetric concentrations to facilitate use of the classification system and to assure that the most restrictive cases are not overlooked. The dose guidelines used in Chapter 4 are incorporated in the analyses in this section.

5.1.2 Description of Specific Materials

The three specific types of radioactive waste investigated in this study are fuel element hulls, decommissioning and decontamination wastes and contaminated, failed equipment. Each of these categories of waste are discussed in detail below.

5.1.2.1 Fuel Element Hulls

In the reprocessing of commercial power reactor fuel elements, there is a considerable volume of undissolved waste residue composed mainly of fragments of Zircaloy, stainless steel and Inconel fuel element cladding material and fuel-bundle hardware. These materials are referred to as cladding hulls. Hulls have a relatively low bulk density (1.1 g/cm^3), are highly activated from neutron bombardment and are contaminated with long-lived transuranic elements and fission products.[23]

The transuranic activity is derived from two different sources. The first source is internal to the metal and cannot be removed through any process short of complete dissolution or resmelting. Approximately .09 Ci/m^3 of this internal activation is present in typical hulls. The second source is surface contamination due partly from leakage of the fuel into the reactor coolant, from fuel element ruptures and partly due to undissolved fuel. This

contamination is estimated to be of the order of 2.3 to 4.6 Ci/m^3.[24]

The zirconium used for cladding in reactor fuel rods usually contains traces of uranium (1 ppm) as an impurity from its original refining.[23] This uranium is uniformly dispersed throughout the metal and is partially transmuted to transuranium elements or fissioned during the time it is in the reactor. About 97% of the internal activity from the transuranic elements is due to ^{241}Pu after five years of cooling time.[24] The total transuranic activity amounts to about 90 $\eta Ci/g$. The complete transuranic contamination due to activation as calculated by the ORIGEN[24] code, after five years of cooling, is shown in Table 5.1.

Other non-transuranic activation products present in the Zircaloy are ^{60}Co, ^{55}Fe, ^{125}Sb, ^{63}Ni, and ^{125}Te. After five years of cooling, the activity of these elements would be 900 $\mu Ci/g$ of Zircaloy and after 100 years would have decayed to about 18 $\mu Ci/g$.[25]

Less data are available for uranium impurities in stainless steel and Inconel. However, they are likely to be much smaller and a similar calculation could be made when impurity data become available.

The volume of the hulls is projected to be approximately 0.5 m^3 for each ton of fuel processed.[25] Reference 25 also estimates the annual fuel requirement to be 35 metric tons of heavy metal (MTHM) per reference reactor year (RRY) of power generation. If the hulls are compacted to 70% of theoretical density (four to one compaction) the volume per RRY would be about 4 m^3.

Projections of cladding hull volumes by Blomeke[26] are similar to the above estimate.

The RCF described in Chapter 4 contains 1000 RRY of waste, or about 630,000 m^3 and 6×10^6 Ci of low-level waste. The volume of hulls would be 400 m^3 or 0.6% of the total volume. At 70% of theoretical density, the mass of the hulls would be 1.8×10^7 kg, and the base metal transuranic activity would be 1600 Ci.

The surface contamination of the cladding hulls reported in Table 4.5 of reference 23 amounts to 37 g of Pu per cubic meter of hulls in the U + Pu recycle process. If U only is recycled, the Pu contamination would be only 18 g/m^3. Using a specific activity for the plutonium mixture of 0.27 Ci/g of plutonium,[5] at the higher contamination level,

TABLE 5.1

TRANSURANIC CONTAMINATION IN ZIRCALOY DERIVED FROM IRRADIATION[a]

	Concentration (ppb)	Activity (nCi/g)
Uranium	972	0.003
Neptunium	0.1	0.02
Plutonium	9	88
Americium	0.3	0.9
Curium	0.03	2.2
Others	< 0.4	< 0.1
TOTAL	981	91.1

[a]Calculations by the ORIGEN code, assuming 253 kg of Zircaloy per metric ton of uranium exposed to a burnup of 33,000 MWd per metric ton of uranium and five years of cooling.

gives 10 Ci/m^3, of which 1.5 Ci/m^3 is ^{239}Pu. This value is of the same order of magnitude as the low-level/high-level MAC for ^{239}Pu given in Chapter 4. However, using the method described in Chapter 4 for mixtures of nuclides indicates that this level of contamination would be above the combined MAC.

Dillon[23] reports that in decontamination studies at PNL, fuel cladding could be decontaminated by a factor greater than 10^3 with a substantial portion of the residual alpha activity being due to the base metal activation. The decontamination was achieved by treatment at 600°C in hydrogen fluoride followed by solutions of ammonium oxalate, ammonium citrate, ammonium fluoride and hydrogen peroxide.

5.1.2.2 Decommissioned Reactor Components

Decontamination and decommissioning (D&D) of the commercial power reactors is not expected to affect waste management facilities until approximately the year 2000. The Dresden 1 reactor was placed in operation in 1960 and the Yankee in 1961. By August of 1963, there were only five commercial power reactors in operation with a combined new power rating of 773 MWe.[22] If a useful life of 40 years is assigned to these plants, then the first D&D efforts for these power reactors would be expected to begin around the turn of the century.

At the present time, reactor D&D experience is relatively limited. Since 1960, a total of 65 nuclear reactors have been decommissioned, which include five nuclear power plants, four demonstration nuclear plants, six test reactors and 28 critical facilities.[27] Many of these reactors were either mothballed or entombed so that little directly applicable reactor decontamination experience can be gleaned from their decommissioning.

One of the first reactors decontaminated was the SL-1 reactor at the INEL, starting in 1961. However, this reactor was decommissioned after a criticality incident, which greatly multiplied its contamination problems.

A reactor more recently decontaminated was the EBR-1 also at the INEL. This reactor was designated as a national historical monument. Consequently, much of the reactor and the associated equipment were merely sealed up to prevent contact with the public, and the building was decontaminated only in the areas where public tours would pass.[28]

A few other small reactors have been at least partially decontaminated, but in general they are not comparable to the power reactors.

The Hallam Nuclear Power Facility was largely entombed in place. The Piqua reactor was partly decontaminated and partly sealed up. The Carolinas Virginia Tube reactor was mothballed. When the Pathfinder reactor was shut down, it was partially decontaminated and converted into a fossil-fuel system.

The Bonus reactor was partly decontaminated and partly entombed, and the Peach Bottom 1 reactor was mothballed.[27]

The best record and description of actual reactor D&D wastes comes from the Elk River reactor in Minnesota. This reactor was very small (58.3 MWth) compared to today's average power reactor (2500 to 3600 MWth) and only operated for 3.5 years; however, the radioactive waste produced by the D&D of the Elk River reactor provides an indication of wastes that may be generated in the future D&D of power reactors. The mass, volume and gross activity of the radioactive waste produced by the D&D of the Elk River reactor are summarized in Table 5.2.[29] The waste is divided into five categories or groups, i.e. reactor pressure vessel (RPV), internal materials, external materials, biological shield and miscellaneous. The volume and mass of miscellaneous waste are given, but not the activity. It is probable that the activity of this waste was very low; undoubtedly far below the levels of the other four categories.

Also given in Table 5.2 are the extrapolations of waste for four larger sizes of power reactors. These calculated masses, volumes and activities are strictly scaled-up from the Elk River reactor, without differentiating between BWR's, PWR's and HTGR's.

Since it was recognized that the waste would not increase linearly with the power of the reactor, an exponential scale-up was utilized. In estimating costs for varying sizes of power plants when the cost C_1 of a plant of capacity R_1 is known, the cost of the second unit C_2 is determined by the relation $(C_2/C_1) = (R_2/R_1)^p$ where p is between 0.6 and 1.0.[30] Recent developments in the construction industry (lengthened schedules, increasing inflation rate, etc.) indicate that p is at least 0.8, which was chosen for this work. It was further assumed that the mass and volume of the plant are proportional to cost such that:

TABLE 5.2

ESTIMATED RADIOACTIVE WASTE FROM POWER REACTORS

Component	Elk River[a]	1000 MWth	2000 MWth	3000 MWth	3600 MWth
Group 1 RPV					
mass (Kg)	3.6 +4	3.5 +5	6.1 +5	8.5 +5	9.8 +5
volume (m^3)	4.6 +0	4.5 +1	7.9 +1	1.1 +2	1.3 +2
activity (Ci)	1.1 +3	1.8 +4	332 +4	4.4 +4	5.1 +4
Group 2 Internals					
mass (Kg)	8.1 +3	7.9 +4	1.4 +5	1.9 +5	2.2 +5
volume (m^3)	1.1 +0	1.0*+1	1.8 +1	2.5 +1	2.9 +1
activity (Ci)	8.7 +3	1.4 +5	2.5 +5	3.5 +5	4.0 +5
Group 3 Externals					
mass (Kg)	5.4 +4	5.3 +5	9.2 +5	1.3 +6	1.5 +6
volume (m^3)	5.3 +0	5.2 +1	9.0 +1	1.3 +2	1.4 +2
activity (Ci)	4.4 +2	7.2 +3	1.3 +4	1.7 +4	2.0 +4
Group 4 Biological Shield					
mass (Kg)	3.9 +4	3.7 +5	6.5 +5	9.1 +5	1.0 +6
volume (m^3)	5.9 +0	5.7 +1	1.0 +2	1.4 +2	1.6 +2
activity (Ci)	5.8 +0	9.6 +1	1.7 +2	2.3 +2	2.7 +2
Group 5 Miscellaneous					
Rad. & Contaminated Mtls.					
mass (Kg)	1.1 +6	1.1 +7	1.8 +7	2.6 +7	2.9 +7
volume (m^3)	1.3 +3	1.3 +4	2.3 +4	3.1 +4	3.6 +4
Contaminated Concrete					
mass (kg)	3.0 +6	3.0 +7	5.1 +7	7.1 +7	8.2 +7
volume (m^3)	1.3 +3	1.2 +4	2.1 +4	3.0 +4	3.4 +4

[a]Elk River Dismantling Plan (DOCKET - 1151 - 46)

$$C_2/C_1 = M_2/M_1 = V_2/V_1 = (R_2/R_1)^{0.8} \qquad (5.1)$$

This scaling relationship was applied component by component.

In extrapolating the activity levels from those of the Elk River reactor, A', it was assumed that the new equilibrium activities are proportional to the mass, i.e.:

$$A_2/A_1 = M_2/M_1 \qquad (5.2)$$

where $A_1 = A'/f$ and f is the fraction of activation equilibrium achieved, such that $A_2/A' = 1/f\ (R_2/R_1)^{0.8}$. This relationship is based on the assumption that the neutron flux levels do not vary significantly with reactor capacity.

The fraction of activation equilibrium achieved in any reactor can be determined from the mass balance equation:

$$dN/dt = \Sigma_a\ \phi - \lambda N \qquad (5.3)$$

where

σ_a = macroscopic absorption cross section, cm^{-1}

λ = decay constant, yr^{-1}

ϕ = neutron flux, neut/cm^2.yr

t = time of reactor operation, yr

N = atom density, cm^{-3}

The solution is

$$N(t) = \frac{\Sigma_a\ \phi}{\lambda}\ (1 - e^{-\lambda t}) \qquad (5.4)$$

Since N(t) is proportional to $(1 - e^{-\ t})$, the fraction of equilibrium achieved at a reactor that operates less than 40 years can be expressed by:

$$f = (1 - e^{-\lambda t})/(1 - e^{-40\lambda}) \qquad (5.5)$$

For the eight nuclides which comprise more than 99% of the activity at Elk River, the fractions of equilibrium achieved are tabulated in Table 5.3.

The extrapolated activity of each nuclide for larger plants is given by the relation:

$$A_2 = \frac{A'}{f} \left(\frac{R_2}{58.3}\right)^{0.8} \qquad (5.6)$$

In the first column of Table 5.4, a distribution of the calculated activation nuclides in the Elk River waste is given, based on actual operating history. Using the scaling relationship given in the foregoing paragraphs, a similar distribution was calculated for reactors of 1000 MWth, 2000 MWth, 3000 MWth, and 3600 MWth. These data are also given in Table 5.4.

It can be seen from Table 5.4 that most of the activity is contributed by the ^{55}Fe, ^{60}Co and ^{63}Ni. In fact, these three isotopes contribute 99.6% of the total activity. Five additional isotopes ^{57}Co, ^{54}Mn, ^{49}V, ^{125}Sb and ^{152}Eu account for another 0.3% of the total activity. Therefore, in calculating the relative hazards for this waste, only these eight isotopes were considered. In wastes from typical power reactors, other nuclides may also be important, such as ^{59}Ni and the transuranics. Because the Elk River reactor operated such a short time, these additional nuclides were not significant. The behavior of the eight nuclides listed will be representative of those likely to be important in decommissioning wastes, however.

A comprehensive study of reactor decommissioning alternatives was undertaken by the Atomic Industrial Forum, Inc. (AIF) and reported in November 1976.[27] In this study, a projection of the mass and the activity of wastes generated in the D&D of commercial power reactors was made but no indication of the volume of waste was given. Without bulk density data, no conversion to volume can be made.

A comparison of the waste projections from the AIF report[27] and the FB&DU scale-up from the Elk River reactor is worthwhile. The AIF report presents considerable detail regarding the activation of the various reactor components. Most of these components fit into one of three categories used for the Elk River scale-up, i.e. reactor pressure vessel, internal components or biological shield. There does not appear to be a group comparable to the Elk

TABLE 5.3

NUCLIDE EQUILIBRIUM FRACTION AT ELK RIVER

Nuclide	Half-Life (yrs)	Fractional Equilibrium At Elk River
^{49}V	0.90	0.93
^{54}Mn	0.86	0.94
^{55}Fe	2.7	0.59
^{57}Co	0.74	0.96
^{60}Co	5.3	0.37
^{63}Ni	100	0.10
^{125}Sb	2.7	0.59
^{152}Eu	13	0.19

TABLE 5.4

DISTRIBUTION OF NUCLIDES IN ELK RIVER WASTE AND POWER REACTORS

Nuclide	Elk River (Ci)	1000 MWth (Ci)	2000 MWth (Ci)	3000 MWth (Ci)	3600 MWth (Ci)
^{49}V	2.3 +0	2.4 +1	4.2 +1	5.9 +1	6.8 +1
^{54}Mn	3.3 +1	3.4 +2	5.9 +2	8.3 +2	9.5 +2
^{55}Fe	5.6 +3	9.2 +4	1.6 +5	2.2 +5	2.6 +5
^{57}Co	2.0 +0	2.0 +1	3.5 +1	4.8 +1	5.5 +1
^{60}Co	3.9 +3	1.0 +5	1.8 +5	2.5 +5	2.8 +5
^{63}Ni	2.9 +2	2.8 +4	4.9 +4	6.9 +4	7.9 +4
^{125}Sb	2.2 +0	3.7 +1	6.4 +1	8.9 +1	1.0 +2
^{152}Eu	2.1 +0	1.1 +2	1.9 +2	2.6 +2	3.0 +2
TOTAL	9.8 +3	2.2 +5	3.9 +5	5.4 +2	6.2 +5

River external components. The AIF report also discusses waste from three types of reactors; namely, PWR's, BWR's and HTGR's. Since the HTGR is a totally different reactor concept than the Elk River reactor, only the PWR and BWR data are compared with the FB&DU scale-up data. A comparison of these data are presented in Table 5.5. The agreement among the masses in the various groupings of waste is encouraging. All projections are of the same order of magnitude except for the biological shield for the PWR. However, this is not truly a biological shield for the PWR, but rather a neutron shield; and therefore, would not be expected to be as massive as the biological shield of the BWR type reactors. The internal components are in particularly close agreement, and since these are the materials with the highest activations, they are the most critical of all the waste materials.

The activity projections of the waste groups are also in agreement within one order of magnitude with the exception of the activity in the BWR biological shield. For the BWR biological shield, the AIF projection forecasts only approximately five curies of activity, while the FB&DU model gives 120 Ci. The FB&DU projections are generally on the conservative side. The activity comparison was made at five years after shutdown, which was considered to be the closest AIF projection to the Elk River reactor decontamination project.

5.1.2.3 Decommissioned Fuel Reprocessing Plants

Closely related to the decontamination of power reactors is the decontamination of fuel reprocessing plants. At the present time there is a moratorium on fuel reprocessing, but in the event that reprocessing is eventually reestablished, an evaluation of the decontamination wastes from retired fuel reprocessing plants would also be needed. A recently completed study by Battelle Pacific Northwest Laboratory,[31] gives a detailed description of the decontamination of a Reference Fuel Reprocessing Plant (RFRP). A tabulation of these projections of waste components including the isotopic distribution, mass and volume are given in Table 5.6. Generally, the nature of the contamination is different than that from reactors in that it consists principally of surface contamination of transuranics and mixed fission products rather than activation isotopes.

In the decontamination of an RFRP, it was projected that 6000 metric tons of low-level waste containing 40,000 curies of activity in 3,500 m^3 would be produced.

TABLE 5.5

COMPARISON OF AIF AND ELK RIVER SCALE-UP WASTE PROJECTIONS

	Mass			Activity (@5 years)			Volume		
	AIF-PWR 3411 MW_{th} Wt. (Kg)	AIF-BWR 3579 MW_{th} Wt. (Kg)	FB&DU 3600 MW_{th} Wt. (Kg)	AIF-PWR 3411 MW_{th} Ci	AIF-BWR 3579 MW_{th} Ci	FB&DU 3600 MW_{th} Ci	AIF-PWR 3411 MW_{th} M^3	AIF-BWR 3579 MW_{th} M^3	FB&DU 3600 MW_{th} M^3
Reactor Pressure Vessel	3.8×10^5	3.0×10^5	9.8×10^5	2.1×10^4	4.7×10^3	5.1×10^4			1.26×10^2
Internals	1.6×10^5	9.7×10^4	2.2×10^5	5.8×10^6	5.1×10^5	4.0×10^5			2.89×10^1
Externals			1.5×10^6			2.0×10^4			1.44×10^2
Biological Shield	3.7×10^4	6.2×10^5	1.0×10^6	3.5×10^2	4.7	2.7×10^2			1.60×10^2

TABLE 5.6

FUEL REPROCESSING PLANT DECOMMISSIONING WASTE[a]

Nuclide	Process Cell Walls (Curies)								HLLW Storage Tanks (Curies)		Waste Solidification Plant (Curies)				
	Remote Process Cell	High Inter-mediate Level Cell	High Level Cell	Uranium Product Cell	Inter-mediate Level Cell	Plutonium Product Cell	Pu Nitrate Storage and Loadout	Remote Mainte-nance Cell	Tank Bottom	Tank Walls and Internals	Waste Vitrifi-cation Cell	Canister Decontami-nation Cell	Hot Pipe Trench	Other	Totals
^{90}Sr, ^{90}Y	1,450	17	100	11	11	---	---	10	110	8,400	1,440	20	<1	---	11,570
^{106}Ru, ^{106}Rh	300	5	22	2	2	---	---	3	24	1,800	40	---	---	---	2,198
^{134}Cs	200	2	13	1	1	---	---	1	31	2,400	80	---	---	---	2,729
^{137}Cs	1,130	13	77	9	9	---	---	6	85	6,600	1,125	15	<1	---	9,070
^{137}Ba	1,060	13	73	9	9	---	---	6	79	6,000	1,050	15	<1	---	8,315
^{144}Ce, ^{144}Pr	310	5	22	2	2	---	---	3	14	1,100	25	---	---	---	1,483
^{147}Pm	150	2	10	1	1	---	---	1	24	1,800	70	---	---	---	2,059
^{154}Eu	60	---	5	---	---	---	---	---	---	68	55	---	---	---	188
^{238}Pu	28	---	---	---	---	50	7	---	---	58	<1	---	---	---	144
^{241}Pu	600	---	---	---	---	990	100	---	---	56	8	---	---	---	1,754
^{241}Am	55	---	7	---	---	35	9	---	---	54	12	---	---	---	172
^{244}Cm	70	<1	7	<1	<1	---	---	<1	6	460	70	<1	---	---	618
TOTALS	5,413	58	336	36	36	1,075	116	31	373	28,796	3,976	51	<3	---	45,300
Mass (Kg)															
Concrete	284,000	204,000	164,000	176,000	197,000	59,000	---	12,700	---	---	5,500	2,500	1,400	723,100	1.8×10^{6}
S.S. Liners	5,500	3,800	2,500	2,900	3,500	1,900	---	21,000	---	86,000	32,000	18,000	28,000	244,400	4.5×10^{5}
S.S. Pipe	---	---	---	---	---	---	---	---	---	565,000	---	---	---	850,000	1.4×10^{6}
Filters	---	---	---	---	---	---	---	---	---	---	---	---	---	8,900	8.9×10^{3}
Equipment	111,000	119,000	176,000	121,000	209,000	65,000	27,000	54,000	---	304,000	---	---	---	513,700	1.7×10^{6}
Other	---	---	---	---	---	---	---	---	---	---	---	---	---	704,000	7.0×10^{5}
TOTALS	400,500	326,800	342,500	299,900	409,500	125,900	27,000	87,700	---	955,000	37,500	20,500	29,400	3,044,100	6.1×10^{6}
Volume (m^3)															
Concrete	119	85	68	74	82	25	---	5	---	---	3	1	1	605	1,068
S.S. Liners	0.7	0.4	0.3	0.4	0.4	0.2	---	2.6	---	8.2	4	2.2	3.4	24	47
S.S. Pipe	---	---	---	---	---	---	---	---	---	64	---	---	---	187	251
Filters	---	---	---	---	---	---	---	---	---	---	---	---	---	45	45
Equipment	69	74	110	150	130	41	17	34	---	190	---	---	---	321	1,136
Other	---	---	---	---	---	---	---	---	---	---	---	---	---	925	925
TOTALS	189	159	178	224	212	66	17	42	---	262	7	3	4	2,107	3,472

[a] From reference 31

In a reference containment facility containing 1000 RRY of waste, it was projected in Chapter 4 that there would be 6.3×10^5 m^3 of low-level waste with an activity level of 6×10^6 Ci. The decommissioning waste projected for one RFRP would therefore constitute 0.5% of the total volume, and 0.3% of the total activity.

The Battelle study was based upon the decontamination of the Barnwell Nuclear Fuel Plant with an annual throughput of 1500 MTHM. Using 35 MTHM/RRY and a 30 year plant lifetime provides reprocessing for about 1300 RRY's worth of waste from each RFRP.

5.1.2.4 Failed Equipment and Noncombustible Waste

In the operation of a power reactor, a mixed oxide fuel fabrication plant or a fuel reprocessing plant, a generally predictable amount of noncombustible waste will be generated. This waste will comprise small items such as worn-out hand tools, used light bulbs, wire, metal scraps, laboratory glassware, and also larger items such as pumps, dissolvers, solvent extraction columns, process piping, and heating, ventilating and air conditioning components from possibly contaminated systems. Anticipated volumes of this material range from 0.02 to 0.4 m^3/MTHM with an average of 0.1 m^3/MTHM.[26] Radioactivities also cover a wide range and are difficult to quantify.

Using 35 MTHM/RRY and the average volume of anticipated failed equipment given above, it can be seen that 3.5 m^3 of failed equipment would be produced per RRY.

In a reference containment facility containing 1000 RRY of waste there would be 3.7×10^3 m^3 of this type of waste. This would be 0.6% of the waste volume in the RCF.

5.1.3 Interface Limits for Specific Materials

In this section, interface limits are determined for the specific nuclear materials described in Chapter 5.1.2. The specific materials are grouped according to three categories:

1. Extremely small pieces of material such as hulls.

2. Surface contamination on large pieces of material such as plutonium plate-out on reprocessing equipment.

3. Volumetric activation on large pieces of material such as activation of the reactor pressure vessel and internals.

In investigating each of these categories for classification purposes, mechanisms for initiating the transport of the contamination through the environment are examined. Once the material is mobilized and available for transport, the analyses of potential pathways from Chapter 4 with associated MAC's are applicable. To relate the MAC's from Chapter 4 with the surface activity on specific materials, adjustment factors are derived appropriate to the mechanism for transporting the activity. The resultant adjusted maximum limiting values are compared to the expected ranges of activity on or in the specific materials as presented in Chapter 5.1.2.

5.1.3.1 Fuel Element Hulls

Fuel element hulls contain both volumetric activation products within the base metal and surface contamination. About 99% of the activation product activity is attributed to ^{241}Pu and curium. Corrosion of the hulls and the resultant availability of the activity is the most restrictive potential mechanism for human exposure. As the hulls corrode, the activity becomes entrained in the corrosion products, postulated to be in a powdery form. The finely divided corrosion products are then available for transport in the same manner as described in Chapter 4, and could be inhaled by a reclaimer, or be moved by the groundwater.

Therefore, for the volumetric activation products, the Maximum Allowable Concentrations in Hulls (MACH) for each nuclide of importance can be related to the MAC's derived in Chapter 4 by correcting for changes in composition and density as the hull base metal corrodes:

$$MACH = \frac{Wo\ \rho_m}{Wm\ \rho_o} MAC \qquad (5.7)$$

where,

MACH = maximum allowable volumetric activation in hulls (Ci/m^3)

MAC = maximum allowable concentration in waste from Chapter 4 (Ci/m^3)

Wo = molecular weight of base metal corrosion product (g/mole)

W_m = molecular weight of base metal (g/mole)

ρ_m = density of base metal (g/cm^3)

ρ_o = density of corrosion product (g/cm^3)

For zirconium oxidizing to ZrO_2, eq (5.7) gives MACH = 1.6 MAC. Using the MAC's from Table 4.14 in Chapter 4 (corrected for decay daughters from Table 4.31 in Chapter 4), the values given in Table 5.7 are obtained for volumetric activation limits in hulls.

The foregoing analysis presumes that a significant amount of corrosion has taken place at the time of the exposure event. Therefore, it is of some interest to investigate expected corrosion rates in buried hulls. The corrosion rate in soil of zirconium, stainless steel or Inconel are all extremely low. Zirconium has been reported to corrode less than 0.002 mils per year in 40% hydrochloric acid at temperatures up to 100°C.[32] Specific data for the corrosion of zirconium in soil could not be obtained but zirconium is reported to be comparable to nickel, nickel-base alloys, high-alloy stainless steel and titanium.[33]

A sample calculation may illustrate the effect of corrosion. The corrosion rate in soils for a high-alloy stainless steel is on the order of 0.002 mg/100 cm^2/day,[33] or equivalently, 9×10^{-7} cm of corrosion per year. Assuming that zirconium also corrodes at this rate, the release of transuranic isotopes from zirconium hulls would be about 6.6×10^{-13} Ci/cm^2/yr.

To determine the relationship between the mass of the compacted hulls and the hulls surface area, the following calculation was made. Typical fuel element cladding is about 1.43 cm outside diameter and 0.0725 cm wall thickness. These hulls are generally chopped into pieces 2 to 4 cm long. The surface area of a tube 1 cm long with these dimensions is:

$$A_s = \pi L \left[D_1 + (D_1 - 2W_{th})\right] \qquad (5.8)$$

where,

A_s = surface area (inside and out)

L = length (1 cm)

D_1 = outside diameter (1.43 cm)

TABLE 5.7

MAXIMUM ALLOWABLE VOLUMETRIC CONTAMINATION CONCENTRATIONS IN HULLS

Nuclide	Original MAC (Ci/m^3)	MACH[a] (Ci/m^3)
^{49}V	1.1 +8	1.8 +8
^{54}n	1.0 +6	1.6 +6
^{55}Fe	3.3 +7	5.3 +7
^{57}Co	3.0 +7	4.8 +7
^{60}Co	5.5 +4	8.8 +4
^{63}Ni	3.0 +5	4.8 +5
^{90}Sr	1.3	2.1
^{125}Sb	4.3 +5	7.2 +5
^{135}Cs	20	3.2 +1
^{152}Eu	6.7 +4	1.1 +5
^{232}Th	1.2 -1	1.9 -1
^{235}U	4.1	6.6
^{238}U	12	1.9
^{237}Np	1.8	2.9
^{238}Pu	2.5	4.0
^{239}Pu	1.0	1.6
^{240}Pu	1.0	1.6
^{241}Pu	5.0 +3	8.0 +3
^{242}Pu	1.1	1.8
^{241}Am	3.6	5.8
^{243}Am	3.1	5.0
^{242}Cm	8.2 +3	1.3 +4
^{244}Cm	2.4 +2	3.8 +2

[a]Maximum allowable concentration limit for hulls where activity is related to total hull volume.

W_{th} = wall thickness (0.0725 cm) (neglecting the surface on the ends)

For one hull segment then, A_s = 8.53 cm^2. The mass of the same metal piece is:

$$M = \frac{\rho t \pi L}{4} \left[D_1^2 - (D_1 - 2W_{th})^2 \right] \qquad (5.9)$$

where,

ρ_t = theoretical density of the metal = (6.4 g/cm^3)

M = mass (grams)

For one segment, the mass is 1.98 g. The ratio of surface area to mass for the hulls is:

$$R = \frac{A_s}{M} = \frac{8.53 \text{ cm}^2}{1.98 \text{ g}} = 4.31 \text{ cm}^2/\text{g} \qquad (5.10)$$

This relationship would remain relatively constant, even when the hulls are compacted, unless they are embrittled to the degree of shattering when compacted.

With this relationship, the release of activity per gram of hulls can be calculated. From the corrosion rate calculations, it was determined that the release of activity per unit area is about 6.6 x 10^{-13} $Ci/cm^2/yr$. Multiplying this quantity by the ratio of surface area to mass gives a corrosion release rate density of 2.9 x 10^{-12} Ci/g/yr.

The annual activity released per RRY would be 52 μCi of volumetric induced activation from hull corrosion. Equivalently, it would take about 3.2 x 10^4 years for complete corrosion of one RRY of hulls at this rate.

The major limiting consideration for hulls, however, is the surface contamination from transuranics and fission products. The limiting release mechanism for surface contamination is the corrosion of the contaminated surface with no further dilutions. The contaminated material is then in a form compatible with the pathway analysis of Chapter 4. For the reclaimer events, the exposure is postulated to occur 150 years after placement. The amount of corrosion that has occurred in 150 years is estimated from the corrosion rate, R from eq (5.10), and the time to the exposure event.

$$CF = (7.3 \times 10^{-6}\ g/cm^2yr)(4.3\ cm^2/g)(100\ yrs) = 3.1 \times 10^{-3}\ g\ corroded/g\ hulls, \quad (5.11)$$

where CF is the fraction of hulls corroded in 100 years. Therefore, if the surface contamination is expressed in Ci/m^3 of hulls, at the end of 150 years this contamination will essentially all be contained in a volume of oxide which is 3.1×10^{-3} times smaller than the original volume of the hulls, with a corresponding increase in concentration of about a factor of $300e^{-\lambda t}$ greater than when buried. The MACH would therefore be 3.1×10^{-3} times the MAC from Chapter 4 for those pathways that are concentration limited. For inventory limited pathways the MACH is the same as the MAC from Chapter 4.

The MACH of ^{239}Pu would therefore be 3×10^{-3} Ci/m^3. The projected concentration for surface plutonium contamination in hulls is 37 g/m^3, which is equivalent to 10 Ci/m^3, of which 2.3 Ci/m^3 is from ^{239}Pu. Therefore, a decontamination factor of about 3×10^3 must be applied to compacted hulls with this level of contamination if the MACH is not to be exceeded. The MACH's for several nuclides are given in Table 5.8.

The MACH is readily converted to a maximum surface count rate by:

$$MSCR = 100 \frac{MACH\ (2.2 \times 10^6\ dpm/\mu Ci)}{\rho_c R} \approx 1.1 \times 10^7\ (MACH) \quad (5.12)$$

where,

MSCR = maximum surface count rate ($dpm/100\ cm^2$)

ρ_c = compacted hulls density = $0.7\ \rho_t$

R = given by eq (5.10)

The last column of the table gives the MSCR for the isotopes listed.

5.1.3.2 Activation of Large Pieces of Material

The sequence of events which limits the concentration of activation products in large pieces of material results from mechanisms similar to those discussed for hulls. Hence, the procedure used is to assume corrosion to powdery form, apply the MAC's from Chapter 4 to the powder, and

TABLE 5-8

MAXIMUM ALLOWABLE CONCENTRATION LIMIT FOR HULLS

Nuclide	Original MAC (Ci/m^3)	MACH (Ci/m^3)	MSCR ($dpm/10^2cm^2$)
^{49}V	1.1 +8	3.4 +5	3.7 +12
^{54}n	1.0 +6	3.1 +3	3.4 +10
^{55}Fe	3.3 +7	1.0 +5	1.1 +12
^{57}Co	3.0 +7	9.3 +4	1.0 +12
^{60}Co	5.5 +4	1.7 +2	1.9 +9
^{63}Ni	3.0 +5	9.3 +2	1.0 +10
^{90}Sr	(1.3)	(1.3)	4.4 +4
^{125}Sb	4.3 +5	1.3 +3	1.4 +10
^{135}Cs	(20)	(20)[a]	6.8 +5
^{152}Eu	6.7 +4	2.1 +2	2.3 +9
^{232}Th	1.2 -1	3.7 -4	4.1 +3
^{235}U	4.1	1.3 -2	1.4 +5
^{238}U	(12)	1.8 -1	4.1 +5
^{237}Np	1.8	5.6 -3	6.2 +4
^{238}Pu	2.5	7.8 -3	8.6 +4
^{239}Pu	1.0	3.1 -3	3.4 +4
^{240}Pu	1.0	3.1 -3	3.4 +4
^{241}Pu	5.0 +3	1.6 +1	1.8 +8
^{242}Pu	1.1	3.4 -3	3.7 +4
^{241}Am	3.6	1.1 -2	1.2 +5
^{243}Am	3.1	9.6 -3	1.1 +5
^{242}Cm	8.2 +3	2.5 +1	2.8 +8
^{244}Cm	2.4 +2	7.4 -1	8.1 +6

[a]Values in parentheses pertain to the maximum allowable concentration limit for hulls where activity is related to total hull volume.

relate the MAC for the large pieces of material to the previous MAC by correcting for material composition and density as before. The relationship between a maximum allowable concentration limit for the activation products and the MAC's from Chapter 4 is the same as for the volumetric activation in the hulls (eq 5.7). Table 5.9 contains the Maximum Allowable Concentration for these specific Materials (MACM) for the significant activation nuclides, based on corrosion of iron to Fe_2O_3.

These limits are compared with estimated nuclide concentrations in the various groups for the Elk River reactor in Table 5.10. In estimating the nuclide concentration in each group, the ratio of the nuclide activation to total activation from Table 5.4 is assumed constant for each group.

The last column of each group lists the ratio of the nuclide concentration to its MACM. The sum of these ratio's is considerably less than unity. As expected, the largest component is the reactor internals, approaching 1.3% of its MACM.

A similar comparison is made for the activation waste from the 3600 MWth reactor D&D. As shown in Table 5.11, the reactor internals also constitute less than 2% of the MACM.

5.1.3.3 Surface Contamination of Large Pieces of Equipment

In discussing the D&D of the Elk River reactor, transuranic element contamination was evidently not a serious consideration, yet it is probable that it would become a factor in most reactor decontamination operations. The only group of waste materials that would likely be contaminated with transuranic isotopes is Group II or the internal components.

On the other hand, nearly all of the equipment and material from fuel reprocessing and waste processing facilities will have surface contamination. Some of this contamination will be on interior surfaces. Contaminated metal oxides from interior surfaces however, could become airborne or waterborne from several types of actions on the equipment. Therefore, it will be assumed that interior surfaces will become accessible to the environmental pathways.

The concentration of contaminants in the oxide powder is given by the ratio of the surface concentration to the volume oxidized. The maximum allowable concentration from surface contaminants (MACS) is given by:

TABLE 5.9

MAXIMUM ALLOWABLE CONCENTRATIONS FOR ACTIVATION OF SPECIFIC MATERIALS

Nuclide	Original [a] MAC ($\mu Ci/cm^3$)	MACM [b] ($\mu Ci/cm^3$)
^{49}V	1.1 +8	6.9 +8
^{54}Mn	1.0 +6	6.2 +6
^{55}Fe	3.3 +7	2.1 +8
^{57}Co	3.0 +7	1.9 +8
^{60}Co	5.5 +4	3.4 +5
^{59}Ni	3.3 +5	2.4 +5
^{63}Ni	3.0 +5	1.9 +6
^{125}Sb	4.3 +5	2.7 +6
^{152}Eu	6.7 +4	4.2 +5

[a] From Chapter 4

[b] MACM = maximum allowable concentration of contaminant in special material

TABLE 5.10

NUCLIDE CONCENTRATION IN WASTE
GROUPS FOR ELK RIVER WASTE

Isotope	MACM[a] (Ci/m^3)	(R) Ratio [b]	Group I (RPV) (R)x Act. (Ci) [c]	Group I (RPV) Conc. (C) (Ci/m^3) [d]	Group I (RPV) FMAC [e]	Group II (Internals) (R)x Act. (Ci)	Group II (Internals) Conc. (C) (Ci/m^3)	Group II (Internals) FMAC	Group III (Externals) (R)x Act. (Ci)	Group III (Externals) Conc. (C) (Ci/m^3)	Group III (Externals) FMAC	Group IV (Biological Shield) (R)x Act. (Ci)	Group IV (Biological Shield) Conc. (C) (Ci/m^3)	Group IV (Biological Shield) FMAC
^{49}V	4.8 +8	2.3 -4	2.5 -1	5.4 -2	7.7 -5	2.0 +0	1.8 +0	2.5 -3	1.0 -1	1.9 -2	2.7 -5	1.3 -3	2.2 -4	3.2 -7
^{54}Mn	4.4 +6	3.4 -3	3.7 +0	8.0 -1	1.8 -7	3.0 +1	2.7 +1	6.1 -6	1.5 +0	2.8 -1	6.4 -8	2.0 -2	3.4 -3	7.7 -10
^{55}Fe	4.8 +5	5.7 -1	6.3 +2	1.4 +2	3.0 -4	5.0 +3	4.5 +3	9.3 -3	2.5 +2	4.7 +1	9.8 -5	3.3 +0	5.6 -1	1.2 -6
^{57}Co	1.3 +8	2.0 -4	2.2 -1	4.8 -2	3.6 -10	1.7 +0	1.5 +0	1.1 -8	8.8 -2	1.7 -2	1.3 -10	1.2 -3	2.0 -4	1.5 -12
^{60}Co	4.3 +6	4.0 -1	4.4 +2	9.6 +1	2.2 -5	3.5 +3	3.2 +3	7.5 -4	1.8 +2	3.4 +1	8.0 -6	2.3 +0	3.9 -1	9.1 -8
^{63}Ni	1.3 +6	3.0 -2	3.3 +1	7.2 +0	5.5 -6	2.6 +2	2.4 +2	1.8 -4	1.3 +1	2.5 +0	1.9 -6	1.7 -1	2.9 -2	2.2 -8
^{125}Sb	1.9 +6	2.2 -4	2.4 -1	5.2 -2	2.7 -8	1.9 +0	1.7 +0	9.1 -7	9.7 -2	1.8 -2	9.5 -9	1.3 -3	2.2 -4	1.2 -10
^{152}Eu	2.9 +5	2.1 -4	2.3 -1	5.0 -2	1.7 -7	1.8 +0	1.6 +0	5.5 -6	9.2 -2	1.7 -2	5.7 -8	1.2 -3	2.0 -4	6.8 -10
TOTAL					4.1 -4			1.3 -2			1.3 -4			1.6 -6

(a) From Table 8

(b) Ratio of curies of the nuclide to the total curies in the Elk River waste - i.e. for ^{57}Co (from Table 4)
2.0/9.8 + 3 = 2.0 -4

(c) Product of Ratio (b) and the total activity for the type of waste group (from Table 2) - i.e. for ^{57}Co
(2.0 -4) x (1.1 +3) = (2.2 -1)

(d) Quotient of the activity due to the isotope (c) and the volume of the waste group - i.e. for ^{57}Co
C = (2.2 -1)/(4.6 +0) = (4.8 -2)

(e) Concentration (d) divided by MACM(a)

TABLE 5.11

NUCLIDE CONCENTRATION IN WASTE GROUPS FOR 3600 MWth REACTOR

Isotope	MACM[a] (Ci/m^3)	(R) Ratio [b]	Group I (RPV) (R)x Act. (Ci) [c]	Group I (RPV) Conc. (C) (Ci/m^3) [d]	Group I (RPV) FMAC	Group II (Internals) (R)x Act. (Ci)	Group II (Internals) Conc. (C) (Ci/m^3)	Group II (Internals) FMAC	Group III (Externals) (R)x Act. (Ci)	Group III (Externals) Conc. (C) (Ci/m^3)	Group III (Externals) FMAC	Group IV (Biological Shield) (R)x Act. (Ci)	Group IV (Biological Shield) Conc. (C) (Ci/m^3)	Group IV (Biological Shield) FMAC
^{49}V	4.8 +8	1.1 -4	5.6 +0	4.3 -2	6.1 -5	4.4 +1	1.5 +0	2.1 -3	2.2 +0	1.6 +0	2.3 -3	3.0 -2	1.9 -4	2.7 -7
^{54}Mn	4.4 +6	1.5 -3	7.7 +1	5.9 -1	1.3 -7	6.0 +2	2.1 +1	4.8 -6	3.0 +1	2.1 -1	4.8 -8	4.1 -1	2.6 -3	5.9 -10
^{55}Fe	4.8 +5	4.2 -1	2.1 +4	1.6 +2	3.4 -4	1.7 +5	5.9 +3	1.2 -2	8.4 +3	6.0 +1	1.3 -4	1.1 +2	6.9 -1	1.4 -6
^{57}Co	1.3 +8	8.9 -5	4.5 +0	3.5 -2	2.7 -10	3.6 +1	1.2 +0	9.1 -9	1.8 +0	1.3 -2	9.8 -11	2.4 -2	1.5 -4	1.1 -12
^{60}Co	4.3 +6	4.5 -1	2.3 +4	1.8 +2	4.1 -5	1.8 +5	6.2 +3	1.4 -3	9.0 +3	6.4 +1	1.5 -5	1.2 +2	7.5 -1	1.8 -7
^{63}Ni	1.3 +6	1.3 -1	6.6 +3	5.1 +1	3.9 -5	5.2 +4	1.8 +3	1.4 -3	2.6 +3	1.9 +1	1.4 -5	3.5 +1	2.2 -1	1.7 -7
^{125}Sb	1.9 +6	1.6 -4	8.2 +0	6.3 -2	3.4 -8	6.4 +1	2.2 +0	1.2 -6	3.2 +0	2.3 -2	1.2 -8	4.3 -2	2.7 -4	1.4 -10
^{152}Eu	2.9 +5	4.8 -4	2.4 +1	1.8 -1	6.1 -7	1.9 +2	6.6 +0	2.3 -5	9.6 +0	6.9 -2	2.3 -7	1.3 -1	8.1 -4	2.7 -9
TOTAL					4.8 -4			1.7 -2			2.5 -3			2.0 -6

(a) through (e) See References of Table 9

$$MACS = CF \frac{W_o \rho_m}{W_m \rho_o} (MAC) \qquad (5.13)$$

where,

MACS = maximum allowable concentration from surface contamination concentration (Ci/m^3)

W_o = molecular weight of base metal corrosion product (g/mole)

W_m = molecular weight of base metal (g/mole)

ρ_o = density of corrosion product (g/cm^3)

ρ_m = density of base metal (g/cm^3)

CF = fraction corroded at the time of exposure event (g corroded/g material)

MAC = MAC from Chapter 4

The fraction of material corroded CF is given by:

$$CF = Cr\ T\ R \qquad (5.14)$$

where,

C_r = corrosion rate of surface ($g/cm^2/yr$)

T = time to exposure event (yr)

R = surface area to mass ratio for material (cm^2/g)

Unfortunately, neither corrosion rates nor surface area to mass ratios are well known for the miscellaneous mixtures of equipment and materials that fall into the waste type being investigated. Therefore, another approach for establishing a limit may be appropriate.

A limit on the amount of surface contamination on equipment and material can be specified based on activity per unit area, eliminating the requirement to know the surface area to mass ratio for the material. This maximum allowable surface contamination (MASC) is given by:

$$MASC = C_r \, T \left(\frac{W_o}{W_m \, \rho_o} \right) MAC \qquad (5.15)$$

where the parameters are as defined above.

Values of MASC for stainless steel and high carbon steel corrosion rates and for several nuclides are given in Table 5.12. The MASC is highly dependent upon the corrosion rate, with higher corrosion rates yielding larger MASC's.

The MASC can be converted to a maximum surface count rate MSCR by:

$$MSCR = 2.2 \times 10^8 \, (MASC) \qquad (5.16)$$

where,

MSCR = maximum surface count rate (dpm/100 cm^2)

MASC = maximum surface concentration ($\mu Ci/cm^2$)

For example, the MASC for ^{239}Pu on stainless steel is 1.1×10^{-4} $\mu Ci/cm^2$. This gives a MSCR of 2.4×10^4 dpm/100 cm^2.

An estimation of the amount of plutonium surface contamination that may be present on the reactor internal surfaces may be made from the decontamination tests conducted by Pacific Northwest Laboratory (PNL) reported by Dillon.[23] In these tests two fuel rods from different reactors were analyzed and found to contain surface Pu contamination of 63 and 4.4 $\mu Ci/cm^2$, about 15% of which is from ^{239}Pu. Therefore, a decontamination factor of about 9×10^4 is needed to bring the surface concentration below the MASC. (Note that this contamination is probably due to undissolved residual fuel. Lower values are expected for other components from a typical reactor.)

Several authors have proposed limits for surface contamination for various uses. The American National Standards Institute (ANSI) has set a limit of 100 dpm/100 cm^2 for Pu for objects to be released to the public. Dunster[34] for "inactive" areas of a plant, recommends a value of 2000 dpm/100 cm^2 averaged over 300 cm^2. Blatz and Eisenbud[35] concluded that a reasonable limit for radium in a radium plant was 10,000 dpm/100 cm^2.

TABLE 5.12

MAXIMUM ALLOWABLE SURFACE CONCENTRATIONS FOR LARGE EQUIPMENT

Nuclide	Original MAC ($\mu Ci/cm^3$)	MASC [a] for S.S. ($\mu Ci/cm^2$)	MASC [b] for H.C.S. ($\mu Ci/cm^2$)
^{49}V	1.1 +8	1.2 +4	3.5 +8
^{54}Mn	1.0 +6	1.1 +2	3.2 +6
^{55}Fe	1.1 +5	1.2 +1	3.5 +5
^{57}Co	3.0 +7	3.3 +3	9.6 +7
^{60}Co	9.8 +5	1.1 +2	3.1 +6
^{63}Ni	3.0 +5	3.3 +1	9.6 +5
^{90}Sr	5.6 +3	6.2 -1	1.8 +4
^{106}Ru	2.2 +5	2.4 +1	7.0 +5
^{106}Rh	3.3 +5	3.6 +1	1.0 +6
^{125}Sb	4.3 +5	4.7 +1	1.4 +6
^{134}Cs	4.4 +5	4.8 +1	1.4 +6
^{135}Cs	1.7 +3	1.9 -1	5.4 +3
^{137}Cs	5.5 +5	6.1 +1	1.8 +6
^{144}Ce	2.2 +5	2.4 +1	7.0 +5
^{144}Pr	1.1 +5	1.2 +1	3.5 +5
^{147}Pm	3.1 +6	3.4 +2	9.9 +6
^{152}Eu	6.7 +4	7.4 +0	2.1 +5
^{154}Eu	1.1 +5	1.2 +1	3.5 +5
^{232}Th	1.2 -1	1.3 -5	3.8 -1
^{235}U	5.2	5.7 -4	1.7 +1
^{238}U	5.0 +1	5.5 -3	1.6 +2
^{237}Np	1.7	1.9 -4	5.4
^{238}Pu	2.6	2.9 -4	8.3
^{239}Pu	1.0	1.1 -4	3.2
^{240}Pu	1.0	1.1 -4	3.2
^{241}Pu	1.1 +2	1.2 -2	3.5 +2
^{242}Pu	1.0	1.1 -4	3.2
^{241}Am	4.0	4.4 -4	1.3 +1
^{243}Am	3.4	3.7 -4	1.1 +1
^{242}Cm	5.0 +2	5.5 -2	1.6 +3
^{244}Cm	1.5 +2	1.7 -2	4.8 +2

[a] MASC * = 1.1×10^{-4} cm (MAC) (Ci/cm^2)

[b] MASC ** = 3.2 cm (MAC) (Ci/cm^2)

* Based on corrosion rate of stainless steel (10^{-6} cm/yr)
** Based on corrosion rate of high carbon steel (3×10^{-2} cm/yr)

5.1.3.4 Additional Future Exposure Events

One variation of the reclamation scenario which could result in contamination exposure to man is through wounds resulting from handling specific material waste. It is postulated that at some future date, an individual may uncover the waste and receive either a puncture wound or a cut from the contaminated metal. (This analogy applies to fuel element hulls and reactor components as well as to failed equipment.)

In modeling a postulated accident, the following assumptions were made:

1. The isotope of major concern is ^{239}Pu.

2. The loss of ^{239}Pu from the wound due to sloughing that occurs during the healing process was ignored.

3. All migration from the wound site is by way of the blood stream.

4. The following biological half-lives were used:

   (a) Short half-life (0.001 day) component fraction = 0.32

   (b) Intermediate half-life (49 day) component fraction = 0.42

   (c) Long half-life (4400 day) component fraction = 0.26

Transport of the ^{239}Pu from the wound site to bone was calculated using Lawrence Livermore Laboratory's AIRIN code.[36] This code provides a print-out of dose accumulation in the bone in rads. Assuming a non-occupational maximum dose equivalent (DE) of 500 mrem/yr to the body:

$$DE = QF \times DF \times DBN \qquad (5.17)$$

where,

DBN = dose accumulated in bone (mrad/yr)

QF = quality factor (10 for α)

DF = relative damage factor (5 for corpuscular α)

DE = dose equivalent (mrem/yr)

then the DBN corresponding to 560 mrem/yr is 10 mrad/yr. The AIRIN code calculates a dose to the bone of 100 mrad/yr from a 15 nCi injection in a wound. A limitation of 10 mrad/yr, therefore, would reduce that injection to 1.5 nCi.

If it is assumed that 10% of the surface contamination on a metal object is transferred to the wound site, and the surface exposed to the wound is 1 cm^2, then the upper limit of contamination on the metal would be 15 nCi/cm^2, which is equivalent to 3.3×10^6 dpm/100 cm^2.

This is several orders of magnitude higher than other surface contamination limits that have been proposed. In other words, the wound scenario does not appear to be the most restrictive pathway in the exposure of man to buried waste.

5.1.4 Conclusions Concerning Specific Material Limits

The MACH for hulls based on surface contamination corrosion will require that decontamination efforts be applied to hulls before they can be considered for disposal by burial. Decontamination factors of about 3×10^3 will be required, and have been reported as achievable.

For volumetric activation of large pieces of equipment and material, the MAC is less restrictive than the MAC from Chapter 4.

For surface contamination on equipment, the MSCR's are comparable to other limits proposed by various organizations. For typical contaminated material from a reactor, some decontamination efforts (up to factors of 10^5) may be required to allow disposal by shallow land burial. However, if surface to mass ratios are know, volumetric concentration limits may be less restrictive than the MSCR's.

5.2 LOW-SPECIFIC ACTIVITY MATERIALS

Another type of waste suitable for investigation with the waste classification methodology is low-specific activity waste. The examples examined in this chapter in detail are the large volumes of low-specific activity uranium mill tailings and coal ash. These types of radioactive waste do not contain the fission products and the transuranic

nuclides generally associated with waste from the nuclear power industry, but do contain members of the natural radioactive decay series of ^{238}U and ^{235}U. These series are given in Tables 5.13 and 5.14. The uranium mill tailings generally still contain about 85% of the radioactivity originally in the ore since thorium and radium are not removed during the uranium milling process. Coal ash contains the residual radioactivity from trace mounts of natural uranium ore in the coal. The release of these naturally occuring radioisotopes and their decay progeny makes mill tailings and coal ash piles potential sources of ionizing radiation in the environment.

In this chapter, a representative uranium mill tailings site is analyzed using the classification methodology. A few tailings disposal alternatives are identified and their associated impacts estimated in order to provide additional information for the waste classification data base. The expected ranges of activity in coal ash are compared to those in uranium mill tailings.

5.2.1 Characteristics of Uranium Mill Tailings

At the present time there are over 10^8 metric tons of uranium mill tailings in the United States. At the projected growth rate of the nuclear power industry there will be about 1.7 billion metric tons by the end of the century.[37]

Therefore, it is very important to provide for the appropriate disposal of this large amount of low-specific activity waste. The application of the waste disposal classification methodology can yield information pertenent to the appropriate disposition of this waste.

This section provides a brief review of the ore milling process and tailings characteristics.

5.2.1.1 Type of Ore

There are a number of commercially important uranium-bearing minerals, found mostly in the western United States. The two most common, Carnotite and Autunite, generally occur in relatively low concentrations in sandstone formations.

Carnotite is a vanadate of potassium and uranium containing small amounts of radium. It generally occurs as a powdery

TABLE 5.13

PRINCIPAL RADIOACTIVE DECAY PRODUCTS OF ^{238}U

Nuclide	Half-Life	Radiation	Atom Ratio ppb in Natural U
^{238}U	4.49×10^{9} yr	α	9.928×10^{8}
^{234}Th	24.10 d	β,γ	0.0146
^{234}Pa	1.175 min	β,γ	4.94×10^{-7}
^{234}U	2.48×10^{5} yr	$\alpha,(\gamma)$	5.48×10^{4}
^{230}Th	8.0×10^{4} yr	α,γ	1.77×10^{4}
^{226}Ra	1622 yr	α,γ	359
^{222}Rn	3.825	α	2.32×10^{-3}
^{218}Po	3.05 min	α	1.28×10^{-6}
^{214}Pb	26.8 min	β,γ	1.13×10^{-5}
^{214}Bi	19.7 min	β,γ	8.28×10^{-4}
^{214}Po	1.637×10^{-4} sec	α	1.15×10^{-8}
^{210}Pb	22 yr	β,γ	4.86
^{210}Bi	5.02 d	β	3.04×10^{-3}
^{210}Po	138.3 d	α	0.0838
^{206}Pb	Stable		

TABLE 5.14

^{235}U DECAY SERIES

Nuclide	Half-Life From ^{235}U	Radiation	Atom Ratio, ppb in Natural U
^{235}U	7.13×10^{8} yr	α,γ	7.15×10^{6}
^{231}Th	25.64 hr	β,γ	2.94×10^{-5}
^{231}Pa	3.43×10^{4} hr	α,γ	344
^{227}Ac	22.0 yr	$(\alpha),\beta,(\gamma)$	0.221
^{227}Th	18.6 d	α,γ	5.11×10^{-4}
^{223}Ra	11.2 d	α,γ	3.08×10^{-4}
^{219}Rn	3.92 sec	α,γ	1.25×10^{-9}
^{215}Po	1.83×10^{-3} sec	α	5.82×10^{-13}
^{211}Pb	36.1 min	β,γ	6.89×10^{-7}
^{211}Bi	2.16 min	α,γ	4.12×10^{-8}
^{207}Tl	4.79 min	$\beta,(\gamma)$	9.14×10^{-8}
^{207}Pb	Stable		

incrustation in loosely cohering masses or as an impregnation in sand or sandstone, and is commonly associated with malachite, azurite, biotite, and magnetite. Some of the largest commercial deposits are in San Miguel, Rio Blanco and Montrose Counties, Colorado; San Juan County, Utah; Maricopa County, Arizona; New Mexico; Wyoming and Mauch Chunk, Pennsylvania.[38]

Autunite is a hydrous phosphate of calcium and uranium. It occurs throughout the uranium belt in the United States generally as a secondary mineral along with Carnotite and other uranium minerals.[38]

5.2.1.2 Typical Milling Operation

In a typical acid leach-solvent extraction milling operation, the ore is introduced into the primary crushing circuit, where it is crushed to 1/2 in., screened, and the oversize material recycled to the crusher. The fine ore is elevated to storage bins which are vented through a dust collector. Air exhaust hoods are located on the crusher, at the screens, and at each transfer point. The air is passed through a dust collector before being discharged through a roof vent.

The ore is then wet ground in rod mills as a slurry containing 65% solids. The ore is ground to less than 28 mesh and discharged into the leach circuit, which consists of several tanks in series with a total residence time of about 5-10 hours. Sulfuric acid and an oxidant, sodium chlorate, are added continuously. The solution containing the dissolved uranium is separated from the solids by countercurrent washing in a countercurrent decantation circuit. The slurry passes through hydroclones to separate the coarse sand fraction and the sand is washed in a series of six classifiers. The overflow from the classifier joins the hydroclone overflow, and the slimes are washed in a series of six thickeners. Flocculants are added to promote settling. Finally the solids are washed with both fresh water and recycled raffinate from the solvent extraction circuit. The washed slimes and sands constitute the tailings and are pumped to the tailings pond.

Approximately 97% of the material processed in the mill is discharged to the tailings pond.

Recovery of the uranium from the leach liquor is accomplished by countercurrent contact in four extraction stages with a long-chain amine dissolved in kerosene. The uranium is stripped from the solvent with an aqueous solution of

ammonium sulfate, and the solvent is recycled back to the extraction circuit. The uranium is precipitated by addition of gaseous ammonia, concentrated, and partially washed in thickeners and collected in filters. A continous steam-heated dryer is used to dry the wasted precipitate. Dried uranium precipitate, commonly called yellow cake, is packaged in 55-gal steel drums for shipment to a refinery.

5.2.1.3 Characteristics of Mill Tailings

The total weight of waste solution accompanying the sands and slimes to the tailings pond is 150% of the ore processed. Although the slimes constitute only about 20% of the solid waste material they may contain up to 80% of the radioactive elements of major concern: ^{230}Th, ^{226}Ra and their daughters.

As the tailings slurry enters the tailings pond, the heavier sand fraction settles out and the fine slimes are carried by water to the lowest part of the pond where further settling occurs. In the post-operational phase of the tailings pond, the liquid evaporates, leaving a pile of the sands and slimes. The slimes are known to retain up to 40% moisture for several years after mill operation.

The concentration of radioactive materials in the tailings is about 10^3 pCi/cm^3 for ^{230}Th and ^{226}Ra. It is noted in Table 5.13 that ^{226}Ra decays to ^{222}Rn, an isotope of radon. Since radon, a noble gas, does not react chemically, it can readily diffuse through the tailings and be transported some distances away. In general, about 20% of the ^{222}Rn which is generated[39] escapes the sand and slime particles and is free to diffuse through the tailings. Analysis of the transport and impact of the radioactive gas provides an interesting application of the RWDCS methodology.

5.2.2 Environmental Pathways and Exposure Mechanisms

Releases of radioactive nuclides from a mill tailings pile could occur either through hydrological or atmospheric pathways, or man himself could come in intimate contact with the tailings pile (reclamation). These types of releases are evaluated in the following sections, based on a consistent set of exposure events and a reference typical mill tailings site.

The exposure events considered in this chapter are discussed in Chapter 4. They include inhalation and water use by a reclaimer, direct gamma exposure to uranium daughter

irradiations, atmospheric transport, groundwater transport and surface cover erosion. Figure 5.1 presents a schematic representation of these potential exposure pathways from the reference site. These pathways are analyzed following the description of the reference uranium mill tailings site.

5.2.2.1 Description of Reference Mill Tailings Site

For the purpose of applying the waste disposal classification methodology to uranium mill tailings, a Reference Mill Tailings Site (RMTS) has been defined. Parameters which describe this site are given in Table 5.15. These parameters are not specific for any one particular site, but have been assumed to be typical for future mill tailings piles and are appropriate for the RMTS. The pile dimensions given in Table 5.15 are based upon the operation of a 2,000 metric tons per day (MT/day) mill for a period of 35-40 years.

Use of the RMTS allows calculations of environmental conditions expected for a consistent set of potential exposure events to facilitate comparisons to dose guidelines and the results from Chapter 4. The analysis of the potential pathways for human exposure are based on the parameters at the RMTS, and result in postulated effects that should be representative of those that may be expected from typical actual sites. However, some of the pathways and associated potential impacts are inventory limited, and are site-specific, so specific analysis of any given site is required before concluding that the effects will be the same as those from the RMTS.

5.2.2.2 Mill Tailings Management at the RMTS

One of the more significant environmental concerns associated with uranium mill tailings requiring waste management consideration at the RMTS is the migration of radon gas from the tailings. The flux from deep tailings crossing the tailings-air interface is given by:[39]

$$J = R E \rho \sqrt{\lambda D_e} \qquad (5.18)$$

where

J = radon flux ($Ci/cm^2 sec$)

R = radium concentration in tailings (Ci/cm^3)

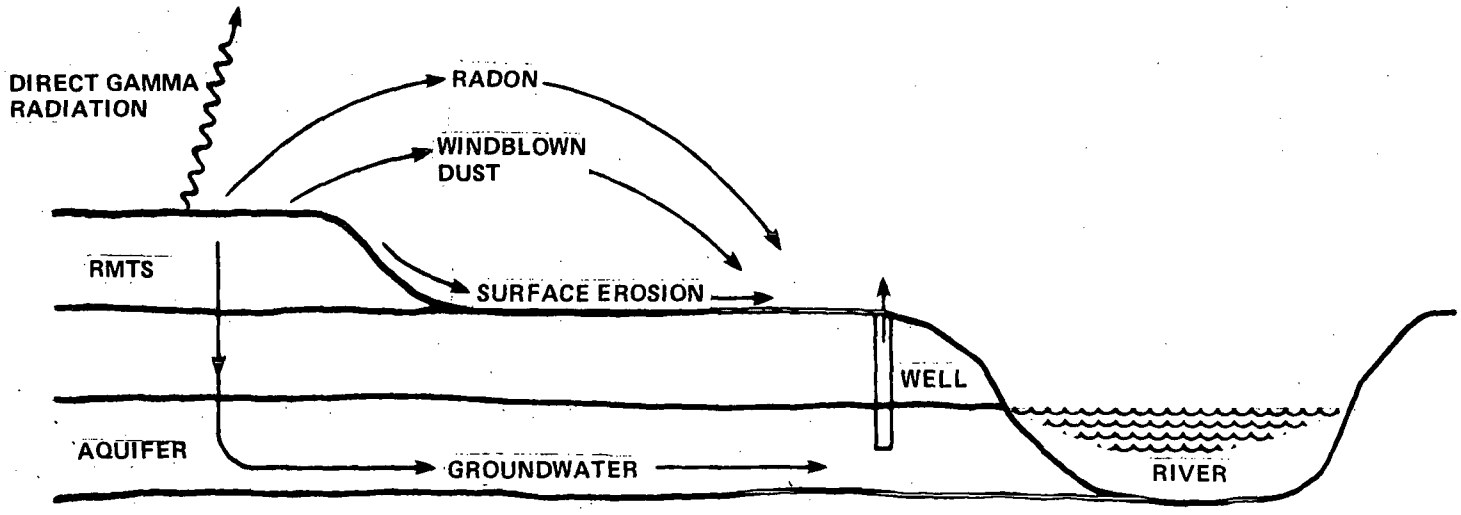


FIGURE 5.1 GENERALIZED POTENTIAL EXPOSURE PATHWAYS FROM RMTS

TABLE 5.15

PARAMETERS OF REFERENCE MILL TAILINGS SITE

Area	71 hectares (180 acres)
Pile height	15 m
Pile volume	1.8×10^7 m^3
Tailings density	1.7 g/cm^3
^{230}Th, ^{226}Ra concentration	10^3 pCi/cm^3
^{238}U concentration	50 pCi/cm^3
^{235}U concentration	2 pCi/cm^3
Vertical distance to aquifer	10 m
Aquifer distance to natural outlets (surface river)	1,000 m
Thickness of clay liner under tailings	0.5 m
Surface river flow rate	8.9×10^{11} l/yr
Water velocity - tailings to aquifer	10 m/yr
Aquifer velocity	100 m/yr
Downstream population at risk	5×10^5 people
Particulate deposition velocity	10^{-2} m/sec
Mean wind velocity	1 m/sec

E = emanating power of tailings (0.20)

λ = decay constant of radon (sec^{-1})

D_e = effective diffusion coefficient of radon through tailings (0.03 cm^2/sec)

ρ = density of tailings (17 g/cm^3)

For 10^3 pCi/cm^3 of radium in tailings, eq (5.18) gives a flux of 5×10^{-10} Ci/m^2 sec entering the atmosphere over dry tailings, or 3.6×10^{-4} Ci/sec from the entire tailings pile. However, after operations at the RMTS, sufficient cover is placed over the tailings to reduce the flux to twice background (2×10^{-12} Ci/m^2sec).

Atmospheric releases from the tailings vary for the operational phase and the post-operational phase. During the operation of the site, it is assumed that one tenth of the total pond area is exposed as a dry annular beach area where the tailings could be a particulated source. The inner radius is 417 m and the outer radius is 440 m.

During the post-operation phases, the radon source is a circular cylinder of a 440 m radius, which has been stabilized such that the radon flux from the pile is twice the background radon flux, where the background radon flux is assumed to be 2.0 $pCi/m^2/s$. Also during the post-operational phase, the tailings pile is assumed to be stabilized so that the dose from airborne particulates is negligible.

The dose conversion factor for ^{222}Rn includes the effects of radon daughters and assumes the bulk of the exposure is inside a structure.[40-41] The conversion factor also assumes a value for the equilibrium state between radon and its daughters, and is 1 pCi/ℓ of ^{222}Rn equals 1 rem/yr to the bronchial epithelium.

5.2.2.3 Inhalation of Radon by Reclaimer

During the post-operational phase of the RMTS, the most significant exposure events occur when the cover over the stabilized pile is removed. For instance, houses could be constructed on the site, disturbing the cover, exposing tailings and providing radon migration pathways.

Radon is very mobile in the environment, and can difuse relatively readily through many barriers, such as concrete. Recent experience has indicated radon levels in buildings built on tailings materials can approach a few hundreds of pCi/ℓ.

The potential concentrations of radon gas inside buildings constructed on mill tailings can be estimated from a mass balance using calculated values of the radon surface fluxes through walls. As a sample calculation, assume a 2.4 m x 9.1 m x 12.2 m (8' x 30' x 40') basement with 17 cm thick walls; about 100 m^2 of area is available for radon diffusion into the building. The basement volume is 270 m^3. The concentrations of radon gas in the structure can be estimated by the steady state continuity equation of:

$$C = 3.6 \frac{JA}{V\lambda} \qquad (5.19)$$

where

C = concentration of radon gas (pCi/ℓ)

J = radon surface flux through the wall (pCi/m^2-sec)

A = area across which flux passes (m^2)

λ = effective loss constant (decay plus removal) (hr^{-1})

V = volume into which flux enters (m^3)

The flux through the concrete walls x cm thick can be estimated by:[19]

$$J = J_o \exp(-\sqrt{\frac{\lambda}{D_e}}\ x) \qquad (5.20)$$

where Jo is the usual flux from eq (5.18) and the other parameters are as defined above. Using Jo = 500 pCi/m^2-sec and D_e through concrete = 6 x 10^{-5} cm^2/sec, gives a flux into the building of 21 pCi/m^2-sec.

The evaluation of C yields, for λ = 1 hr^{-1}, a radon concentration of 28 pCi/ℓ inside the structure. This is equivalent to a dose rate of about 30 rem/yr using the conversion factor of 1 rem/yr = 1 pCi/ℓ of ^{222}Rn and daughters.[40]

5.2.2.4 Well Water Reclamation Event

If the area of the RMTS is utilized at some time in the future, one possibility is that a well for culinary water may be drilled to the aquifer underlying the site. Contamination from the mill tailings may leach into the groundwater and be consumed via use of the well water.

To calculate this effect at the RMTS, the methodology for groundwater transport described in Chapter 4 was used with the parameters from Table 5.15. The concentrations in the well water are a maximum at the site boundry. Using the volume of water flowing under the site in the first 50 m of aquifer depth, and the annual rainfall water on the site as the volume into which leached nuclides will mix allows calculation of expected concentration in the well water at the time of peak release for each nuclide of importance in the tailings. Table 5.16 contains a summary of the releases and maxiumum individual doses to people who obtain all of their drinking water from the well.

It can be seen from the values given in Table 5.16 that ^{226}Ra is the most important nuclide in the well water, resulting in doses of 350 mrem/yr to individuals obtaining all of their drinking water from the well. Doses from other nuclides occur later in time and are orders of magnitude less.

5.2.2.5 Direct Gamma Exposure

Another related possible exposure event is the direct gamma exposure of individuals residing in a house built directly in tailings. The gamma flux through the concrete floors and walls is approximated by:

$$\phi_g = \frac{0.2R}{2N} E_2 (b_1) \qquad (5.21)$$

where,

ϕ_g = gamma flux (photons/cm^2 sec)

R = radium concentration (pCi/cm^3)

N = linear attenuation coefficient for tailings (0.11 cm^{-1})

$b_1 = \mu_c h$

μ_c = linear attenuation coefficient for concrete (0.16 cm^{-1})

h = thickness of walls and floors (17 cm)

E_2 = exponential integral

TABLE 5.16

MAXIMUM INDIVIDUAL DOSES FROM WELL WATER

Nuclide	Time of Peak Release (yr)	Peak Peak Release (pCi/yr)	Peak Peak Concentration (pCi/ℓ)	Maximum Individual Dose Rate (mrem/yr)
^{226}Ra	3000	1.8×10^{9}	16	350
^{230}Th	9×10^{4}	4.6×10^{8}	4.1	6.2
^{235}U	1.2×10^{5}	1.3×10^{7}	0.11	0.064
^{238}U	1.2×10^{5}	3.4×10^{8}	3.0	1.7

Equation (5.21) is an extension of eq (A-7) found in Section 2.3 of Appendix B, accounting for the shielding effects of the concrete wall.

The term 0.2R is the gamma emission rate[41] in units of photons/cm^3/sec.

Substitution of the appropriate values into eq (5.21) yields a gamma flux of:

$$g = 4.3 \text{ photons/cm}^2 \text{ sec}$$

The exposure rate associated with this gamma flux is obtained using the conversion factor:[42]

$$1 \text{ mrem} = 2.6 \times 10^6 \text{ photons/cm}^2 \qquad (5.22)$$

This yields an annual dose of 50 mrem/yr.

5.2.2.6 Atmospheric Releases of Radioactivity

The procedures used to calculate concentrations and dose rates from airborne radionuclides has been discussed in Chapter 4. The particulate source function was determined by correlation with actual measurements of airborne particulate concentration and windspeeds. It is given by:[40]

$$Q(u) = (3 \times 10^{-6}) \; R \; u^{2.6} \qquad (5.23)$$

Where,

Q(u) = particulate source term (pCi/m^2/sec)

R = nuclide concentration in tailings (pCi/cm^3)

u = average windspeed

The radon source function has been described in the literature[19] and is given by:

$$\phi = 10^4 \frac{RE\lambda f}{q} \tanh (qh) \qquad (5.24)$$

$$q = \sqrt{\frac{\lambda p}{D}}$$

where

ϕ = radon flux, (pCi/m^2sec)

D = bulk diffusion coefficient, (0.028 cm^2/sec)

E = emanating Power, (0.20)

λ = decay constant of ^{222}Ra, (2.1×10^{-6} sec^{-1})

p = porosity, (0.40)

h = thickness of source, (cm)

R = soil radium concentration, (pCi/cm^3)

f = soil moisture correction (moist, 0.69; dry, 1.0)

The meteorology used for this evaluation is an annual average wind velocity of 1.0 m/sec in the direction of the population center, 55% of the time. The balance of the time, the frequencies of wind directions are assumed to be distributed isotropically in each of the remaining 15 compass directions, with the constant annual average speed of 1.0 m/sec. Stability class E is assumed.

Population doses to the lung and bone from the airborne particulated concentrations and radon concentrations are presented in Tables 5.17 and 5.18 for the operation phase. Table 5.19 presents radon concentrations, individual dose rates and population dose rates during the post-operational phase.

The maximum individual dose rate during operation is from radon. It occurs at the site boundary and is about 1 rem/yr.

The largest population dose rate is also attributed to radon emissions.

5.2.2.7 Groundwater Migration

The calculated release rates of ^{238}U, ^{235}U, ^{230}Th and ^{226}Ra from the RMTS were determined in the manner described in Chapter 4. Table 5.20 contains the calculated releases, maximum individual and population dose rates using the dose conversion factors from reference 2.

TABLE 5.17

POPULATION DOSE RATES
DURING OPERATIONAL PHASE

Radius mi	Population Persons	Population Dose Rate to the Lung (manrem/yr) ^{222}Rn	^{226}Ra	^{230}Th	^{238}U
0-0.5	30	6.6	1.2×10^{-1}	6×10^{-1}	2.2×10^{-3}
1	30	1.8	2.3×10^{-1}	1.2×10^{-1}	4.3×10^{-4}
4	250	2.1	1.6×10^{-2}	8.3×10^{-2}	3.0×10^{-3}
5	150	0.9	6.2×10^{-2}	3.2×10^{-2}	1.1×10^{-4}
10	2,500	5.5	2.5×10^{-2}	1.3×10^{-1}	4.6×10^{-4}
15	1,500	1.8	6.9×10^{-3}	3.4×10^{-2}	1.2×10^{-4}
20	4,000	3.2	1×10^{-2}	5.3×10^{-2}	1.9×10^{-4}
30	8,000	4.2	1.1×10^{-2}	5.6×10^{-2}	2.0×10^{-4}
40	13,000	5.2	1.0×10^{-2}	5.7×10^{-2}	2.0×10^{-4}
50	25,600	8.2	1.5×10^{-2}	8.0×10^{-2}	2.9×10^{-4}
TOTAL	55,060	39.5	0.45	1.2	7.2×10^{-3}

TABLE 5.18

POPULATION DOSE RATES
DURING OPERATIONAL PHASE.

Radius (mi)	Population (persons)	Population Dose Rate to the Bone (manrem/yr) ^{226}Ra	^{230}Th	^{238}U
0.05	30	1.3×10^{-1}	2.2	4.5×10^{-4}
1	30	2.5×10^{-2}	0.45	8.7×10^{-5}
4	250	1.8×10^{-2}	0.31	6.3×10^{-5}
5	150	6.7×10^{-3}	0.12	2.4×10^{-5}
10	2,500	2.7×10^{-2}	0.46	9.6×10^{-5}
15	1,500	7.3×10^{-3}	0.13	2.6×10^{-5}
20	4,000	1.1×10^{-2}	0.20	4.0×10^{-5}
30	8,000	1.2×10^{-2}	0.21	4.2×10^{-5}
40	13,000	1.2×10^{-2}	0.21	4.3×10^{-5}
50	25,600	1.6×10^{-2}	0.30	5.9×10^{-5}
TOTAL	55,060	0.27	4.6	9.3×10^{-4}

TABLE 5.19

DOSE COMMITMENTS FROM RADON
DURING POST-OPERATIONAL PHASE

Radius (mi)	Population	Radon Conc. (pCi/ℓ)[a]	Individual Dose Rate (mrem/yr)	Population Dose Rate (manrem/yr)
0-0.5	30	5.2×10^{-2}	8.4×10^{-1}	2.5×10^{-2}
1	30	3.0×10^{-2}	4.8×10^{-1}	1.4×10^{-2}
4	250	4.2×10^{-3}	7.0×10^{-2}	1.8×10^{-2}
5	150	3.0×10^{-3}	5.0×10^{-2}	0.8×10^{-2}
10	2,500	1.1×10^{-3}	1.8×10^{-2}	4.5×10^{-2}
15	1,500	6.2×10^{-4}	1.0×10^{-2}	1.5×10^{-2}
20	4,000	4.2×10^{-4}	6.8×10^{-3}	2.7×10^{-2}
30	8,000	2.8×10^{-4}	4.6×10^{-3}	3.6×10^{-2}
40	13,000	2.0×10^{-4}	3.4×10^{-3}	4.4×10^{-2}
50	25,600	1.7×10^{-4}	2.8×10^{-3}	7.2×10^{-2}
Total	55,060	--	--	0.3 manrem/yr

[a]Radon flux assumes to be twice background or 4.0 pCi/m^2 sec.

TABLE 5.20

GROUNDWATER RELEASES AND DOSE RATES

Parameter	^{230}Th	^{226}Ra	^{238}U	^{235}U
Initial Inventory (Ci)	1.8 +4	1.8 +4	8.3 +2	38
Time of Max Release Rate (yr)	1.2 +5	5.0 +3	2.0 +5	2.0 +5
Dose Rate to Max Individuals (mrem/yr)	2	1.5 +2	5.0 -2	3.0 -3
Population Dose Rates (manrem/yr)	24	1.8 +3	--	--

As noted in the table, ^{226}Ra is the dominant nuclide for this pathway. The clay liner greatly inhibits the transport of radionuclides from the pile and therefore reduces the nuclide release rates over values generally used in Chapter 4. Maximum individual exposures will not occur until 5000 years after RMTS operations, and will amount to 150 mrem/yr from the radium. The releases from other nuclides will occur much later. Radon in the water from radium decay may be an important daughter effect deserving more investigation.

5.2.2.8 Surface Erosion

It is reasonably conservative to assume that future mill tailings piles will not be sited in areas where substantial erosion is likely to occur. However, if they were located in an area where water erosion were occurring, some contamination could ultimately be released to surface waters or dispersed into the atmosphere after erosion of the cover material. Design features can be incorporated which would tend to eliminate or at least minimize erosion processes.

For the sake of determining whether erosion may pose a substantial health risk, a simplistic yet conservative calculation based on a representative erosion rate was performed. There are a number of site specific parameters influencing erosion rates. Some of these are surface slope, amount of precipitation, distances to watercourses, distances from peaks, amount and type of vegetation and soil properties. However, six tons of soil per acre per year is a typical sheet erosion rate.[21] Using this rate, a total of 5×10^8 g/yr would erode towards the river. This gives a release rate to the river of 0.3 Ci/yr for ^{230}Th and ^{226}Ra. Assuming that 0.1% of the radionuclides become dissolved in the river water, the river water will yield a concentration of 3.4×10^{-4} pCi/ℓ for ^{230}Th and for ^{226}Ra. This concentration yields a maximum dose rate of 3×10^{-3} mrem/yr, clearly not limiting.

5.2.3 Cost-Benefit Considerations

Population dose rates total 40 manrem/yr from radon inhalation and 5 manrem/yr from particulates during the operational phase, 0.3 manrem/yr from radon in the post-operational phase and 24 manrem/yr from groundwater migration of thorium 1.2×10^5 year which will occur in the future. If 100 people are involved in the reclamation event, the population dose would equal about 4000 manrem/yr.

Based on a value of $1000/manrem, then, the operation of the RMTS could plausably spend up to $4 million/yr to

reduce the doses from the mill tailings. Possible methods for reducing doses include intermediate depth burial of the tailings and removal of radium and thorium from the ore during the milling operation.

5.2.4 Radioactivity in Coal Ash

A summary of available information[41,42,43] concerning the uranium content of coal and coal ash in the U.S. is presented in Table 5.21. In secular equilibrium, the ^{226}Ra and ^{238}U activities will be equal, as is expected to be the case in coal and coal ash. When coal is burned, uranium, thorium and radium form non-volatile compounds and remain in the coal ash. Disposal of the coal ash, then, deserves radioactive waste management consideration.

Using eq (5.18) to determine the radon surface fluxes expected from coal ash piles, assuming the effective diffusion coefficient for radon through coal ash is similar to that for mill tailings, gives ratios of fluxes from coal ash to those from mill tailings that range from 10^{-3} to 10^{-1}. Considering total areas of coal ash piles and mill tailings piles required to produce 1 GW(e)yr of electricity gives ratios of total fluences of radon gas from coal ash piles to those from mill tailings that range from 10^{-1} to 5. These calculations are summarized in Table 5.22.

5.2.5 Summary and Conclusions

As noted in Section 5.2.2, the critical nuclide and pathway for these materials is the radon in the reclamation scenario, although radon at the boundary during the operational phase is of concern and gamma exposures from reclamation could also be of concern. The radon daughter dose from the reclamation event clearly exceeds the dose guideline by about two orders of magnitude. Several other factors should be taken into consideration. For example, removal of most of the radium and thorium by an additional operation in the milling process, retaining some means of restricted use of the site in perpetuity or burying the coal ash or tailings sufficiently deep to preclude exposures to unsuspecting reclaimers (about 10 m). The last factor reduces the doses from all events to less than the dose guidelines. A summary of the potential impacts resulting from the incorporation of these factors is given in Table 5.23.

As observed in the table, the potential impacts from the base case of about 2 m of cover exceeds the guidelines for both the operational phase and the reclaimer. In both

TABLE 5.21

SUMMARY OF URANIUM CONTENT OF COAL AND COAL ASH IN THE U.S.

Region	Appalachia and Interior	Western
Coal Rank	Bituminous	Bituminous and Sabbituminous
Heating Value	12,000	8,500
Uranium Concentration (ppm)		
Range	0.2 - 43	0.2 - 23.8
Geometric Mean of Samples	1.2	0.8
Ash Content (wt %)	12	6.8
Coal Burned in 1000 MW(e) Power Plant (kg/hr)	2.38×10^9	3.38×10^9
Ash Collected and Stored (kg/yr)	2.83×10^8	2.27×10^8
Uranium in Stored Ash at Average U Concentration		
(kg/yr)	2.4×10^3	3.3×10^3
(^{238}U Ci/yr)	0.8	1.1
Uranium in Stored Ash at Maximum U Concentration		
(kg/yr)	1.0×10^5	8.0×10^4
(^{238}U Ci/yr)	34	26

TABLE 5.22

RADON FLUXES AND FLUENCES FROM URANIUM MILL TAILINGS AND COAL ASH PILES REQUIRED FOR 10000 MW(e) POWER PLANT

	Coal Ash		
	Appalachia & Interior	Western	Uranium Mill Tailings
Volume of Waste Pile (m^3)	2.0×10^5	1.6×10^5	1.5×10^4
Mass of Waste Pile (MT)	2.8×10^5	2.3×10^5	2.3×10^4
Area of Waste Pile (m^2)	2.8×10^4	2.3×10^4	9.1×10^2
Radon Flux (pCi/m^2-sec)			
Average U Concentration	4.2	7.5	820
Maximum U Concentration	183	176	---
Total Radon Release (Flux times Area) (pCi/sec)			
Average U Concentration	1.2×10^5	1.7×10^5	7.5×10^5
Maximum U Concentration	5.1×10^6	1.4×10^6	---
Ratio of Coal Ash Flux to Mill Tailings Flux			
Average U Concentration	0.005	0.009	---
Maximum U Concentration	0.22	0.21	---
Ratio of Coal Ash Radon Release to Mill Tailings Radon Release			
Average U Concentration	0.16	0.23	
Maximum U Concentration	6.8	5.3	

TABLE 5.23

IMPACTS FROM MOST RESTRICTIVE EVENTS

	Maximum Individual Doses (rem/yr)		
Land Use	Base Case-2m Burial	Base Case plus 90% Ra + Th removal	Intermediate Ground Burial (10m)
Restricted	1	0.1	0.1
Unrestricted	40[a]	4[a]	0.1

[a]Reclamation event limited. All others are operational airborne events.

cases radon is the critical nuclide. The impacts would be greatly reduced if 90% of the Ra and Th were removed in the milling operation, but the guidelines would still be exceeded in the reclamation event with unrestricted land use. Deep burial may not necessarily be the best available solution, however, the guidelines are not exceeded if the investigation of intermediate depth burial of mill tailings is suggested.

It has been assumed in the analysis of potential low-level waste using the RWDCS methodology that the site be in a condition acceptable for unrestricted use after 150 years. This assumption may not be applicable to mill tailings or coal ash piles. If it is not applicable, minor modifications to tailings management during mill or power plant operation could bring the base case impact below the dose guide lines established for this project.

The environmental impacts from uranium milling and tailings waste management is being examined in a generic environmental impact study. That study will also address the factors and alternatives identified in this section. Impacts from coal ash will also receive additional future study.

5.3 RADIOLOGICAL IMPACTS FOR MAXEY FLATS NORMALIZED TO A REFERENCE REACTOR YEAR

In applying the RWDCS methodology, a reasonable, consistent set of events are analyzed to estimate a range of probable impacts and to provide a data base for establishing interface values of a radioactive waste disposal classification system. In order to further evaluate the appropriateness of the set of events and the probable impacts that result from the analysis the use of parameters representative of a specific commercial burial site was examined.

5.3.1 Introduction

Parameters of the Maxey Flats, Kentucky, site were selected for the analysis because there are extensive and useful monitoring data available for this site. It has a large inventory of waste with much of the radioisotopic composition known and it is one of the longest operating commercial burial grounds that contains many features of the generic model RCF reported in Chapter 4.

In order to facilitate a comparison with the generic RCF, the site inventory was measured and calculated releases have been normalized to a RRY of waste. It is recognized

that a large part of the waste buried in the Maxey Flats Site was not from power reactors, yet, the bulk of waste generated in the future will be from the power reactor cycle. This normalization facilitates comparisons with other analysis of fuel cycle waste disposal facility. Because Maxey Flats is an existing facility, the possible exposure events are modified to fit the specific circumstances. Groundwater, surface water and air pathways have been considered in the analysis. The results are expressed in terms of curies released per RRY, dose to maximum individuals per RRY and man rem per RRY.

5.3.2 Facility Description and History of Operations

Maxey Flats is located near Morehead, Kentucky, not far from Lexington. The following sections provide details concerning the area and waste management operations there.

5.3.2.1 General Description of the Area

The 1.3 km^2 site is situated on Maxey Flats, a flat-topped, highly-dissected ridge in the Knobs region of Kentucky. The ridge rises approximately 100 m above the wide, flat alluvial-filled valleys. The upland surface is gently rolling and is generally less than 600 m wide. The Maxey Flats region has a humid continental climate with sharp contrasts between winter and summer temperatures. The mean annual precipitation ranges from 100-120 cm and averages about 110 cm.[44] Much of the region is cleared for agricultural use. Only the upland surfaces and slopes are heavily forested.

Maxey Flats is located on the eastern flank of the Cincinnati arch and is directly underlain by shales, siltstones and sandstones that gently dip to the southeast (4.7 m/km). A generalized geologic cross-section of the rocks that outcrop at the burial site is presented in Figure 5.2.[45]

At Maxey Flats only the lower 12 m of the Nancy Member of the Borden Formation are present. The trenches are entirely within the Nancy Member. It is a poorly fissile, plastic when wet, green shale, with siltstone and sandstone interbeds. The Farmers Member, the lower unit of the Borden Formation, directly underlies the Nancy Member. It is a ledge-forming, well sorted and well indurated, very fine grained, evenly bedded, quartzose sandstone with shale interbeds less than 1 m thick. This highly competent formation has well developed jointing and fracturing. The Henley Bed, a 1.5 m thick, greenish-gray clayey shale

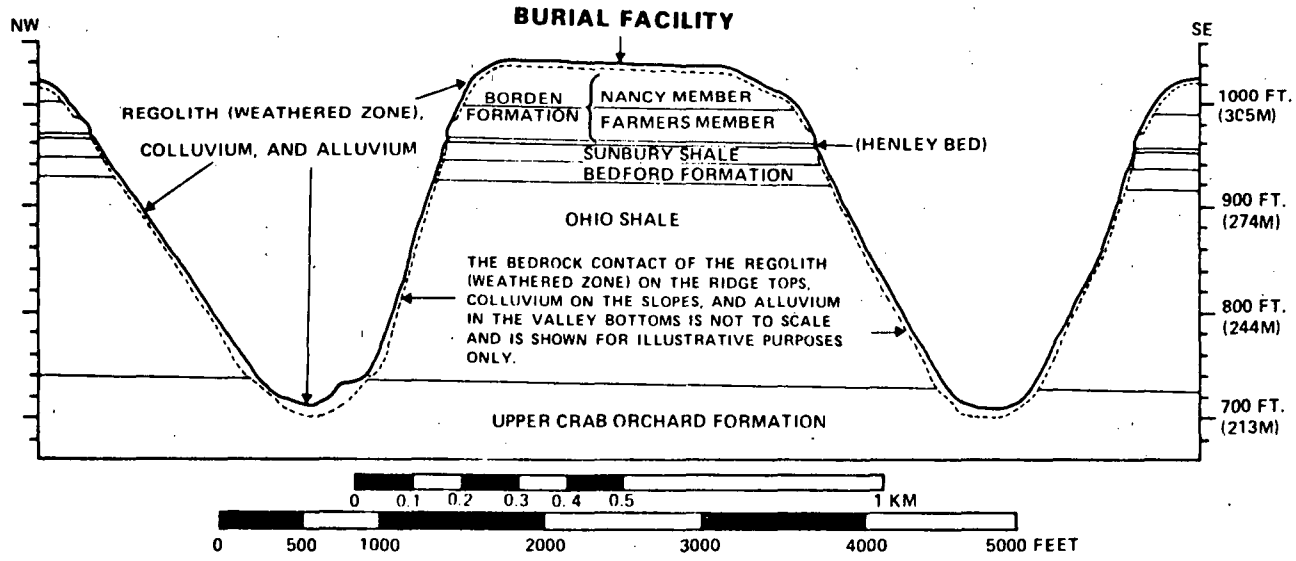


FIGURE TAKEN FROM REFERENCE 54

FIGURE 5.2 GENERALIZED GEOLOGICAL CROSS–SECTION OF THE MAXEY FLATS REGION.

with 2-5 cm thick sandstone and siltstone interbeds, lies at the base of the Farmers Member.

Papadopulos and Winograd[46] have recently described the occurrence of groundwater at Maxey Flats. The uppermost water table is perched in the soil zone above the poorly permeable Nancy Member and slopes southeastward paralleling the regional dip. This water table supplies the shallow dug wells on Maxey Flats. Hopkins[47] reported that deeper wells elsewhere on Maxey Flats have water levels as much as 9 m below the perched water in the soil zone. Other perched water tables may exist within the sandstones of the Farmers Member and conceivably in the Sunbury Shale above the reportedly expansive clay shales of the Bedford Formation.[48]

The main water table occurs in the Ohio Shale, a black fissile jointed shale.[48] This unit supplies several domestic wells in the Maxey Flats region. The Upper Crab Orchard Formation, a plastic shale which underlies the Ohio Shale may be thought of as the hydrologic basement for the Maxey Flats site. Papadopulos and Winograd[46] point out that the near-perennial nature of the creeks in the hollows surrounding Maxey Flats indicates that the creek discharge is primarily base flow from the Ohio Shale and/or the Upper Crab Orchard Formation and possibly from the colluvium blanketing part of the slopes around the site.

The formations at Maxey Flats are, in general, aquitards through which the intergranular movement of groundwater is very slow. However hydraulic tests of wells indicate that water can move along the joints and bedding planes. Walker[49] reported that the loss of water during a pressure test of a well* indicated that there are individual channels (joints) large enough and continuous enough to constitute a potential path for the migration of contaminants from the trenches. EMCON Associates, Inc. (EMCON), a geological consulting firm,[50] reported a 100 per cent loss of drilling water a number of times in the Nancy, Farmers, Sunbury and Ohio Shale during drilling and coring operations. More recently, Zehner[48] reported conducting withdrawal hydraulic tests which indicate that water movement occurs along joints or bedding planes.

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*Assuming that no leakage occurred around the packers, the results of Walker's tests indicate that injection rates range from 0.00006-0.015 $(\ell/s)/m/(kgf/cm^3)$.

5.3.2.2 History of Operations

Maxey Flats was first authorized as a commercial radioactive waste disposal facility in January 1963 and the first material was buried in May 1963. Packaged radioactive wastes were buried as received at the site in trenches about 40 ft wide, 25 ft deep and 300 ft long. A trench was opened to its full length and backfilled with soil as it was filled.

Between 1963 and 1974 approximately 104,000 m^3 of solid low-level radioactive waste was buried there. During the late 1960's and early 1970's rainwater began to infiltrate and accumulate in completed trenches, and the site operator initiated a program to pump the leachates from trenches to above-ground storage tanks and to an evaporator which concentrated the liquids to solids. This system was also used to solidify liquid radioactive wastes which were shipped to the site.

In 1972 the site operator disposed of liquid radioactive waste directly to disposal trenches. Late in 1972, environmental monitoring detected elevated levels of radioactivity in the Maxey Flats environs. The results of additional monitoring indicate that the disposal area appears to be the source of the elevated concentrations.

5.3.2.3 Meteorology

As stated previously, the annual precipitation averages about 110 cm. The wind is predominately from a southerly direction at a speed of about 5 m/sec. More detailed wind information from the Lexington, Kentucky, Weather Bureau is given in Table 5.24.

5.3.2.4 Radioactive Nuclide Inventory

The first radioactive material was received at Maxey Flats in May 1963. During the early years of the burial ground operation, the quantity of waste buried and recorded from each shipment was often only a rough estimate of volume and activity. Generally, no attempt was made to specify the isoptopic content of the inventory. Early preliminary estimates were based upon an assumed burial rate of 750 curies per year. It was soon obvious that this was a gross underestimation, for in 1963, 2,206 m^3 of waste containing 22,556 curies of "by-product material" were buried at this site.

TABLE 5.24

AVERAGE WIND SPEED AND
DIRECTION FREQUENCIES FOR
LEXINGTON, KENTUCKY (1956-1960)

Direction	Frequency (%)	Ave. Windspeed (m/sec)
N	4.7	4.3
NNE	5.7	4.4
NE	6.5	4.0
ENE	4.6	3.9
E	3.2	3.7
ESE	3.2	3.4
SE	4.8	3.5
SSE	7.8	4.4
S	13.5	4.8
SSW	10.4	5.4
SW	7.9	5.5
WSW	6.2	5.6
W	4.8	5.4
WNW	5.4	5.5
NW	3.8	4.8
NNW	3.5	4.3
CALM	4.0	-

In 1972 a project was undertaken to reconstruct the isotopic distibution of the various pits, and to obtain an inventory of the site.[51,52] The inventories of the major nuclides considered in this analysis are given in Table 5.25. The total activity of waste deposited between 1963 and 1972 is estimated at over 2.6×10^6 Ci.

This inventory is normalized to a RRY of waste using the waste production rates from reference 53. It is assumed that ^{239}Pu is representative of all the transuranic, and that the inventory of ^{239}Pu was selected to obtain a total concentration of 10 nCi/g of waste. The second column of the table gives the RRY inventory, and the last column is the ratio of the Maxey Flats inventory to the RRY inventory.

5.3.3 Environmental Pathways and Exposure Mechanisms

Direct inhalation of dust from buried waste by a reclaimer, direct gamma exposures and use of water from a well at the site boundary are considered to be significant potential exposure pathways. Other airborne and waterborne transport mechanisms, usually quite site-specific, are also described for the Maxey Flats disposal facility.

Two atmospheric pathways have been considered in evaluating the magnitude of atmospheric releases from Maxey Flats. The first is a low-level continuous release which results from normal operational activity. It is assumed that as a consequence of normal operations, there will be some small releases of radioactive material to the atmosphere. These releases are then transported off-site by atmospheric dispersion.

The second source of airborne releases is the evaporator installed at Maxey Flats in 1973.

Almost all water discharging from the site and from the sub-surface formations beneath it originates from precipitation falling on the site. Average precipitation provides about 1.2×10^9 ℓ/yr of water on the site, with about 6.2×10^8 ℓ/yr discharged by surface runoff, 3.5×10^7 ℓ/yr entering the groundwater from the trenches and the rest discharged via evapotranspiration. These flow paths form potential water pathways for the migration of contamination from the site. In addition, routine operational releases and accidental spills during disposal efforts could provide additional mechanisms for contamination spread. The evaporator plume may constitute a fourth

pathway for contamination. Figure 5.3 represents a schematic of the major pathways for water passing through the system.

Water entering the ground as recharge moves downward and laterally through the hydrogeologic system with its movement complicated by differences in permeablility between strata, the dip of the strata and extensive jointing. Zones of lower permeability within formations and formations with lower permeability tend to retard the downward movement of water, which causes some of the water to move laterally, and in some cases, causes perched aquifers.[46] The gentle dip of the strata to the east-southeast may cause much of the perched and main water tables to slope and discharge to the southeast. However, fracturing makes it very difficult to predict the direction and rate of subsurface flow. It also greatly increases soil permeability and could cause the channeling and movement of groundwater at unexpectedly high velocities.

A less obvious but equally serious effect of fracturing is the reduction in the ion exchange reaction between the rock media and contaminants.

The dip of the strata, plus grading and earthmoving activities also cause the land surface at Maxey Flats to slope southeasterly, channeling most surface runoff from the site into a main east drainageway. The layering and sloping tend to channel and direct the movement of water through the system with some predictability.

5.3.3.1 Inhalation of Dust by a Reclaimer

One of the consistent set of possible exposure events evaluated in the RWDCS methodology is the possibility of a reclaimer being exposed to airborne radioactivity from the buried wastes after institutional control is relinquished. For the Maxey Flats site, the parameters and methods used in evaluating this pathway in Chapter 4 are generally appropriate. Using concentrations in the waste normalized to 1 RRY of waste volume and activity from Table 5.25, the maximum individual rates/RRY listed in Table 5.26 are obtained. It should be noted that except for ^{90}SR, the actual concentrations in the waste are less than the normalized concentrations, and in no case do they exceed the MAC's from Chapter 4.

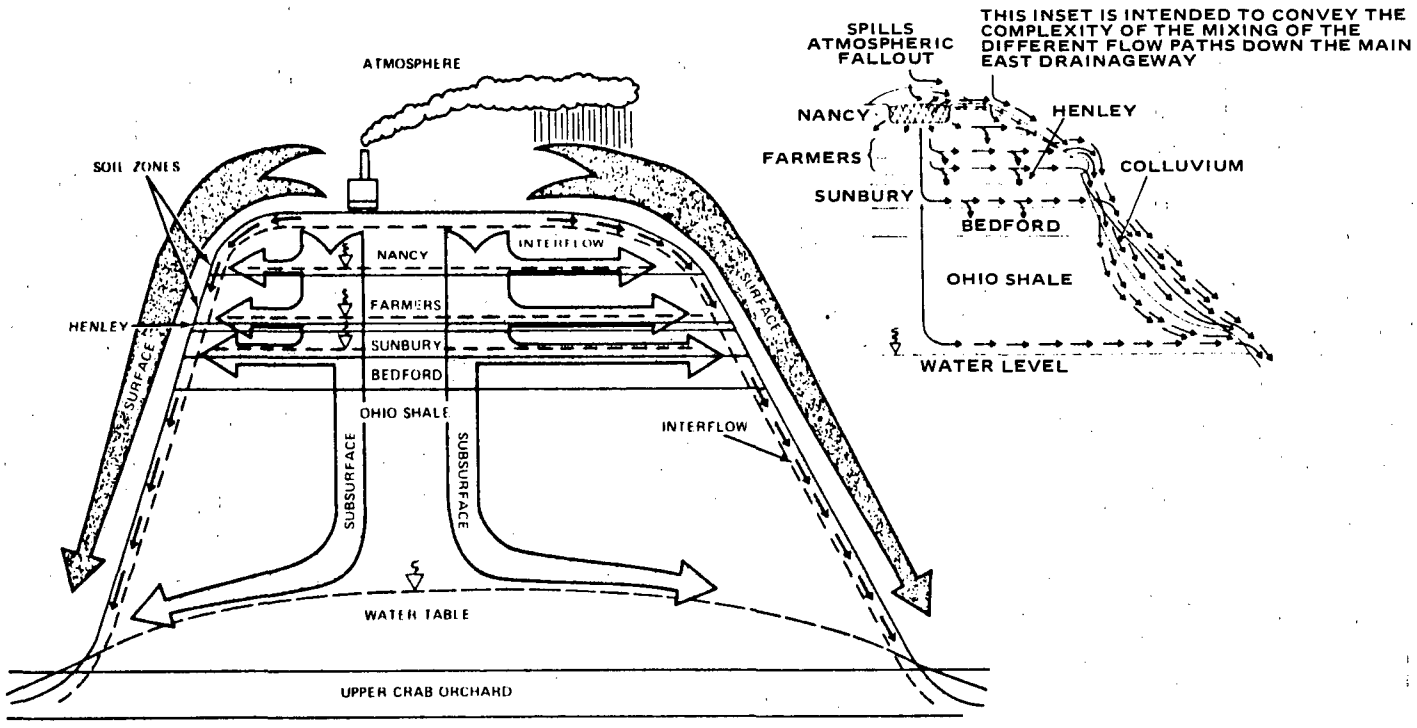

FIGURE TAKEN FROM REFERENCE 54

FIGURE 5.3 SCHEMATIC DIAGRAM OF WATER PATHWAYS AT MAXEY FLATS.

TABLE 5.25

NORMALIZATION OF MAXEY FLATS INVENTORY TO A RRY OF WASTE

Nuclide	Maxey Flats Initial Inventory (Ci)	RRY (Ci)	Rn
^{3}H	6.5×10^{5}	4.4×10^{3}	1.5×10^{2}
^{55}Fe	1.3×10^{2}	4.4×10^{2}	3.0×10^{-1}
^{60}Co	1.0×10^{5}	1.5×10^{3}	6.7×10^{1}
^{90}Sr	7.8×10^{3}	2.2	3.6×10^{3}
^{99}Tc	7	2.2×10^{-2}	3.2×10^{2}
^{129}I	1.1×10^{-2}	4.4×10^{-3}	2.5
^{137}Cs	2.4×10^{4}	1.1×10^{3}	2.2×10^{1}
^{239}Pu	1.2×10^{3}	12.6	9.5×10^{1}

TABLE 5.26

INHALATION DOSE RATES/RRY OF WASTE

Nuclide	Maximum Individual Dose Rate (mRem/yr/RRY)
^{3}H	0
^{55}Fe	0
^{60}Co	0
^{90}Sr	5.7×10^{-4}
^{99}Tc	2.8×10^{-9}
^{129}I	6.1×10^{-6}
^{137}Cs	1.6×10^{-3}
^{239}Pu	9.6

TABLE 5.27

MAXIMUM INDIVIDUAL DOSE RATES/RRY
OF WASTE FROM WELL WATER

Nuclide	Maximum Individual Dose Rate (mRem/yr/RRY)
^{3}H	2.3
^{55}Fe	0
^{60}Co	0
^{90}Sr	0.013
^{99}Tc	2.7×10^{-6}
^{129}I	3.6×10^{-4}
^{137}Cs	0
^{239}Pu	1.1×10^{-3}

5.3.3.2 Well Water Scenario

As noted in Chapter 4, the well water scenario is site inventory limited. The normalized Maxey Flats inventory consists of about 160 RRY of waste volume, and averages about 530 RRY of activity for the nuclides listed in Table 5.25. The effects of consumption of water from a well potentially drilled at the site boundary to the aquifer were investigated using the methodology from Chapter 4. Table 5.27 lists the dose rate/RRY from the well water event.

There are monitoring data available from shallow wells on the Maxey Flats site, but doubt exists as to the source of the activity found in the water. Some authorities are of the opinion that the concentrations of radioactivity in the wells come from surface contamination transported down the holes via precipitation runoff. Others think it could have resulted from pumping contaminated trash liquids onto the ground. Questions exist concerning whether liquid wastes have migrated in the groundwater, obviating the need for leaching as a transport mechanism. Comparisons with the well data are therefore not especially meaningful in this case.

5.3.3.3 Direct Gamma Exposure

Present environmental and operational health physics monitoring on the site assure that worker exposures are within established values. As shown in Chapter 4, a final soil cover of one meter is sufficient to reduce direct gamma doses to near background levels. Direct gamma exposure is significant only for ^{137}Cs for the nuclides in Table 5.25 after 100 years. Exposures from ^{137}Cs after 100 years would amount to 10 mrem/yr/RRY of waste.

5.3.3.4 Atmospheric Releases of Contamination

For operational releases from normal waste handling at the site, the rate at which material becomes available for dispersion in the atmosphere is assumed to be 10^{-7} of the amount of material processed annually. This factor is consistent with experience at the INEL and with other analyses.[19] Nuclide concentrations in air are calculated using the Gaussian Plume model and associated methodology described in Chapter 4.

The values of $_y$ and $_z$ used in the dispersion calculation have been determined for varying atmospheric stability

classes. The values of these parameters given in Reference 35 for conservative Pasquill Type E stability class are 30 m and 13 m respectively.

In the calculations, the sampling point is downwind of the release (y = 0) near the closest site boundary in that direction (x = 500 m). The average wind speed is 5.1 m/sec from the south, (24% of the time in a southerly direction), and the releases are from ground level (h = 0).

Particle deposition during migration from the point of release to the site boundary has been included in the calculations.

A value for Q' of 0.48 Q_o is used in the calculation. This is based on a 500 m source to boundary distance with a 5 cm/sec deposition velocity and wind speed of 5.1 m/sec.[55]

Using ^{239}Pu as an example of continuous airborne releases, it is assumed that the inventory was accumulated over a 10 yr period so that:

$$Q_o = \frac{I_o\ (10^{-7})}{3.15 \times 10^8} = 4 \times 10^{-3}\ \text{pCi/sec} \qquad (5.25)$$

where

Q_o = initial source strength (pCi/sec)

I_o = initial normalized inventory (12.6 Ci/RRY = 1.26×10^{13} pCi/RRY)

10^{-7} = release fraction

3.15×10^8 = seconds in 10 years of operation

Using this value, a site boundary concentration of 7.4×10^{-11} pCi/ℓ/RRY for ^{239}Pu is obtained from the calculation.

The results of the continuous airborne release calculations are shown in Table 5.28, where the centerline (y = 0) concentrations at the northern site boundary and resultant doses from continuous exposure are listed.

It is assumed in this analysis that the release from the evaporator is a function of the activity in the water evaporated. Dispersion in the atmosphere is calculated

TABLE 5.28

CONTINUOUS OPERATIONAL REEEASE
DOSE RATES/RAY OF WASTE

Nuclide	Annual Release Rate (pCi/sec/RRY)	Concentration At Site Boundary (pCi/ℓ/RRY)	Inhalation Dose to Maximum Individual (mrem/yr/RRY)
^{3}H	1.4	2.6×10^{-8}	3.3×10^{-8} WB
^{60}Co	0.48	8.9×10^{-9}	5.3×10^{-5} L
^{90}Sr	6.3×10^{-4}	1.3×10^{-11}	1.3×10^{-7} B
^{129}I	1.4×10^{-6}	2.6×10^{-14}	1.2×10^{-9} T
^{137}Cs	0.35	6.5×10^{-9}	4.0×10^{-6} LI
^{239}Pu	4.1×10^{-3}	7.4×10^{-11}	1.8×10^{-3} B

B = Bone
WB = Whole Body
L = Lung
GI = GI-LLI
T = Thyroid
LI = Liver

using the same methodology as described above for routine releases. The evaporator stack height is 10 m, the evaporation rate is 19 ℓ/min, and the evaporator is assumed to have operated 24 hours a day, 5 days a week for 3 years from 1974 to 1977 (a 71% duty cycle). The same meteorology conditions and deposition correction used for the continuous release case were also used to determine the concentrations from the evaporator releases.

The evaporator feed concentration (Cf) for ^{239}Pu is 4.5 x 10^3 pCi/ℓ from reference 59 divided by the RRY ratio for ^{239}Pu of 95, yielding 47 pCi/ℓ/RRY. The normalized release rate Q_o is obtained using eq (5.26):

$$Q_o = C_f \; E \; f_R \tag{5.26}$$

where

Q_o = the normalized release rate (pCi/sec/RRY)

C_f = the normalized evaporator feed water concentration (pCi/ℓ/RRY)

E = the evaporation rate (ℓ/sec)

f_R = the fraction of the nuclide in the feed water that escapes out the stack

The concentration at the site boundary is then calculated as in Chapter 4. The parameter f in eq (A-13) in Appendix B is defined as the product of the fraction of the time the wind is blowing in a given direction (24% from the south) with the fraction of the time the evaporator is operating (71% of the time for this case).

The feed concentrations, release fractions and calculated release rates are given in Table 5.29. Also shown in the table are measured release rates[52] normalized using the RRY ratio for each nuclide from Table 5.25, and calculated concentrations at the boundary per RRY based on the calculated release rates. The calculated boundary concentrations are higher than those calculated in the other report[52] because of the more conservative parameters used in the atmospheric dispersion modeling.

5.3.3.5 Groundwater Migration

In modeling the transport in groundwater of radionuclides

TABLE 5.29

DOSE RATES FROM EVAPORATOR PLUME RELEASES

Nuclide	Feed Concentration (pCi/ℓ)	Release Fraction	Calc. Release Rate (pCi/sec)	Meas. Release Rate (pCi/sec)	Conc. at Boundary (pCi/ℓ/RRY)	Dose (mrem/yr/RRY)
^{3}H	1.4×10^{7}	1.0	4.5×10^{6}	4.1×10^{6}	4.3×10^{-2}	5.5×10^{-2} WB
^{60}Co	3.4×10^{3}	10^{-2}	1.1×10^{1}	8.5	1.1×10^{-7}	6.3×10^{-4} L
^{90}Sr	1.3×10^{2}	10^{-2}	4.2×10^{-1}	9.0×10^{-2}	4.1×10^{-9}	4.0×10^{-4} B
^{137}Cs	2.0×10^{4}	10^{-2}	6.4×10^{1}	3.3×10^{2}	6.3×10^{-7}	5.5×10^{-4} LI
^{239}Pu	4.7×10^{1}	10^{-4}	1.5×10^{-3}	2.0×10^{-3}	1.4×10^{-11}	3.4×10^{-4} B

WB = whole body
L = lung
B = bone
LI = liver

at Maxey Flats, it is necessary to characterize the aquifer. Little information has been published describing the hydrogeologic characteristics of the site. Conservative values, however, are assigned to those parameters lacking site-specific data.

The aquifer was assumed to be homogeneous with a length of 800 m, roughly the distance from the site to Rock Lick Creek. The groundwater velocity was assumed to be 10 m/yr, or roughly 0.1 ft/day, and the dispersion coefficient was 10 m^2/day. Nuclide specific parameters such as decay constant, leach constant and sorption coefficient are those reported in Chapter 4. The sorption coefficient used for strontium and cesium are consistent with experimental values reported in reference 54.

Results of these calculations are presented in Table 5.30. Those nuclides listed in Table 5.24 that are not shown in Table 5.30 had negligible releases to the creek. The largest potential releases are for tritium after 85 years from the onset of release to the groundwater. Basically, off-site contamination in the groundwater is predicted not to occur for several years. The groundwater model, however, does not consider rapid movement of contaminated water through channels and fissures at rates greater than 10 m/yr.

5.3.3.6 Surface Water Transport

For Maxey Flats, it appears that the surface and near-surface runoff of precipitation is important. The dynamics of surface runoff and near surface interflow are extremely complex and highly site-specific. In order to estimate the possible magnitude of this effect, experimental measurements have been used to obtain dilution factors from surface runoff. Using information on sampling results from Tables 2-6 and 2-14 in reference 54, the ratio A_t of the radionuclide concentration in water in the waste pits to the concentration in the creek water ranges from 10^4 to 10^6. In the present analysis, a value of 10^5 has been used for this ratio to estimate the potential releases through the surface runoff pathway.

In Table 5.31, the concentration of nuclides in the trench water was calculated from eq (5.27) for t = 0 using the volumes of water (6.2×10^8 ℓ/yr) reported as surface water, and the leach constants from Chapter 4. These values are compared in the table to the experimentally determined concentrations reported in reference 54.

TABLE 5.30

DOSE RATES FROM GROUNDWATER TRANSPORT

Nuclide	Leach Constant (yr^{-1})	Peak Release Rate Into Surface Water (Ci/yr/RRY)	Time of Peak (yr)	Dose to Maximum Individual (mrem/yr/RRY)
^{3}H	10^{-1}	1.9×10^{0}	85	2.2×10^{-1}
^{99}Tc	10^{-4}	2.0×10^{-6}	95	1.3×10^{-6}
^{129}I	10^{-1}	2.1×10^{-4}	85	1.7×10^{-1}
^{59}Ni	10^{-4}	6.1×10^{-7}	2.5×10^{5}	6.5×10^{-7}
^{239}Pu	10^{-5}	6.2×10^{-15}	8.0×10^{5}	5.2×10^{-13}

TABLE 5.31

DOSE RATES FROM SURFACE WATER TRANSPORT

Nuclide	Normalized Trench Water Concentration (pCi/ℓ/RRY) Measured[a]	Calculated[b]	Activity Entering Creek[c] (Ci/yr/RRY)	Maximum Individual Ingestion Dose (mrem/yr/RRY)	
^{3}H	7.4×10^{5}	4×10^{6}	4.6×10^{-3}	5.3×10^{-5}	WB
^{60}Co	2.4×10^{4}	3×10^{3}	1.5×10^{-4}	6.6×10^{-4}	GI-LIT
^{90}Sr	3.7×10^{1}	1.4×10^{2}	2.3×10^{-7}	1.9×10^{-4}	B
^{99}Tc	---	3.4×10^{-3}	2.1×10^{-11}	1.4×10^{-11}	GI-LIT
^{129}I	---	7.2×10^{-1}	4.4×10^{-9}	3.5×10^{-6}	T
^{137}Cs	1.8×10^{3}	1.8×10^{4}	1.1×10^{-5}	1.3×10^{-4}	LI
^{239}Pu	2.1×10^{1}	4.2×10^{1}	1.3×10^{-9}	1.1×10^{-7}	B

[a]Measured values from Reference 36 divided by appropriate RRY ratio from Table 5.25.

[b]From eq (5.27)

[c]Based on calculated trench water concentrations

B = Bone
LI = Liver
GI-LIT = Gastrointestinal and Lower Intestinal Tracts
T = Thyroid
WB = Whole Body

$$C = \frac{\lambda_L I_o \exp(-\lambda_L t)}{V_w} \tag{5.27}$$

where

C = concentration of nuclide in leachate (Ci/ℓ)

λ_L = leach constant (yr^{-1})

I_o = normalized nuclide inventory (Ci/RRY)

V_w = average annual surface water runoff (6.2×10^8 ℓ)

The concentration of ^{239}Pu was calculated to be 21 pCi/ℓ/RRY by using a leach constant of 10^{-3} yr^{-1} in eq (5.27). The calculated concentrations are the same order of magnitude as the measured values, which are also included in the table.

In calculating the activity entering the creek from Maxey Flats, the number of curies of each nuclide released was determined using the normalized inventory. Knowing the volume of surface water runoff, the reduction factor A_t (10^5) and the trench water concentrations, the normalized activity released to the creek per year can be calculated using eq (5.28):

$$A = \frac{V_w C}{A_t} \tag{5.28}$$

where

A = activity release (Ci/yr/RRY)

V_w = average annual surface water runoff (ℓ/yr)

C = initial concentration of trench water (Ci/ℓ)

A_t = reduction factor (10^5)

These calculated activity releases per RRY are also given in Table 5.31. Maximum individual doses from consumption of all one's drinking water from the creek are also listed.

5.3.4 Population Dose Calculations

Population doses can be calculated based on the duration of the release, the affected downstream population and the exposure mechanisms.

5.3.4.1 Population Doses from Groundwater Migration

Another parameter of interest in considering the potential impacts from shallow land burial facilities is the infinite population dose per RRY from groundwater releases. The infinite population dose is a measure of the maximum potential environmental impact expected to be resultant from releases of a given magnitude as the radioactivity migrates from the source to the sea. The infinite population dose is calculated from the following expression:

$$D = \Sigma_i \frac{Pi}{Fi} U(DCF) R_s \qquad (5.29)$$

where

D = infinite population dose (manrem/RRY)

F_i = annual flow rate of surface water i (ℓ/yr)

P_i = downstream population using surface water i

U = water usage factor for standard man (ℓ/yr)

DCF = dose conversion factor (rem/Ci)

R_s = Total release to surface waters (Ci/RRY)

For the Maxey Flats case, population doses from the surface water pathway and the airborne pathways are a small fraction of the doses from the groundwater pathway. Only the groundwater pathway, however, is presented here. Table 5.32 contains the surface water flow and downstream population information needed in eq (5.29).

For groundwater releases the infinite population doses are presented in Table 5.33. The ^{129}I dominates the dose because of its long half-life and high dose conversion factor.

Because ^{129}I appears to be the most important dose contributor and yet has an extremely long half-life (16 million

TABLE 5.32

SURFACE WATERS AND

DOWNSTREAM POPULATIONS

Waterway	Flow Rate Fi (ℓ/yr)	Population, Pi
Rock Lick Creek	6.4×10^{9}	5,000
Licking River	3.2×10^{12}	300,000
Ohio River	1.0×10^{14}	2,400,000
Mississippi	1.6×10^{14}	3,000,000

TABLE 5.33

INFINITE POPULATION DOSES FOR GROUNDWATER RELEASES

Nuclide	Total Integrated Release (Ci/RRY)	D_∞ (manrem/RRY)
^{3}H	31	1.1
^{59}Ni	0.056	0.19
^{99}Tc	0.023	4.8×10^{-2}
^{129}I	0.004	10
^{239}Pu	7.7×10^{-10}	2.0×10^{-7}
	TOTAL	11 manrem

years), it deserves further consideration. It takes six kg of ^{129}I to make one curie. Therefore, as calculated by Rodger,[19] if the ratio of ^{129}I to ^{127}I in the thyroid is 2% or less, it is not possible to exceed the annual permissible thyroid dose. In any real situation, an individual's iodine consumption will be from many sources, making the contribution from a radioactive waste disposal site a small percentage of his bodily intake. It appears, therefore, that the thyroid dose calculated in Table 5.33 is probably not as important as it initially appears.

5.3.4.2. Population Doses From Surface Water Transport

It is also of interest to estimate population dose rates using the measured radionuclide concentrations in Rock Lick Creek as a starting point. The measured concentrations, given in column two of Table 5.34 are converted to equivalent annual releases per RRY using the flow rate of the creek and the ratios in Table 5.25. These normalized annual releases are then substituted for R in eq (5.29) and the population dose rates are calculated. The final results are also shown in the table. While the population dose rate listed for ^{239}Pu is low, it is still much larger than the infinite population dose given in Table 5.33. The concentration for ^{239}Pu used in the table was actually measured in the main East wash during a wet spring. A factor of ten was assumed for the dilution factor between water in the wash and the creek. In actuality, it is expected that the concentration reaching the creek would be smaller than a factor of 10, but this conservative value was used to estimate the potential magnitude of the doses.

5.3.5 Summary and Conclusions

Comparing the potential releases with measured values of radioactivity in the Maxey Flats environment verifies that the RWDCS methodology appears to adequately describe the migration pathways from this specific waste disposal site. Calculated doses are also in agreement with other estimates. A recent report[54] on the Maxey Flats site contains a summary of off-site dose calculations performed by the NRC, EPA and Dames & Moore. These are compared to the doses calculated using this methodology in Table 5.35.

Table 5.35 contains a tabulation of the dose rates per RRY for releases of the three most important nuclides. There are wide variations in values used for dose conversion

TABLE 5.34

POPULATION DOSE RATES

Nuclide	Creek Concentration[a] (pCi/ℓ)	Release (Ci/yr/RRY)	Population Dose Rate (manrem/yr/RRY)
^{3}H	2.9×10^{3}	0.13	4.7×10^{-3}
^{90}SR	5.2	9.4×10^{-6}	2.4×10^{-2}
^{137}CS	0.2	5.9×10^{-5}	2.2×10^{-3}
^{239}PU	0.007	4.7×10^{-7}	1.2×10^{-4}

[a] From Table 2-14, Reference 36.

TABLE 5.35

COMPARISON OF CALCULATED
DOSE RATES
(mrem/yr/RRY)

Aquatic Pathways

Nuclide	NRC	EPA	D&M	FB&DU
^{3}H	1×10^{-1}	7.2×10^{-4}	2.3×10^{-2}	2.2×10^{-1}
^{90}Sr	4.3×10^{-3} [a]	--	1.8×10^{-1} [b]	1.9×10^{-4}
^{239}Pu	-	--	8.2×10^{-2} [b]	1.1×10^{-7}

Airborne Pathways

Nuclide	NRC	EPA	D&M	FB&DU
^{3}H	3.4×10^{-2}	1.8×10^{-2}	1.7×10^{-2}	5.5×10^{-2}
^{90}Sr	4.3×10^{-4}	2×10^{-5}	1.7×10^{-3} [b]	4.0×10^{-4}
^{239}Pu	-	5.7×10^{-6}	1.2 [b]	3.4×10^{-4}

a. Total alpha and beta dose including natural activity

b. Upper bound calculation assuming all beta activity is from ^{90}Sr, and all alpha activity is from ^{239}Pu.

factors and pathways parameters which generally account for the differences reported. The NRC calculations are upper bound estimations only. In the Dames & Moore calculations, all beta activity was attributed to ^{90}Sr, all measured gross alpha activity was represented by ^{239}Pu only, and the maximum measured value was used for each nuclide. Differences in distances at which concentrations were calculated account for variations in the airborne pathways.

5.4 LATTY AVENUE SITE, HAZELWOOD, MISSOURI

This chapter describes the application of the RWDCS methodology to a contaminated site that was not initially intended for radioactive waste disposal (the Latty Avenue site in Hazelwood, Missouri). Included in this analysis is an evaluation of the potential for radioactive releases from the site and the potential doses to individuals living or working near the site. It is assumed that control of the site is maintained, and that access to the public is restricted. Doses from decontamination and decommissioning activities have not been assessed in this report.

5.4.1 Introduction

The site is located within the City of Hazlewood approximately 20 kilometers from St. Louis and adjacent to the Lambert, St. Louis airport. The site covers about 4.5 hectares and was used from 1966 to 1974 as a process drying facility for ore residues and uranium and radium-bearing wastes that were processed during the period 1942 through the late 1950's at another site in St. Louis.

During subsequent attempts at decontamination of the site, most of the ore residues and the mill machinery were removed from the site. Measurable levels of contamination, however, still remain in the topsoil and on some building surfaces.

Late in the summer of 1977, the Health and Safety Research Division of the Oak Ridge National Laboratory made a radiological survey of the site.[56] The results of this survey and the additional field work performed as part of the present effort, contained in Appendix D, showed that approximately 13,500 m^3 of contaminated soil weighing 2.64×10^7 kg remain at the site.

Analysis of this site was selected because of the availability of environmental measurements at the site, and to allow evaluation of the RWDCS methodology applied to a

contaminated site not originally intended for radioactive waste disposal. ORNL[56] and FBDU (Appendix D) field measurement data provide the bases for the parameters used in the methodology for calculating doses from wastes at the site. Details of the field measurements and investigations are included in the report of those activities included as Appendix D.

5.4.2 Description of Site and Calculation of Inventories

The following is a general description of the Latty Avenue site and a discussion of the calculations that were made to derive an inventory of radioactive nuclides present.

A more detailed description is contained in Appendix D.

5.4.2.1 General Description of Latty Avenue Site

The Latty Avenue site is a 4.5 hectare tract located within the city of Hazelwood in north St. Louis County, Missouri. The site is located at 9200 Latty Avenue, in low rolling hill terrain, at an altitude of approximately 170 m above sea level.

The eastern portion of the site (3 hectares) is vacant. The western 1.4 hectares portion has 4 structures, along with a railroad spur. The structures are vacant and are not being used for any purpose at present. Most of the site is weed-covered and there is some debris on the site. The site is only partly fenced.

The Latty Avenue site is located in the Florissant Basin, a shallow oval-shaped depression. The soils in the area of the site consist of a veneer of alluvium, a few feet of dark silt loam, up to 3 m of silty fine sands and clays (i.e. loess) of glacial origin underlain by 11 to 26 m of blue and gray clays, silts, and some sands of glacial lacustrine origin. The bedrock under the site is Mississippian age limestone, which is a unit in a sequence of almost horizontal Ordovician, Mississippian, and Pennsylvanian sedimentary formations.

From 1966 to 1974, the Latty Avenue site was used by the Continental Mining and Milling Company of Chicago as a process drying facility for ore residues and uranium and radium-bearing processed wastes previously generated at a different location.

About 85% of the total radioactivity originally in uranium ore remains in the processed wastes after removal of the uranium because the radium and thorium, principal contributors to radioactive emissions, are not normally removed from the uranium ores during milling. The principal environmental radiological impact and associated health effects arise from the ^{230}Th, ^{227}Ac, ^{226}Ra, ^{222}Rn and ^{222}Rn daughters contained in the waste materials. Other isotopes of uranium and thorium and their daughter products may also be present depending upon the type of ore present. Although these radionuclides occur in nature, their concentrations in the residue material are several orders of magnitude greater than their average concentrations in the earth's crust.

The ore residues have been removed from the site and transferred to other storage areas. Analyses of soil samples taken during an NRC investigation of the site in 1976, indicated the presence of uranium and thorium-bearing residues, and direct readings of radiation on the site exceeded the criteria established by the NRC for release of the site for unrestricted use. Late in the summer of 1977 the Health and Safety Research Division of the Oak Ridge National Laboratory made a radiological survey of the site.[56] Selected results from this survey are given in paragraph 5.4.2.2.

The Latty Avenue site is on the floodplain of Coldwater Creek, a tributary of the Missouri River. There is an abundance of surface water on or near the site, including ponded water during and after precipitation, a sewer system, septic tank(s), a waterline, several ditches, a storm drainage ditch and a tributary to Coldwater Creek. There is very little flow of off-site waters onto the site. Water does flow off-site to the north into a ditch and to the south into a stream, both of which drain into Coldwater Creek.

Ground waters in the area are contained largely in unconsolidated deposits and to a lesser extent in bedrock. The many silt and clay layers act to slow movement of the waters and provide ample opportunity for natural filtration and purification. Consequently, there is very little chance of contamination of ground waters from on-site existing contamination. Surface waters are, however, the primary source of water usage in the St. Louis area.

5.4.2.2 Calculations of Nuclide Inventory

The inventory of radionuclides that are present in the soil

at the Latty Avenue site can be estimated from surface and core samples reported in reference 56. The analyses and locations of the soil samples are reported in Appendix D.

An average of the surface soil samples was calculated for each of the four nuclides of concern. These averages are given in Table 5.36. It was assumed that these samples were representative of the soil between zero and 15 cm and that the average of the subsurface samples between 15 and 30 cm was representative of the next 15 cm of soil. These subsurface averages are also given in Table 5.36. These assumptions are conservative because the sample profiles indicate that the contamination levels decrease with increasing soil depth and the values used represent a high estimate for each segment of soil. The volume of soil contained in the top 30 cm of the entire site is equal to the volume of contaminated soil reported in Appendix D, Table 7-1; i.e. 13,700 m^3. The density of the soil was estimated to be 1.9 g/cm^3.

In general, the small areas where contamination extends below the 30 cm level have very low contamination concentration and would be insignificant compared to the total volume and activity.

The site inventory for each nuclide in Table 5.36 was calculated by multiplying the mean soil activity by the volume of soil in the top 0.3 m.

Sample No. S49 was not included in calculating the inventory. This sample was extremely high in ^{238}U and was obtained in an area that did not fit into the normal sample grid. It is a biased sample and if included would have an unjustified influence on the total inventory.

5.4.3 Environmental Pathways and Exposure Mechanisms

Since the Latty Avenue site is no longer being used for processing radioactive materials there are no routine radioactive releases from this facility. However, the following potential environmental pathways have been examined from the consistent set developed in the RWDCS methodology:

1. Inhalation by reclaimer
2. Use of well water
3. Direct gamma exposure
4. Atmospheric releases
5. Groundwater migration
6. Surface water transport.

TABLE 5.36

NUCLIDE AVERAGE CONCENTRATIONS AND INVENTORY

	^{226}Ra	^{238}U	^{227}Ac	^{232}Th
Surface Sample Ave. (pCi/g)	236	234	234	2.8
Subsurface Ave. (pCi/g)	97	150	59	1.3
Mean Activity (pCi/g)	167	192	147	2.1
Site Inventory (curies)	4.4	5.1	3.9	5.5×10^{-2}

5.4.3.1 Inhalation of Radon by Reclaimer

One of the consistent set of potential exposure events evaluated in the RWDCS methodology is the possibility of a reclaimer being exposed to airborne radioactivity from wastes on the site after institutional control is relinquished.

At the Latty Avenue site, particulate material is not generally a problem. However, radon gas does diffuse from the radium in the wastes. Maximum airborne concentrations of radon are estimated to be a few pCi/ℓ from measurements taken at the site. From measurements of indoor radon concentrations around the site, a value of from 5-20 pCi/ℓ of radon in the air appears to be a reasonably expected maximum. Using 1 pCi/ℓ = 1 rem/yr for full time exposure indicates that continuous occupation of the buildings where maximum radon readings occurred (Building 1, see Appendix D) may lead to doses up to 20 rem/yr. However, at present no one is occupying or expected to occupy the building. Ventilation can be provided to reduce these radon levels, if occupancy becomes desirable.

5.4.3.2 Well Water Scenario

At some time in the future, someone might drill a water well near the Latty Avenue site and use this water for food production and as drinking water. Current practice in the region is to rely on surface water for human needs because it is plentiful and less expensive than using wells for water. Although improbable, it is possible that a shallow well might be drilled and the water used by humans, therefore, this pathway was investigated. Parameters used in the modeling are listed on Table 5.37.

The maximum releases for ^{226}Ra are calculated to occur 3000 years in the future and peak maximum individual dose rates will then be 350 mrem/yr. For ^{230}Th, the maximum individual dose rates will occur 9×10^4 years in the future and will be 6 mrem/yr, which is well below the guideline limits.

It is concluded that release via the groundwater pathway is within the dose guidelines, even in the unlikely event that a water well is used for culinary water.

5.4.3.3 Direct Gamma Exposure

Measured values of direct gamma radiation exposure rates

TABLE 5.37

PARAMETERS USED IN GROUNDWATER TRANSPORT CALCULATIONS

Area of site	4.4×10^4 m^2
Activity of nuclides at site,	Given in Table 5.36
Velocity of water through soil to water table	1 m/yr
Distance to groundwater table	1 m
Distance from center of site to hypothetical well (approximately at the site boundary)	120 m
Velocity of water to the hypothetical well	100 m/yr

are greatest in Building 1 (see Appendix D) and range up to about 200 μrem/hr there. Therefore, an individual could spend over 2,500 hours in the building and as a result would receive a dose of 500 mrem/yr. Because there is presently no occupation of the site and public access is restricted, direct gamma exposures are not a significant problem at this time.

Background gamma radiation was determined at four locations within 6.4 kilometers of the site. The measurements ranged from 7 to 9 μR/hr. Cosmic rays are part of the measured background radiation levels. The contribution from cosmic rays is generally dependent upon the altitude and is approximately 5 μR/hr in the St. Louis area[57] or approximately 60% of the measured average background value. No significent contribution to background radiation readings were found from the Latty Avenue site.

5.4.3.4 Atmospheric Releases of Contaminated Dust and Radon

The basis of the airborne pathway calculations has been described in Chapter 4. Measurements at the site and physical observation of dense weed cover over most of the site indicate that wind blown particulates are not a significant problem at the Latty Avenue facility.

Radon gas, however, from the uranium decay chain is free to diffuse and become airborne. To calculate the radon source, the site was divided into several regions and characteristic ^{226}Ra soil concentrations. Using the total mass of each region, a mass weighted radium concentration was determined. The site was then modeled as a right circular cylinder conserving the volume of soil and area of the site. The cylinder radius was 120 m and the average ^{226}Ra soil concentration was 130 pCi/g.

The meteorology parameters used are from the Springfield, Missouri, Municipal airport. Stability classes D and E were assumed to dominate, each for 50% of the time. The average wind speed was found to be 2.2 m/sec and winds blow from the southeast, south or southwest 41% of the time.

The population distribution used in calculating population dose is given in Table 5.38, with the calculated radon concentrations and resultant doses as a function of distance from the site. Doses to both workers and residents have been tabulated. Figure 5.4 shows the radon concentration as a function of distance from the site.

TABLE 5.38

POPULATION, CONCENTRATIONS AND DOSES VS. DISTANCE FROM SITE

Distance (km)	Distance (mi)	Population Residents (persons)	Population Workers (persons)	Normalized Particulate Concentration (pCi/ℓ per pCi/gm)	Radon Concentration (pCi/ℓ)	Total Population Dose Rate to Bone Residents (manrem/yr)	Total Population Dose Rate to Bone Workers (manrem/yr)	Total Population Dose Rate to Lung Residents (manrem/yr)	Total Population Dose Rate to Lung Workers (manrem/yr)
0-0.16	0.1-0.1	0	800	2.8×10^{-11}	2.7×10^{-2}	0	5.0×10^{-2}	0	1.3×10^{-1}
0.16-0.32	0.1-0.2	0	2600	1.3×10^{-11}	1.3×10^{-2}	0	7.5×10^{-2}	0	2.2×10^{-1}
0.32-0.48	0.2-0.3	52	1600	7.1×10^{-12}	7.9×10^{-3}	3.3×10^{-3}	2.5×10^{-2}	8.3×10^{-3}	6.5×10^{-2}
0.48-0.64	0.3-0.4	22	2800	4.3×10^{-12}	5.1×10^{-3}	8.5×10^{-4}	2.6×10^{-2}	2.3×10^{-3}	7.4×10^{-2}
0.64-0.80	0.4-0.5	120	4400	2.7×10^{-12}	3.6×10^{-3}	2.9×10^{-3}	2.6×10^{-2}	9.3×10^{-3}	8.3×10^{-2}
0.80-1.21	0.5-0.75	466	9800	1.7×10^{-12}	1.9×10^{-3}	7.0×10^{-3}	3.8×10^{-2}	2.1×10^{-2}	1.1×10^{-1}
1.21-1.61	0.75-1.0	3900	25000	8.7×10^{-13}	1.2×10^{-3}	3.1×10^{-2}	4.8×10^{-2}	1.0×10^{-1}	1.7×10^{-1}
1.61-2.41	1.0-1.5	18700	5100	4.7×10^{-13}	6.2×10^{-4}	7.9×10^{-2}	5.3×10^{-3}	2.9×10^{-1}	2.0×10^{-2}
TOTAL		23260	52100	--	--	1.2×10^{-1}	2.9×10^{-1}	4.3×10^{-1}	8.3×10^{-1}

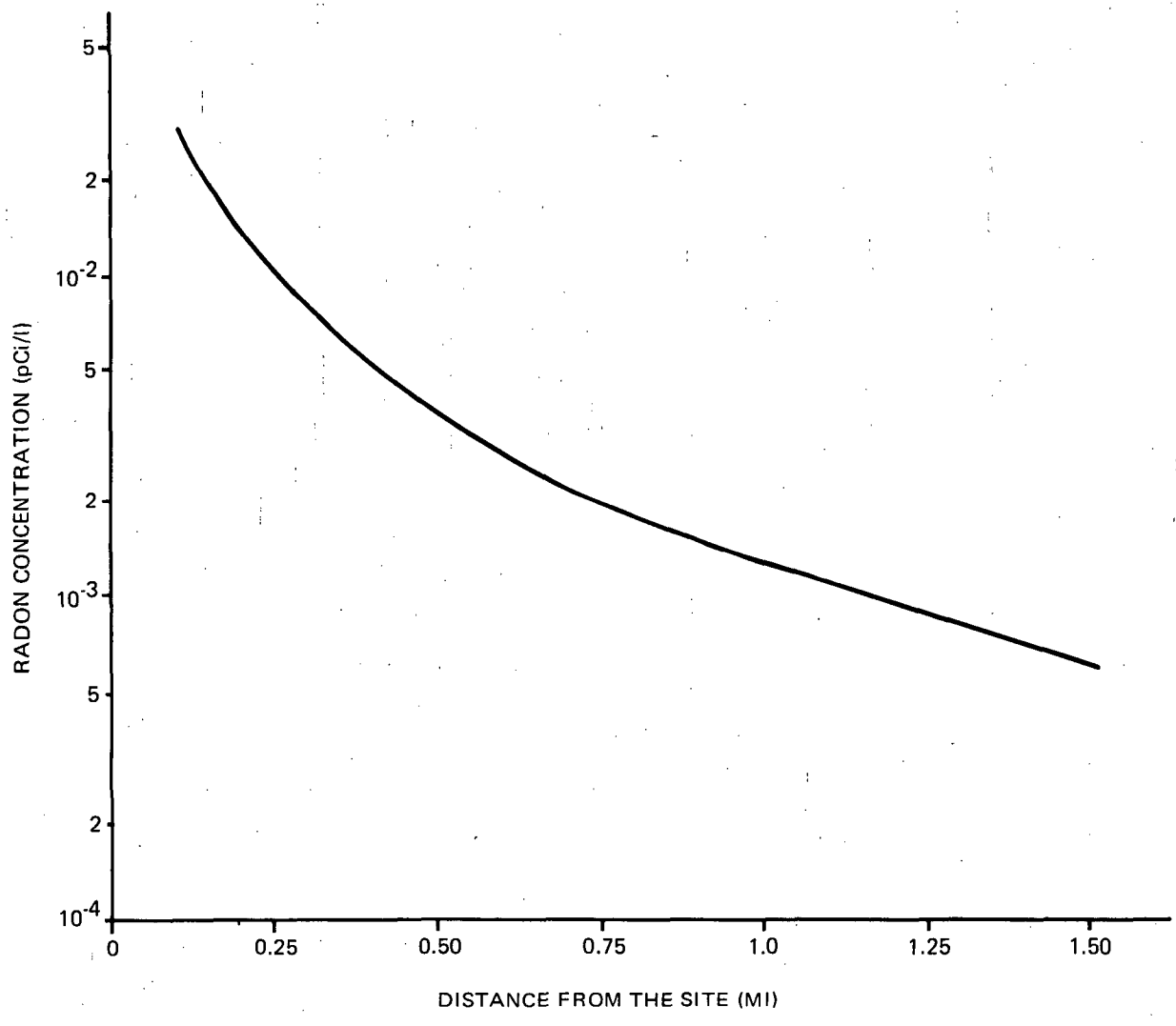

FIGURE 5.4 RADON CONCENTRATION AS A FUNCTION OF DISTANCE FROM THE SITE

It is observed from the table that workers close to the site boundary will receive the greatest doses (up to 0.2 mrem/yr). No residents live closer than 0.2 mi to the site. Doses to workers dominate because they are more abundant in the area. The population doses are distributed relatively evenly over the 1.5 mi radius area calculated. The total population dose from the site is 1.6 manrem/yr. These doses are small compared to natural background doses.

5.4.3.5 Groundwater Migration

As discussed in Chapter 5.4.3.2 for water wells, the potential for releases of radionuclides from Latty Avenue via the groundwater pathway is also very low.

The annual precipitation of Latty Avenue site is 89 cm per year. This water percolates into the soil or runs into the surface ponds and streams and only a small amount is suspected of entering the deeper groundwater zones. A complete discussion of the surface and groundwater is given in Chapter 2 of Appendix D.

In the well water scenario it was shown that the maximum individual dose from ^{226}Ra for a person drawing all of his water supply from a well located only 120 m from the source would amount to only 50 mrem/yr and that the peak would occur 3000 years in the future. Transportation of the nuclides via the groundwater to a seepage zone many kilometers away, where the aquifer could release the nuclides to the environment, would require several orders of magnitude longer and the amounts released could not be distinguished from background radiation. This analogy applies to the other radionuclides present at Latty Avenue. Therefore, groundwater migration does not appear to be a significant problem.

5.4.3.6 Surface Water Transport

The radionuclides measured in water and sediment samples at the Latty Avenue site are listed in Table 3-4 of Appendix D.

The first event considered is the inadvertent use of water from either of the three sources known to contain ^{210}Pb, ^{226}Ra and ^{230}Th.

For ^{210}Pb, using the dose conversion factors from reference 2, the dose to the bone would calculate to be 78 mrem/yr

if all of an individual's drinking water requirements (730 ℓ/yr) came from the ditch. The probability that one person would ingest such a large amount of this untreated water is very low, however. Water of this quality would not be considered potable and fit for human consumption, therefore, humans under normal circumstances would not be using it. The doses due to the other nuclides are 44 mrem/yr for ^{226}Ra, and 3 mrem/yr for ^{240}Th. Summing these doses for the maximum adult dose from the three nuclides gives 126 mrem/yr.

The dose received if sediment from the drainage ditch was ingested, (again, a highly improbably event) is readily obtained from the measured concentration of activity in the sediment. For ^{210}Pb, the dose would be 0.8 mrem/g; for ^{226}Ra, 0.1 mrem/g; and for ^{230}Th, 0.2 mrem/g. Therefore, several hundred grams of material would have to be consumed to result in 500 mrem/yr doses. Ingestion of that much sediment is very unlikely.

Water from the drainage ditch flows into Coldwater Creek which enters the Missouri River a few miles from the site. The dilution of the contaminants which results is extremely large and will lower concentrations of the nuclides to innocuous levels. Consumption of the water in the ditch would result in doses that are negligible in any case.

5.4.4 Population Dose and Cost-Benefit

The total population doses from airborne radon is 1.6 manrem/yr based on the calculations performed. Maximum individual doses will be obtained by workers who work at facilities adjacent to the Latty Avenue site, and will be about 0.22 mrem/yr. The maximum dose received by the nearest full-time resident will be 0.16 mrem/yr.

Inhalation of radon by full time occupants of Building 1 could result in dose rates of up to 20 rem/yr. If 10 individuals were involved, this would be an exposure of 200 manrem/yr. Direct gamma exposure would result in only 4 manrem/yr for 10 full time workers on the site if they spent full time in Building 1, where exposure rates are highest. Therefore, relatively little effort is justified to reduce the contamination levels and resultant dose rates on this site for the short-term. However, release of the site for unrestricted use and other long-term considerations could provide additional incentives for site cleanup.

5.4.5 Summary and Conclusions

Based on the dose guidelines for this study, the wastes now at the Latty Avenue site can be considered as low-level radioactive wastes in their present configuation. Table 5.39 summarizes the pathways analyses and resultant maximum individual and population doses. Although this site is not designed for long-term disposal of radioactive wastes, it appears that leaving the site as it is now would not present an unacceptable hazard to the public. If, however, use of the site is anticipated, then additional factors will need to be considered.

It should be noted that the concentrations of radioactivity in the soil at the Latty Avenue site are in the general range that may be considered below the nonradioactive limit from Chapter 4. However, the analysis upon which the hazard threshold concentrations were determined was based on implacement of wastes in a reference sanitary landfill and final covering with clean soil. It appears that the Latty Avenue site does, therefore, deserve consideration as containing low-level radioactive wastes.

5.5 SUMMARY OF APPLICATIONS IN CHAPTER 5

Chapter 5 of the report on the radioactive waste disposal classification system RWDCS has explored the application of the methodology to wastes other than routine low-level wastes. The methodology has been shown to be sufficiently consistent and versatile to allow calculation of expected effects over broad ranges of conditions and extensively varied waste forms and types.

Specific waste materials generated in the nuclear fuel cycle, such as hulls, large items activated in reactors, contaminated equipment and decommissioning wastes have been analyzed using the methodology. Limits on concentrations and surface activities have been derived. Generally, possibly with some decontamination efforts, most of these materials can be considered as suitable for disposal by burial.

Large volume, low-specific-activity materials, specifically uranium mill tailings, were found to possibly require deeper burial than the typical 1 m of soil cover now envisioned, or removal of radium and thorium during milling to assure that potential intruders do not receive unacceptable exposures. Maintaining institutional controls for longer periods (essentially in perpetuity because of the long half-lives involved) does not seem feasible. However,

TABLE 5.39

SUMMARY OF PATHWAYS AND DOSES

Pathway	Maximum Individual Dose Rate mrem/yr	Population Dose Rate manrem/yr
Consumption of Contaminated Surface Water	126	1.3[a]
Ingestion of 1 Gram of Sediment Per Year	1.1	0.011[a]
Off-Site Airborne Radon	0.22	1.2
Direct Gamma Exposure	400[b]	4[c]

[a] Based on 10 individuals exposed per year.

[b] Based on 2,000 hours of exposure per year.

[c] Based on 10 workers spending full time in Building 1.

siting criteria could possibly be selected such that future reuse of the site for human habitation is unlikely.

Application of the methodology to wastes now buried at Maxey Flats showed that predicted and measured environmental concentrations of radioactivity from the disposal site are not inconsistent. Even though specific information about actual migration pathways are not directly used in the methodology, the general pathways analyses appear to be representative of the types of mechanisms that actually allow transport of radioactivity through the environment. Investigation of the Latty Avenue site further verified this conclusion.

Some items were discovered or are suggested that may warrant further investigation and definition. These items include a more detailed analysis of deeper burial for certain waste types, clarification of the role of inventory limited pathways and their application to waste classification, identification of possible symergestic exposure and migration effects and further investigation of the role of corrosion rates and products in waste mobilization.

In conclusion, the RWDCS methodology, as developed, appears to provide a reasonably conservative basis upon which regulatory classification decisions can be based. More restrictive exposure events can certainly be be postulated, but the set presented adequately represents reasonably expected conditions for future waste disposal.

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APPENDIX A

THE ADVISORY PANEL MEETING ON WASTE CLASSIFICATION HELD IN TUCSON, ARIZONA, ON MARCH 9 AND 10, 1978

The sections in this Appendix contain the following discussions.

Section	Topic
A.1	Summary
A.2	List of Advisory Panel Members

APPENDIX A

THE ADVISORY PANEL MEETING ON WASTE CLASSIFICATION HELD IN TUCSON, ARIZONA, ON MARCH 9 and 10, 1978

A.1 SUMMARY

The panel convened with opening remarks by Dr. William P. Bishop, manager of the Nuclear Regulatory Commission's (NRC) waste management programs. Dr. Bishop thanked the panel members for their cooperation and willingness to lend their expertise in developing a waste classification system. In discussing the present reports, Dr. Bishop requested the panel to consider carefully the methodologies and concepts used rather than to dwell upon specific numeric details. He also added that the comments and suggestions of the panel would be considered and, as far as possible, incorporated into future work and revisions of the classification system.

Mr. John Adam of the NRC's waste management staff presented the background of the present study. He explained that FB&DU was given the charge to calculate the data base for the waste classification system as factually as possible without building in large degrees of conservatism. It will be the responsibility of the NRC to later determine the amount of conservatism that must be applied in order to set the final regulatory standards for radioactive waste disposal. The FB&DU study will form the data base upon which these regulatory standards will be established.

Mr. Adam explained that FB&DU was charged with the following obligations:

1. Devise a waste classification system for establishing the limits for disposing of waste via three principal methods:

   a. Landfill operation
   b. Land burial
   c. Geologic isolation

   (It was conceded that other waste classes might be developed based on the results of the analyses.)

2. Keep all calculations as reasonable as possible.

3. Make the data base information available as early as possible.

The basic assumptions used for the study were discussed by the panel after the introductory remarks. Considerable attention was directed towards the dose guidelines used, and specifically, questions of calculating population doses. The concept of truncating the calculations when individual exposures fall below some lower limit was discussed, but was not incorporated in the methodology, even though dose rates to the entire population were not estimated. Integrating over time to determine dose commitments was also discussed. Performing the double integral over time and populations would facilitate comparisons with other work. Fifty-year-dose commitments were suggested as appropriate for comparisons. Appropriate dose guidelines for future generations were discussed. Some panel members felt that future generation doses should be more restrictive and others felt that they should be discounted. This study applies the same guidelines to present and future generations.

In applying ALARA guidelines, $1000/manrem has been suggested from Appendix I of 10CFR50. The panel discussed whether that value was appropriate. The validity of expending costs now for benefits later was questioned. It was explained that this approach is being used for comparison with ALARA guidelines, and that the specific dollar value used is not significant as long as it is demonstrated that the ALARA guideline is generally not the most restrictive in limiting the waste activities.

Use of food pathways in calculating exposures was suggested, as was consistent use of Reference Reactor Year (RRY) and GWeyr units throughout the reports. Consideration of the on-site well pathway for exposure was also suggested. Discussions of competing risks (chemical toxicity, etc.) are outside the scope of this study, but are being pursued in other efforts.

John Adam discussed the tentative waste classification system comprised of the following categories:

Class A - destined for isolation
Class B - destined for licensed confinement
Class C - destined for unlicensed confinement

The panel discussed the meaning of isolation and concluded that isolation was not meant to necessarily imply zero release. Comparisons of the interface concentrations with present regulations (including 10 nCi/g of transuranic alpha activity) were also discussed.

The need for more information concerning parametric variations in calculations was suggested. Variations in burial

depth, containerization, and waste forms were discussed. Classification of wastes by source and desirability of segregation of wastes at the source are to be looked at in future studies.

Each of the various task reports was discussed along general lines. These seven task reports contained the basic methodology, interface concentrations, and applications to specific wastes. In discussing ground water migration in the Task I report, the problem of groundwater "short circuiting" in limestones was mentioned as a caution to be avoided.

The problem of surface erosion was also discussed. It was pointed out that there is some question as to how fast land surface will erode once it has been disturbed. Also the problem of gully erosion was mentioned as more serious than sheet erosion. These specific considerations would need to be included in determining site selection criteria.

The 100-year control period was discussed. Regulations specify what is to be allowed, not what is actually expected to happen. The 100-year period, later expanded to 150 years, is sufficient to greatly reduce the fission product activity.

A panel member representing a commercial burial site suggested that the burial ground operator should be given extra credit in establishing burial limits for such things as deeper burial or resistant containerization. This, however, was considered outside of the scope of this study.

There was considerable discussion on the Task 3 report with regard to the surface plate out on fuel element hulls and on internal reactor components. The extent of contamination plate out into such areas as the heat exchanger for instance is not known.

The table of isotopes considered should be expanded. In particular, ^{59}Ni should be included.

The validity of scaling activities with reactor power was questioned and explained.

In applying the methodology to mill tailings, the well-water scenario should be included. In addition, coal ash should also be examined as a low specific-activity radioactive waste.

Most of the remaining discussions on the site specific reports concerned questions on how the methodology was

applied but no specific problems with the reports were brought up.

Several other questions on numerical values in tables and figures were mentioned. These will be investigated and resolved in the final report.

The panel was thanked for participating in such a meaningful manner in the NRC's decisions making process. The comments and suggestions received will prove very useful in the study.

1.2 LIST OF ADVISORY PANEL MEMBERS

1. Bruce Bishop - New York State Energy Research and Development Administration
2. Michael Cloninger - Battelle Northwest Laboratories
3. George DeBuchananne - U.S. Geological Survey
4. Paul Dinner - Canada - Ontario Power
5. Thomas Elleman - North Carolina State University
6. Richard Frendburg - Atomic Industrial Forum - Nuclear Safety Associates
7. Philip Garrett - Atomic Industrial Forum
8. Bill Holcomb - U.S. Environmental Protection Agency
9. Terry Lash - Natural Resources Defense Council
10. Robert Newman - Allied Chemical
11. Richard Park - National Council on Radiation Protection
12. Robert Pohl - Cornell University
13. Walton Rodger - Atomic Industrial Forum - Nuclear Safety Associates
14. Heyward Shealy - South Carolina State Dept. of Health & Environmental Control
15. Gary Simon - California State Nuclear Assessment Office
16. James Steger - Division of Waste Management U.S. Department of Energy
17. David Symthe - Canada - Atomic Energy Control Board
18. Richard Turley - Western Interstate Energy Board
19. Jene Vance - Atomic Industrial Forum - Bechtel

Observers

William Bishop - U.S. Nuclear Regulatory Commission
Sally Mann - U.S. Department of Energy
Ragnwald Muller - Advisory Committee on Reactor Safeguards

APPENDIX B

BASIS OF PATHWAY MODELS

The sections in this Appendix contain the following discussions.

Section	Topic
B.1	Groundwater Migration
B.2	Direct Gamma Exposure
B.3	Air Transport-Transient
B.4	Air Transport-Continuous Release

APPENDIX B. BASIS OF PATHWAY MODELS

B.1 Groundwater Migration

In this investigation, the following second-order differential mass balance equations are used to describe nuclide migration in a two region ground water medium:

$$D \frac{\partial^2 C_1}{\partial x1^2} - V_1 \frac{\partial C_1}{\partial x_1} - K\lambda_d C_1 = 0 \qquad (A.1)$$

and

$$D \frac{\partial^2 C_2}{\partial x_2^2} - V_2 \frac{\partial C_2}{\partial x_2} - K \frac{\partial C_2}{\partial t_2} - K\lambda_d C_2 = 0$$

where,

D = dispersion coefficient (m^2/s)

C = nuclide release rate (Ci/yr)

V = water velocity (m/sec)

λ_d = decay constant (yr^{-1})

K = equilibrium sorption coefficient

x = distance along the region (m)

t = time (yr)

Because studies of ground water flow in soils suggest axial convection and dispersion are much greater than transverse convection and dispersion, a one-dimensional transport path is assumed.

The boundary condition needed to solve the mass balance equation in the first region is provided by the following expression:

$$C_1 (x_1 = 0, t_1) = \lambda_L I_m \exp(-\lambda_E t_1) \qquad (A.2)$$

where,

I_m = initial inventory (Ci)

λ_L = leach rate (yr^{-1})

$\lambda_E = \underset{(yr^{-1})}{\lambda_L} + \lambda_d$ = inventory total loss constant

λ_d = decay constant (yr^{-1})

The boundary condition for the second region is the release rate from the first region, which is approximated by the expression:

$$C_2\ (x_2 = 0, t_2) = Ae^{-a(t_2-\tau)} - Be^{-b(t_2-\tau)}$$

where,

$t_2 > \tau$

τ = arrival time at $x_2 = 0$ (yr)

A,B,a,b = constants determined by the form of the transient at the outlet from region 1.

Using these boundary conditions, the solutions to the mass balance equation for the two regions are found to be:

Region 1:

$$f_01 = \frac{C_1(x_1,t_1)}{\lambda_L I_m} = \frac{1}{2}\left\{\exp\left[\frac{V_1x_1}{2D} - \lambda_E t_1 - G_1x_1\right]\operatorname{erfc}\left[\frac{\sqrt{\frac{K}{D}}\,x_1 - 2\sqrt{\frac{D}{K}}\,G_1t_1}{2\sqrt{t_1}}\right]\right.$$
$$\left. + \exp\left[\frac{V_1x_1}{2D} - \lambda_E t_1 + G_1x_1\right]\operatorname{erfc}\left[\frac{\sqrt{\frac{K}{D}}\,x_1 + 2\sqrt{\frac{D}{K}}\,G_1t_1}{2\ \ t_1}\right]\right\}$$

where,

$$G_1 = \sqrt{\frac{V_1^2}{4D^2} - \frac{K(\lambda_E - \lambda_d)}{D}} \qquad (A.4)$$

and

$$\mathrm{erfc}(y) = 1 - \frac{2}{\sqrt{\pi}}\int_0^y e^{-z^2}dz$$

Region 2:

$$f_0\ (x_2,t_x) = \left\{ \frac{1}{2} \quad A \exp\left[\frac{V_2x_2}{2D} - a(t_2-\tau) - G_2x_2\right] \mathrm{erfc}\left[\frac{\frac{K}{D}\ x_2 - 2\ \frac{D}{K}\ G_2t_2}{2\ t_2}\right] \right.$$

$$-B \exp\left[\frac{V_2x_2}{2D} - b(t_2-\tau) - G_3x_3\right] \mathrm{erfc}\left[\frac{\frac{K}{D}\ x_2 - 2\ \frac{D}{K}\ G_3t_2}{2\ t_2}\right]$$

$$+A \exp\left[\frac{V_2x_2}{2D} - a(t_2-\tau) + G_2x_2\right] \mathrm{erfc}\left[\frac{\frac{K}{D}\ x_2 + 2\ \frac{D}{K}\ G_2t_2}{2\ t_2}\right]$$

$$\left. -B \exp\left[\frac{V_2x_2}{2D} - b(t_2-\tau) + G_3x_2\right] \mathrm{erfc}\left[\frac{\frac{K}{D}\ x_2 + 2\ \frac{D}{K}\ G_3t_2}{2\ t_2}\right] \right\}$$

(A.5)

where,

$$t_2 = t_1 - \tau \ \text{(yrs)}$$

$$G_2 = \sqrt{\frac{V_2^2}{4D_2} - \frac{K(\lambda_L - a)}{D}}$$

$$G_3 = \sqrt{\frac{V_2^2}{4D^2} - \frac{K(\lambda_L - b)}{D}}$$

It is noted that G_2 and G_3 can be imaginary for large K, λ_L or D and for small v. The form of the solution is not applicable because of limitations imposed by the form of the boundary conditions. A conservative adjustment is to reduce K or D. This is the modification employed in the present analysis.

In applying this formalism to an areal source, particularly for the well water event, the equations are applied at each mesh point and the result is summed and renormalized to obtain the proper value for f_0.

B.2 Direct Gamma Exposure

The basic equation for the dose rate from a flux of gamma rays is:

$$D = 0.0576\ \phi_g E_g \left(\frac{\mu a}{\rho}\right) T_x \qquad (A.6)$$

where

D = dose rate (mrem/yr)

ϕ_g = gamma flux (gammas/cm^2/sec)

E_g = average gamma ray energy (MeV)

μ_a/ρ = mass absorption coefficient for tissue

T_x = exposure time per year (hr/yr)

For an infinite disc source of infinite depth, ϕ_g is given by:

$$\phi_g = \frac{C_m G}{2\mu f} \qquad (A.7)$$

where

G = gamma fraction (gammas/sec/μCi)

f = average to maximum fraction of waste in site soil

μ = attenuation coefficient for waste

Substituting eq (A-7) into eq (A-6) and rearranging gives:

$$C_m = \frac{2\mu D_r}{(0.0576)G(\frac{\mu}{\rho})\ f\ E_g T_x} \qquad (A.8)$$

B.3 Air Transport - Transient Releases

The concentration at the plume centerline for an instantaneous point source is given by:[4]

$$d\chi_t = \frac{Qt\ (2\pi)^{-3/2}}{\sigma_x \sigma_y \sigma_z} \exp\left[-\frac{(x-\bar{u}t)^2}{2\sigma_x\ 2}\right] \qquad (A.9)$$

where

dx_t = concentration at time t (Ci/m^3)

Q_t = source strength, (Ci)

σ_i = dispersion coefficients, (m)

$\bar{u}$ = average wind speed, (m/s)

x = direction of plume axis (wind)

This equation is integrated and normalized to get the integrated concentration-time exposure:

$$\left(\frac{\chi_t}{Q_t}\right) = \frac{1}{\pi \sigma_y \sigma_z \bar{u}} \qquad (A.10)$$

where

(χ_t/Q_t) = normalized concentration time (sec/m^3)

For an F stability level, σ_y=7 and σ_z=3.5 and using $\bar{u}$ = 1.56 m/sec (3.5 mph):

$$\frac{\chi_t}{Q_t} = 8.33 \times 10^{-3}\ \frac{\text{Ci sec}}{\text{Ci released m}^3} \qquad (A.10a)$$

In order to correct for deposition of dust particles before they arrive at the site boundary, an effective source is calculated:

$$\frac{Q'}{Q} = \left[\exp \int_o^x \frac{dx}{\sigma_z \exp(h^2/2\sigma_z^2)}\right] = \sqrt{\frac{2}{\pi} \frac{Vd}{u}} \qquad (A.11)$$

x = distance from source to observation point in wind direction. The ratio Q_{eff}/Q_t at 160 m from the source for the conditions given above is 0.3 at 4 cm/sec deposition velocity.

The normalized integrated concentration exposure becomes,

$$\frac{\chi_t}{Q_t} = 2.5 \times 10^{-3} \text{ sec/m}^3 \qquad (A.12)$$

B.4 Air Transport-Continuous Releases

The model used for diffusion/dispersion transport is the gaussian plume dispersion technique described by Slade,[4] integrated over the area source. The concentration at point (x,y) resulting from a differential element source AdA can be described by:

$$\frac{\chi}{Q} = \frac{10^{-3}\, dAf_i}{\pi\sigma_y\, 3\bar{u}} \exp\left[\frac{-h^2}{2\sigma_z^2} - \frac{y^2}{2\sigma y^2}\right] \qquad (A.13)$$

where,

χ = concentration of contaminant at point (x,y) (Ci/l)

Q = source flux (Ci/m^2/sec)

dA = differential area element (m^2)

h = elevation of source release (m)

x = downwind distance (m)

y = distance from plume centerline (m)

f_i = frequency for direction i at each stability category

The source flux is corrected for particle deposition using eq (A-11).

In using eq (A.13) as a point source, the term QdA is replaced by Q(Ci/sec) and no area integration is performed.

REFERENCES FOR APPENDIX B

1. Coates, R.L. and N.R. Horton, "RSAC--A Radiological Safety Analysis Computer Program," IDO-17151, May 1966.

2. Morgan, Karl Z., W.S. Snyder, and J.A. Auxier (eds), "Report of ICRP Committe II on Permissable Dose for Internal Radiation (1959), with Bibliography for Biological, Mathematical, and Physical Data," Health Phys. 3 (June, 1960).

3. "AERIN Computer Code", Lawrence Livermore Laboratory Report.

4. Slade, D.H., Meterology and Atomic Energy 1968, U.S. A.E.C., TID 24190, July 1968.

APPENDIX C

IMPACTS OF DECAY DAUGHTERS

Tables 1-13 contain the relative impacts at 100 years from the major decay chains. Figures 1-11 contain the relative impacts as a function of time.

TABLE 1 DECAY DAUGHTER CONCENTRATIONS AND RELATIVE IMPACTS FOR THE ^{244}Cm DECAY CHAIN

ISOTOPE	INITIAL PARENT CONCENTRATION (μCi/cc)	MPC (μCi/cc)	*	HALF LIFE (YEAR)	BRANCHING RATIO	CONCENTRATION AT 100 YEARS (μCi/cc)	RELATIVE IMPACT †
CM244	2.4+02	3.-13	S	1.8+01	1.000000	5.1+00	1.0+00
PU240		6.-14	S	6.5+03	1.000000	6.5-01	6.5-01
U 236		4.-12	I	2.3+07	1.000000	1.5-06	2.2-08
TH232		1.-12	S	1.4+10	1.000000	3.4-15	2.1-16
RA228		1.-12	I	5.7+00	1.000000	2.9-15	1.7-16
AC228		6.-10	I	6.9-04	1.000000	2.9-15	2.9-19
TH228		2.-13	I	1.9+00	1.000000	2.7-15	8.2-16
RA224		2.-11	I	9.8-03	1.000000	2.7-15	8.2-18
RN220		2.-14	A	1.7-06	1.000000	2.7-15	8.2-15
PO216		2.-14	A	5.1-09	1.000000	2.7-15	8.2-15
AT216		2.-14	A	9.5-12	.000100	2.7-19	8.2-19
PB212		6.-10	S	1.2-03	.999900	2.7-15	2.7-19
BI212		3.-09	S	1.1-04	1.000000	2.7-15	5.4-20
PO212		2.-14	A	9.5-15	.663000	1.8-15	5.4-15
TL208		3.-06	C	5.9-06	.337000	9.2-16	1.8-23

*S=(SOLUABLE)
I=(INSOLUABLE)
THE FOLLOWING INDICATE DEFAULT VALUES:
A=(ALPHA EMITTER NOT LISTED IN 10CFR20, MPC=2×10^{-14})
B=(BETA EMITTER WITH HALF LIFE GREATER THAN 2 HOURS AND MPC NOT LISTED IN 10CFR20, MPC=1×10^{-10})
C=(BETA EMITTER WITH HALF LIFE LESS THAN 2 HOURS AND MPC NOT LISTED IN 10CFR20, MPC=3×10^{-6})

TABLE 2 DECAY DAUGHTER CONCENTRATIONS AND RELATIVE IMPACTS FOR THE ^{240}Pu DECAY CHAIN

ISOTOPE	INITIAL PARENT CONCENTRATION (μCi/cc)	MPC (μCi/cc)	*	HALF LIFE (YEAR)	BRANCHING RATIO	CONCENTRATION AT 100 YEARS (μCi/cc)	RELATIVE IMPACT †
PU240	1.0+00	6.-14	S	6.5+03	1.000000	9.9-01	9.9-01
U 236		4.-12	I	2.3+07	1.000000	3.0-06	4.5-08
TH232		1.-12	S	1.4+10	1.000000	8.0-15	4.8-16
RA228		1.-12	I	5.7+00	1.000000	6.9-15	4.2-16
AC228		6.-10	I	6.9-04	1.000000	6.9-15	6.9-19
TH228		2.-13	I	1.9+00	1.000000	6.6-15	2.0-15
RA224		2.-11	I	9.8-03	1.000000	6.6-15	2.0-17
RN220		2.-14	A	1.7-06	1.000000	6.6-15	2.0-14
PO216		2.-14	A	5.1-09	1.000000	6.6-15	2.0-14
AT216		2.-14	A	9.5-12	.000100	6.6-19	2.0-18
PB212		6.-10	S	1.2-03	.999900	6.6-15	6.6-19
BI212		3.-09	S	1.1-04	1.000000	6.6-15	1.3-19
PO212		2.-14	A	9.5-15	.663000	4.3-15	1.3-14
TL208		3.-06	C	5.9-06	.337000	2.2-15	4.4-23

*S=(SOLUABLE)
I=(INSOLUABLE)
THE FOLLOWING INDICATE DEFAULT VALUES:
A=(ALPHA EMITTER NOT LISTED IN 10CFR20, MPC=2×10^{-14})
B=(BETA EMITTER WITH HALF LIFE GREATER THAN 2 HOURS AND MPC NOT LISTED IN 10CFR20, MPC=1×10^{-10})
C=(BETA EMITTER WITH HALF LIFE LESS THAN 2 HOURS AND MPC NOT LISTED IN 10CFR20, MPC=3×10^{-6})

TABLE 3 DECAY DAUGHTER CONCENTRATIONS AND RELATIVE IMPACTS FOR THE ^{232}Th DECAY CHAIN

ISOTOPE	INITIAL PARENT CONCENTRATION (μCi/cc)	MPC (μCi/cc)	*	HALF LIFE (YEAR)	BRANCHING RATIO	CONCENTRATION AT 100 YEARS (μCi/cc)	RELATIVE IMPACT †
TH232	1.7+01	1.-12	S	1.4+10	1.000000	1.7+01	1.0+00
RA228		1.-12	I	5.7+00	1.000000	1.7+01	1.0+00
AC228		6.-10	I	6.9-04	1.000000	1.7+01	1.7-03
TH228		2.-13	I	1.9+00	1.000000	1.7+01	5.1+00
RA224		2.-11	I	9.8-03	1.000000	1.7+01	5.1-02
RN220		2.-14	A	1.7-06	1.000000	1.7+01	5.1+01
PO216		2.-14	A	5.1-09	1.000000	1.7+01	5.1+01
AT216		2.-14	A	9.5-12	.000100	1.7-03	5.1-03
PB212		6.-10	S	1.2-03	.999900	1.7+01	1.7-03
BI212		3.-09	S	1.1-04	1.000000	1.7+01	3.4-04
PO212		2.-14	A	9.5-15	.663000	1.1+01	3.4+01
TL208		3.-06	C	5.9-06	.337000	5.7+00	1.1-07

*S=(SOLUABLE)
I=(INSOLUABLE)
THE FOLLOWING INDICATE DEFAULT VALUES:
A=(ALPHA EMITTER NOT LISTED IN 10CFR20, MPC=2×10^{-14})
B=(BETA EMITTER WITH HALF LIFE GREATER THAN 2 HOURS AND MPC NOT LISTED IN 10CFR20, MPC=1×10^{-10})
C=(BETA EMITTER WITH HALF LIFE LESS THAN 2 HOURS AND MPC NOT LISTED IN 10CFR20, MPC=3×10^{-6})

TABLE 4 DECAY DAUGHTER CONCENTRATIONS AND RELATIVE IMPACTS FOR THE ^{243}Am DECAY CHAIN

ISOTOPE	INITIAL PARENT CONCENTRATION (μCi/cc)	MPC (μCi/cc)	*	HALF LIFE (YEAR)	BRANCHING RATIO	CONCENTRATION AT 100 YEARS (μCi/cc)	RELATIVE IMPACT †
AM243	3.4+00	2.-13	S	7.4+03	1.000000	3.4+00	1.0+00
NP239		2.-08	I	6.4-03	1.000000	3.4+00	1.0-05
PU239		6.-14	S	2.4+04	1.000000	9.8-03	9.8-03
U 235		4.-12	I	7.0+08	1.000000	4.8-10	7.3-12
TH231		4.-08	I	2.9-03	1.000000	4.8-10	7.3-16
PA231		4.-14	S	3.2+04	1.000000		
AC227		8.-14	S	2.2+01	1.000000		
TH227		6.-12	I	5.1-02	.988000		
FR223		3.-06	C	4.2-05	.012000		
RA223		8.-12	I	3.1-02	1.000000		
RN219		2.-14	A	1.2-07	1.000000		
PO215		2.-14	A	5.6-11	1.000000		
AT215		2.-14	A	3.2-13	.000005		
PB211		3.-06	C	6.8-05	.999995		
BI211		2.-14	A	4.2-06	1.000000		
PO211		2.-14	A	1.6-08	.003200		
TL207		3.-06	C	9.1-06	.996800		

*S=(SOLUABLE)
I=(INSOLUABLE)
THE FOLLOWING INDICATE DEFAULT VALUES:
A=(ALPHA EMITTER NOT LISTED IN 10CFR20, MPC=2×10^{-14})
B=(BETA EMITTER WITH HALF LIFE GREATER THAN 2 HOURS AND MPC NOT LISTED IN 10CFR20, MPC=1×10^{-10})
C=(BETA EMITTER WITH HALF LIFE LESS THAN 2 HOURS AND MPC NOT LISTED IN 10CFR20, MPC=3×10^{-6})

TABLE 5 DECAY DAUGHTER CONCENTRATIONS AND RELATIVE IMPACTS FOR THE ^{239}Pu DECAY CHAIN

ISOTOPE	INITIAL PARENT CONCENTRATION (μCi/cc)	MPC (μCi/cc)	*	HALF LIFE (YEAR)	BRANCHING RATIO	CONCENTRATION AT 100 YEARS (μCi/cc)	RELATIVE IMPACT †
PU239	1.0+00	6.-14	S	2.4+04	1.000000	1.0+00	1.0+00
U 235		4.-12	I	7.0+08	1.000000	9.9-08	1.5-09
TH231		4.-08	I	2.9-03	1.000000	9.9-08	1.5-13
PA231		4.-14	S	3.2+04	1.000000	1.1-10	1.6-10
AC227		8.-14	S	2.2+01	1.000000	5.9-11	4.4-11
TH227		6.-12	I	5.1-02	.988000	5.8-11	5.8-13
FR223		3.-06	C	4.2-05	.012000	7.1-13	1.4-20
RA223		8.-12	I	3.1-02	1.000000	5.9-11	4.4-13
RN219		2.-14	A	1.2-07	1.000000	5.9-11	1.8-10
PO215		2.-14	A	5.6-11	1.000000	5.9-11	1.8-10
AT215		2.-14	A	3.2-13	.000005	3.0-16	8.9-16
PB211		3.-06	C	6.8-05	.999995	5.9-11	1.2-18
BI211		2.-14	A	4.2-06	1.000000	5.9-11	1.8-10
PO211		2.-14	A	1.6-08	.003200	1.9-13	5.7-13
TL207		3.-06	C	9.1-06	.996800	5.9-11	1.2-18

*S=(SOLUABLE)
I=(INSOLUABLE)

THE FOLLOWING INDICATE DEFAULT VALUES:
A=(ALPHA EMITTER NOT LISTED IN 10CFR20, MPC=2×10^{-14})
B=(BETA EMITTER WITH HALF LIFE GREATER THAN 2 HOURS AND MPC NOT LISTED IN 10CFR20, MPC=1×10^{-10})
C=(BETA EMITTER WITH HALF LIFE LESS THAN 2 HOURS AND MPC NOT LISTED IN 10CFR20, MPC=3×10^{-6})

TABLE 6 DECAY DAUGHTER CONCENTRATIONS AND RELATIVE IMPACTS FOR THE ^{235}U DECAY CHAIN

ISOTOPE	INITIAL PARENT CONCENTRATION (μCi/cc)	MPC (μCi/cc)	*	HALF LIFE (YEAR)	BRANCHING RATIO	CONCENTRATION AT 100 YEARS (μCi/cc)	RELATIVE IMPACT †
U 235	6.7+01	4.-12	I	7.0+08	1.000000	6.7+01	1.0+00
TH231		4.-08	I	2.9-03	1.000000	6.7+01	1.0-04
PA231		4.-14	S	3.2+04	1.000000	1.4-01	2.2-01
AC227		8.-14	S	2.2+01	1.000000	1.0-01	7.6-02
TH227		6.-12	I	5.1-02	.988000	1.0-01	1.0-03
FR223		3.-06	C	4.2-05	.012000	1.2-03	2.4-11
RA223		8.-12	I	3.1-02	1.000000	1.0-01	7.6-04
RN219		2.-14	A	1.2-07	1.000000	1.0-01	3.0-01
PO215		2.-14	A	5.6-11	1.000000	1.0-01	3.0-01
AT215		2.-14	A	3.2-13	.000005	5.0-07	1.5-06
PB211		3.-06	C	6.8-05	.999995	1.0-01	2.0-09
BI211		2.-14	A	4.2-06	1.000000	1.0-01	3.0-01
PO211		2.-14	A	1.6-08	.003200	3.2-04	9.7-04
TL207		3.-06	C	9.1-06	.996800	1.0-01	2.0-09

*S=(SOLUABLE)
I=(INSOLUABLE)
THE FOLLOWING INDICATE DEFAULT VALUES:
A=(ALPHA EMITTER NOT LISTED IN 10CFR20, MPC=2×10^{-14})
B=(BETA EMITTER WITH HALF LIFE GREATER THAN 2 HOURS AND MPC NOT LISTED IN 10CFR20, MPC=1×10^{-10})
C=(BETA EMITTER WITH HALF LIFE LESS THAN 2 HOURS AND MPC NOT LISTED IN 10CFR20, MPC=3×10^{-6})

TABLE 7 DECAY DAUGHTER CONCENTRATIONS AND RELATIVE IMPACTS FOR THE ^{242}Cm DECAY CHAIN

ISOTOPE	INITIAL PARENT CONCENTRATION (μCi/cc)	MPC (μCi/cc)	*	HALF LIFE (YEAR)	BRANCHING RATIO	CONCENTRATION AT 100 YEARS (μCi/cc)	RELATIVE IMPACT †
CM242	4.0+03	4.-12	S	4.5-01	1.000000	0.0	0.0
PU238		7.-14	S	8.8+01	1.000000	9.4+00	8.0+00
U 234		4.-12	I	2.4+05	1.000000	4.1-03	6.1-05
TH230		8.-14	S	7.7+04	1.000000	2.1-06	1.5-06
RA226		2.-12	I	1.6+03	1.000000	3.1-08	9.3-10
RN222		1.-09	S	1.0-02	1.000000	3.1-08	1.9-12
PO218		2.-14	A	5.7-06	1.000000	3.1-08	9.3-08
AT218		2.-14	A	4.1-08	.000300	9.3-12	2.8-11
PB214		3.-06	C	5.1-05	.999700	3.1-08	6.2-16
BI214		2.-14	A	3.8-05	1.000000	3.1-08	9.3-08
PO214		2.-14	A	5.2-12	.999600	3.1-08	9.3-08
TL210		3.-06	C	2.5-06	.000400	1.2-11	2.5-19
PB210		4.-12	S	2.2+01	1.000000	1.5-08	2.2-10
BI210		2.-10	S	1.4-02	1.000000	1.5-08	4.4-12
PO210		2.-11	S	3.8-01	.999999	1.5-08	4.4-11
TL206		3.-06	C	8.0-06	.000001	1.5-14	2.9-22

*S=(SOLUABLE)
I=(INSOLUABLE)
THE FOLLOWING INDICATE DEFAULT VALUES:
A=(ALPHA EMITTER NOT LISTED IN 10CFR20, MPC=2×10^{-14})
B=(BETA EMITTER WITH HALF LIFE GREATER THAN 2 HOURS AND MPC NOT LISTED IN 10CFR20, MPC=1×10^{-10})
C=(BETA EMITTER WITH HALF LIFE LESS THAN 2 HOURS AND MPC NOT LISTED IN 10CFR20, MPC=3×10^{-6})

TABLE 8 DECAY DAUGHTER CONCENTRATIONS AND RELATIVE IMPACTS FOR THE ^{238}Pu DECAY CHAIN

ISOTOPE	INITIAL PARENT CONCENTRATION (μCi/cc)	MPC (μCi/cc)	*	HALF LIFE (YEAR)	BRANCHING RATIO	CONCENTRATION AT 100 YEARS (μCi/cc)	RELATIVE IMPACT †
PU238	2.6+00	7.-14	S	8.8+01	1.000000	1.2+00	1.0+00
U 234		4.-12	I	2.4+05	1.000000	5.2-04	7.8-06
TH230		8.-14	S	7.7+04	1.000000	2.6-07	2.0-07
RA226		2.-12	I	1.6+03	1.000000	4.0-09	1.2-10
RN222		1.-09	S	1.0-02	1.000000	4.0-09	2.4-13
PO218		2.-14	A	5.7-06	1.000000	4.0-09	1.2-08
AT218		2.-14	A	4.1-08	.000300	1.2-12	3.6-12
PB214		3.-06	C	5.1-05	.999700	4.0-09	8.0-17
BI214		2.-14	A	3.8-05	1.000000	4.0-09	1.2-08
PO214		2.-14	A	5.2-12	.999600	4.0-09	1.2-08
TL210		3.-06	C	2.5-06	.000400	1.6-12	3.2-20
PB210		4.-12	S	2.2+01	1.000000	1.9-09	2.9-11
BI210		2.-10	S	1.4-02	1.000000	1.9-09	5.8-13
PO210		2.-11	S	3.8-01	.999999	1.9-09	5.7-12
TL206		3.-06	C	8.0-06	.000001	1.9-15	3.8-23

*S=(SOLUABLE)
I=(INSOLUABLE)
THE FOLLOWING INDICATE DEFAULT VALUES:
A=(ALPHA EMITTER NOT LISTED IN 10CFR20, MPC=2×10^{-14})
B=(BETA EMITTER WITH HALF LIFE GREATER THAN 2 HOURS AND MPC NOT LISTED IN 10CFR20, MPC=1×10^{-10})
C=(BETA EMITTER WITH HALF LIFE LESS THAN 2 HOURS AND MPC NOT LISTED IN 10CFR20, MPC=3×10^{-6})

TABLE 9 DECAY DAUGHTER CONCENTRATIONS AND RELATIVE IMPACTS FOR THE ^{242}Pu DECAY CHAIN

ISOTOPE	INITIAL PARENT CONCENTRATION (μCi/cc)	MPC (μCi/cc)		HALF LIFE (YEAR)	BRANCHING RATIO	CONCENTRATION AT 100 YEARS (μCi/cc)	RELATIVE IMPACT †
PU242	1.0+00	6.-14	S	3.9+05	1.000000	1.0+00	1.0+00
U 238		3.-12	S	4.5+09	1.000000	1.5-08	3.1-10
TH234		1.-09	I	6.6-02	1.000000	1.5-08	9.2-13
PA234		1.-10	B	2.3-06	1.000000	1.5-08	9.2-12
U 234		4.-12	I	2.4+05	1.000000	1.6-12	2.4-14
TH230		8.-14	S	7.7+04	1.000000		
RA226		2.-12	I	1.6+03	1.000000		
RN222		1.-09	S	1.0-02	1.000000		
PO218		2.-14	A	5.7-06	1.000000		
AT218		2.-14	A	4.1-08	.000300		
PB214		3.-06	C	5.1-05	.999700		
BI214		2.-14	A	3.8-05	1.000000		
PO214		2.-14	A	5.2-12	.999600		
TL210		3.-06	C	2.5-06	.000400		
PB210		4.-12	S	2.2+01	1.000000		
BI210		2.-10	S	1.4-02	1.000000		
PO210		2.-11	S	3.8-01	.999999		
TL206		3.-06	C	8.0-06	.000001		

*S=(SOLUABLE)
I=(INSOLUABLE)
THE FOLLOWING INDICATE DEFAULT VALUES:
A=(ALPHA EMITTER NOT LISTED IN 10CFR20, MPC=2×10^{-14})
B=(BETA EMITTER WITH HALF LIFE GREATER THAN 2 HOURS AND MPC NOT LISTED IN 10CFR20, MPC=1×10^{-10})
C=(BETA EMITTER WITH HALF LIFE LESS THAN 2 HOURS AND MPC NOT LISTED IN 10CFR20, MPC=3×10^{-6})

TABLE 10 DECAY DAUGHTER CONCENTRATIONS AND RELATIVE IMPACTS FOR THE ^{238}U DECAY CHAIN

ISOTOPE	INITIAL PARENT CONCENTRATION (μCi/cc)	MPC (μCi/cc)	*	HALF LIFE (YEAR)	BRANCHING RATIO	CONCENTRATION AT 100 YEARS (μCi/cc)	RELATIVE IMPACT †
U 238	5.0+01	3.-12	S	4.5+09	1.000000	5.0+01	1.0+00
TH234		1.-09	I	6.6-02	1.000000	5.0+01	3.0-03
PA234		1.-10	B	2.3-06	1.000000	5.0+01	3.0-02
U 234		4.-12	I	2.4+05	1.000000	1.4-02	2.2-04
TH230		8.-14	S	7.7+04	1.000000	6.6-06	4.9-06
RA226		2.-12	I	1.6+03	1.000000	1.6-07	4.8-09
RN222		1.-09	S	1.0-02	1.000000	1.6-07	9.6-12
PO218		2.-14	A	5.7-06	1.000000	1.6-07	4.8-07
AT218		2.-14	A	4.1-08	.000300	4.8-11	1.4-10
PB214		3.-06	C	5.1-05	.999700	1.6-07	3.2-15
BI214		2.-14	A	3.8-05	1.000000	1.6-07	4.8-07
PO214		2.-14	A	5.2-12	.999600	1.6-07	4.8-07
TL210		3.-06	C	2.5-06	.000400	6.4-11	1.3-18
PB210		4.-12	S	2.2+01	1.000000	1.1-07	1.7-09
BI210		2.-10	S	1.4-02	1.000000	1.1-07	3.3-11
PO210		2.-11	S	3.8-01	.999999	1.1-07	3.3-10
TL206		3.-06	C	8.0-06	.000001	1.1-13	2.2-21

*S=(SOLUABLE)
I=(INSOLUABLE)
THE FOLLOWING INDICATE DEFAULT VALUES:
A=(ALPHA EMITTER NOT LISTED IN 10CFR20, MPC=2×10^{-14})
B=(BETA EMITTER WITH HALF LIFE GREATER THAN 2 HOURS AND MPC NOT LISTED IN 10CFR20, MPC=1×10^{-10})
C=(BETA EMITTER WITH HALF LIFE LESS THAN 2 HOURS AND MPC NOT LISTED IN 10CFR20, MPC=3×10^{-6})

TABLE 11 DECAY DAUGHTER CONCENTRATIONS AND RELATIVE IMPACTS FOR THE ^{241}Pu DECAY CHAIN

ISOTOPE	INITIAL PARENT CONCENTRATION (μCi/cc)	MPC (μCi/cc)	*	HALF LIFE (YEAR)	BRANCHING RATIO	CONCENTRATION AT 100 YEARS (μCi/cc)	RELATIVE IMPACT †
PU241	3.0+03	3.-12	S	1.5+01	1.000000	3.0+01	5.9-01
AM241		2.-13	S	4.3+02	1.000000	9.1+01	2.7+01
NP237		1.-13	S	2.1+06	1.000000	2.5-03	1.5-03
PA233		6.-09	I	7.5-02	1.000000	2.5-03	2.5-08
U 233		4.-12	I	1.6+05	1.000000	4.7-07	7.1-09
TH229		2.-14	A	7.3+03	1.000000	1.3-09	4.0-09
RA225		1.-10	B	3.8-02	1.000000	1.3-09	8.1-13
AC225		2.-14	A	2.7-02	1.000000	1.3-09	4.0-09
FR221		2.-14	A	9.5-06	1.000000	1.3-09	4.0-09
AT217		2.-14	A	6.3-10	1.000000	1.3-09	4.0-09
BI213		6.-06	C	8.9-05	1.000000	1.3-09	1.3-17
PO213		2.-14	A	1.3-13	.980000	1.3-09	3.9-09
TL209		3.-06	C	3.8-06	.020000	2.7-11	5.4-19
PB209		1.-10	B	3.4-04	1.000000	1.3-09	8.1-13

*S=(SOLUABLE)
I=(INSOLUABLE)
THE FOLLOWING INDICATE DEFAULT VALUES:
A=(ALPHA EMITTER NOT LISTED IN 10CFR20, MPC=2×10^{-14})
B=(BETA EMITTER WITH HALF LIFE GREATER THAN 2 HOURS AND MPC NOT LISTED IN 10CFR20, MPC=1×10^{-10})
C=(BETA EMITTER WITH HALF LIFE LESS THAN 2 HOURS AND MPC NOT LISTED IN 10CFR20, MPC=3×10^{-6})

TABLE 12 DECAY DAUGHTER CONCENTRATIONS AND RELATIVE IMPACTS FOR THE ^{241}Am DECAY CHAIN

ISOTOPE	INITIAL PARENT CONCENTRATION (μCi/cc)	MPC (μCi/cc)	*	HALF LIFE (YEAR)	BRANCHING RATIO	CONCENTRATION AT 100 YEARS (μCi/cc)	RELATIVE IMPACT †
AM241	4.0+00	2.-13	S	4.3+02	1.000000	3.4+00	1.0+00
NP237		1.-13	S	2.1+06	1.000000	1.2-04	7.3-05
PA233		6.-09	I	7.5-02	1.000000	1.2-04	1.2-09
U 233		4.-12	I	1.6+05	1.000000	2.7-08	4.1-10
TH229		2.-14	A	7.3+03	1.000000	8.8-11	2.6-10
RA225		1.-10	B	3.8-02	1.000000	8.8-11	5.3-14
AC225		2.-14	A	2.7-02	1.000000	8.8-11	2.6-10
FR221		2.-14	A	9.5-06	1.000000	8.8-11	2.6-10
AT217		2.-14	A	6.3-10	1.000000	8.8-11	2.6-10
BI213		6.-06	C	8.9-05	1.000000	8.8-11	8.8-19
PO213		2.-14	A	1.3-13	.980000	8.6-11	2.6-10
TL209		3.-06	C	3.8-06	.020000	1.8-12	3.5-20
PB209		1.-10	B	3.4-04	1.000000	8.8-11	5.3-14

*S=(SOLUABLE)
I=(INSOLUABLE)
THE FOLLOWING INDICATE DEFAULT VALUES:
A=(ALPHA EMITTER NOT LISTED IN 10CFR20, MPC=2×10^{-14})
B=(BETA EMITTER WITH HALF LIFE GREATER THAN 2 HOURS AND MPC NOT LISTED IN 10CFR20, MPC=1×10^{-10})
C=(BETA EMITTER WITH HALF LIFE LESS THAN 2 HOURS AND MPC NOT LISTED IN 10CFR20, MPC=3×10^{-6})

TABLE 13 DECAY DAUGHTER CONCENTRATIONS AND RELATIVE IMPACTS FOR THE ^{237}Np DECAY CHAIN

ISOTOPE	INITIAL PARENT CONCENTRATION (μCi/cc)	MPC (μCi/cc)	*	HALF LIFE (YEAR)	BRANCHING RATIO	CONCENTRATION AT 100 YEARS (μCi/cc)	RELATIVE IMPACT †
NP237	1.7+00	1.-13	S	2.1+06	1.000000	1.7+00	1.0+00
PA233		6.-09	I	7.5-02	1.000000	1.7+00	1.7-05
U 233		4.-12	I	1.6+05	1.000000	7.4-04	1.1-05
TH229		2.-14	A	7.3+03	1.000000	3.5-06	1.0-05
RA225		1.-10	B	3.8-02	1.000000	3.5-06	2.1-09
AC225		2.-14	A	2.7-02	1.000000	3.5-06	1.0-05
FR221		2.-14	A	9.5-06	1.000000	3.5-06	1.0-05
AT217		2.-14	A	6.3-10	1.000000	3.5-06	1.0-05
BI213		6.-06	C	8.9-05	1.000000	3.5-06	3.5-14
PO213		2.-14	A	1.3-13	.980000	3.4-06	1.0-05
TL209		3.-06	C	3.8-06	.020000	6.9-08	1.4-15
PB209		1.-10	B	3.4-04	1.000000	3.5-06	2.1-09

*S=(SOLUABLE)
I=(INSOLUABLE)
THE FOLLOWING INDICATE DEFAULT VALUES:
A=(ALPHA EMITTER NOT LISTED IN 10CFR20, MPC=2×10^{-14})
B=(BETA EMITTER WITH HALF LIFE GREATER THAN 2 HOURS AND MPC NOT LISTED IN 10CFR20, MPC=1×10^{-10})
C=(BETA EMITTER WITH HALF LIFE LESS THAN 2 HOURS AND MPC NOT LISTED IN 10CFR20, MPC=3×10^{-6})

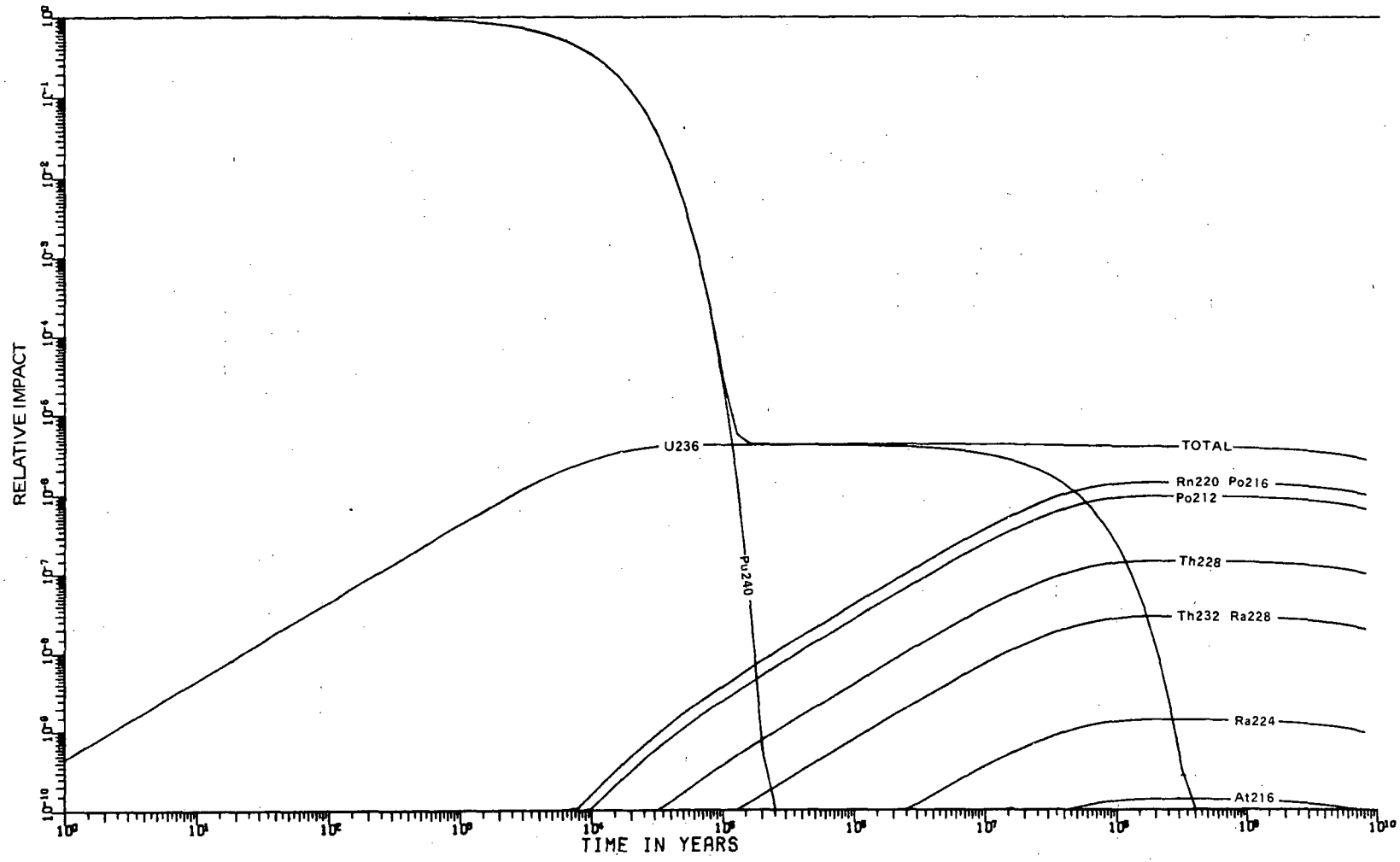

FIGURE 1 RELATIVE IMPACT OF MEMBERS OF THE ^{240}Pu DECAY CHAIN

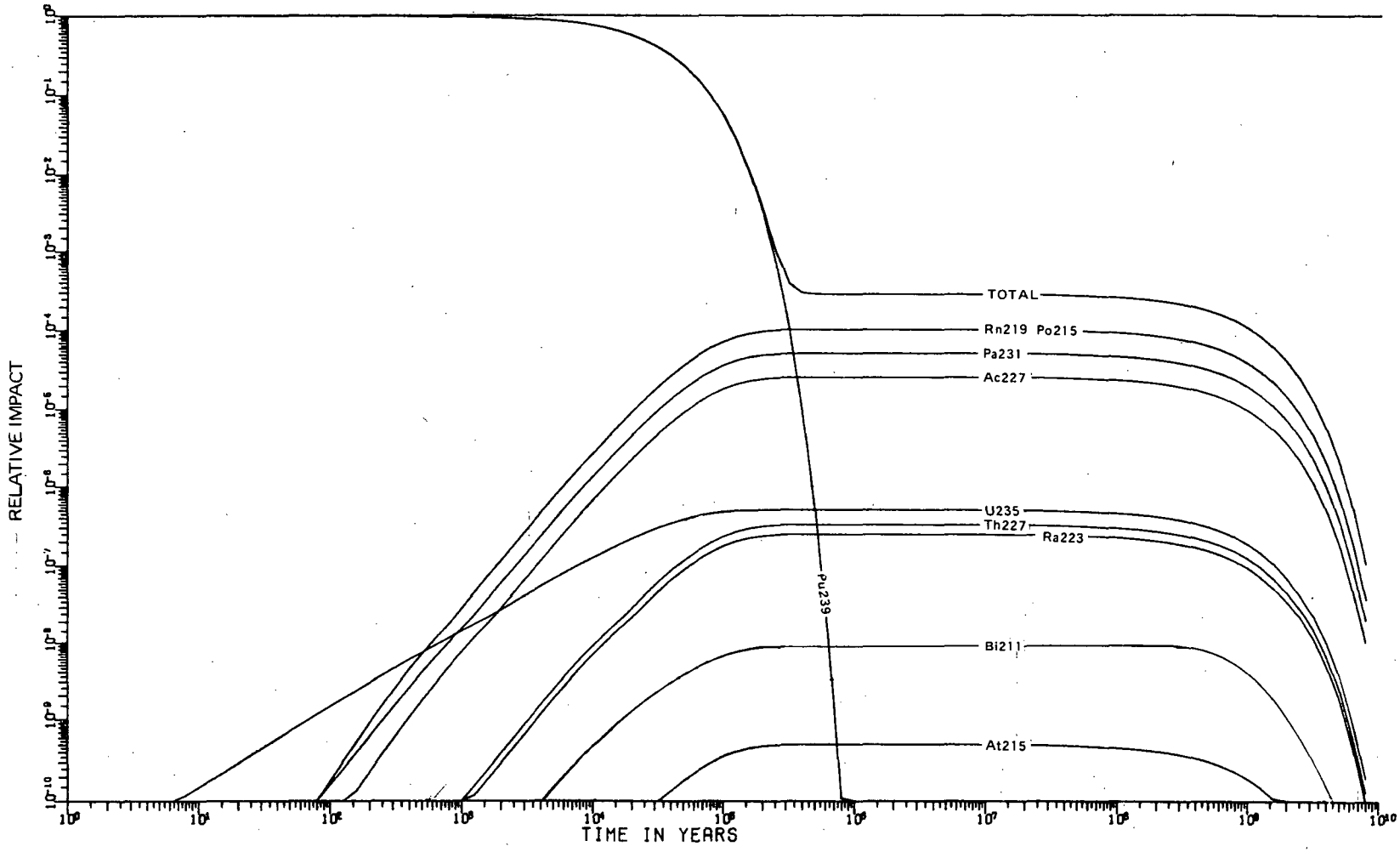

FIGURE 4 RELATIVE IMPACT OF THE MEMBERS OF THE ^{239}Pu DECAY CHAIN

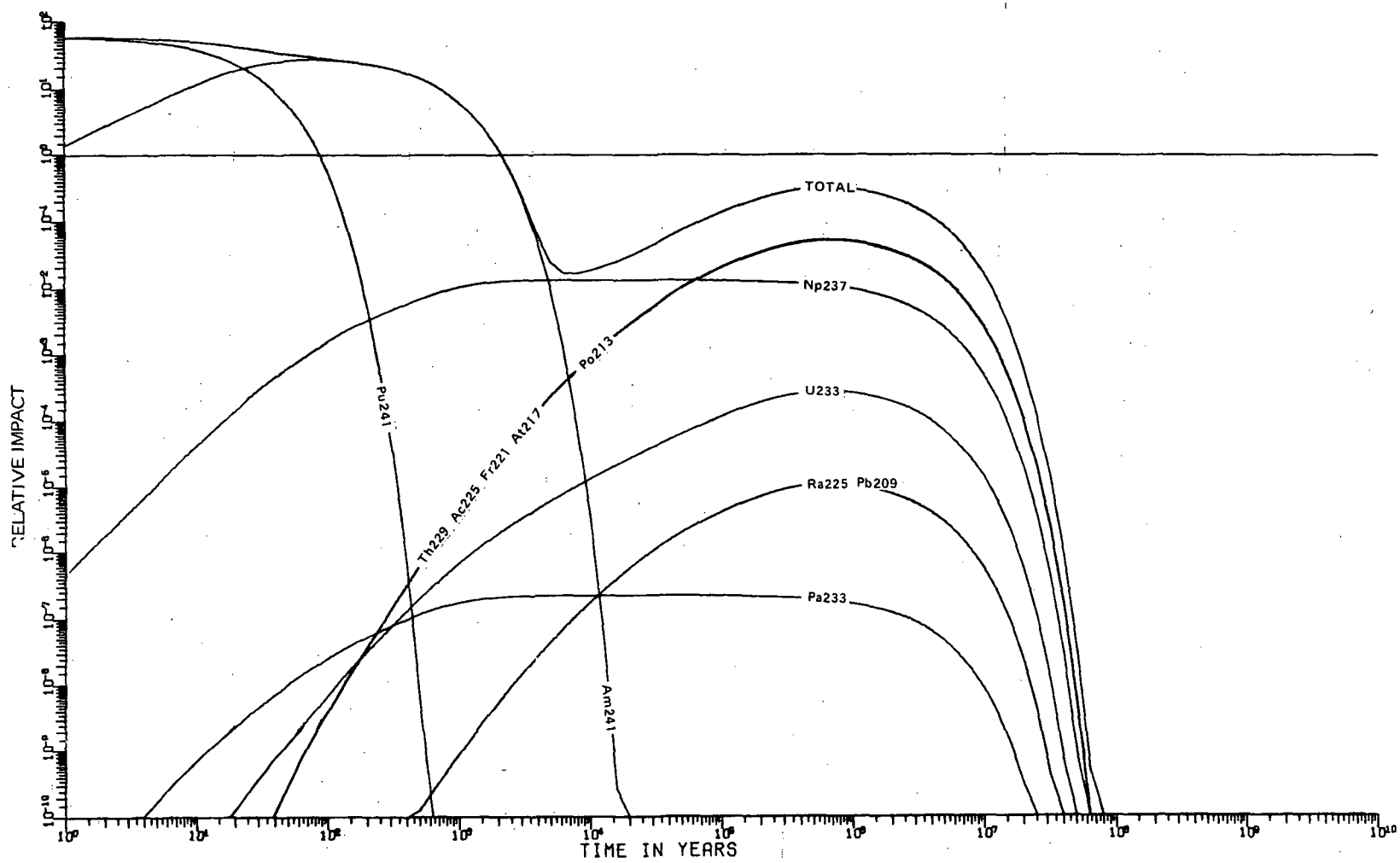

FIGURE 9 RELATIVE IMPACT OF THE MEMBERS OF THE ^{241}Pu DECAY CHAIN

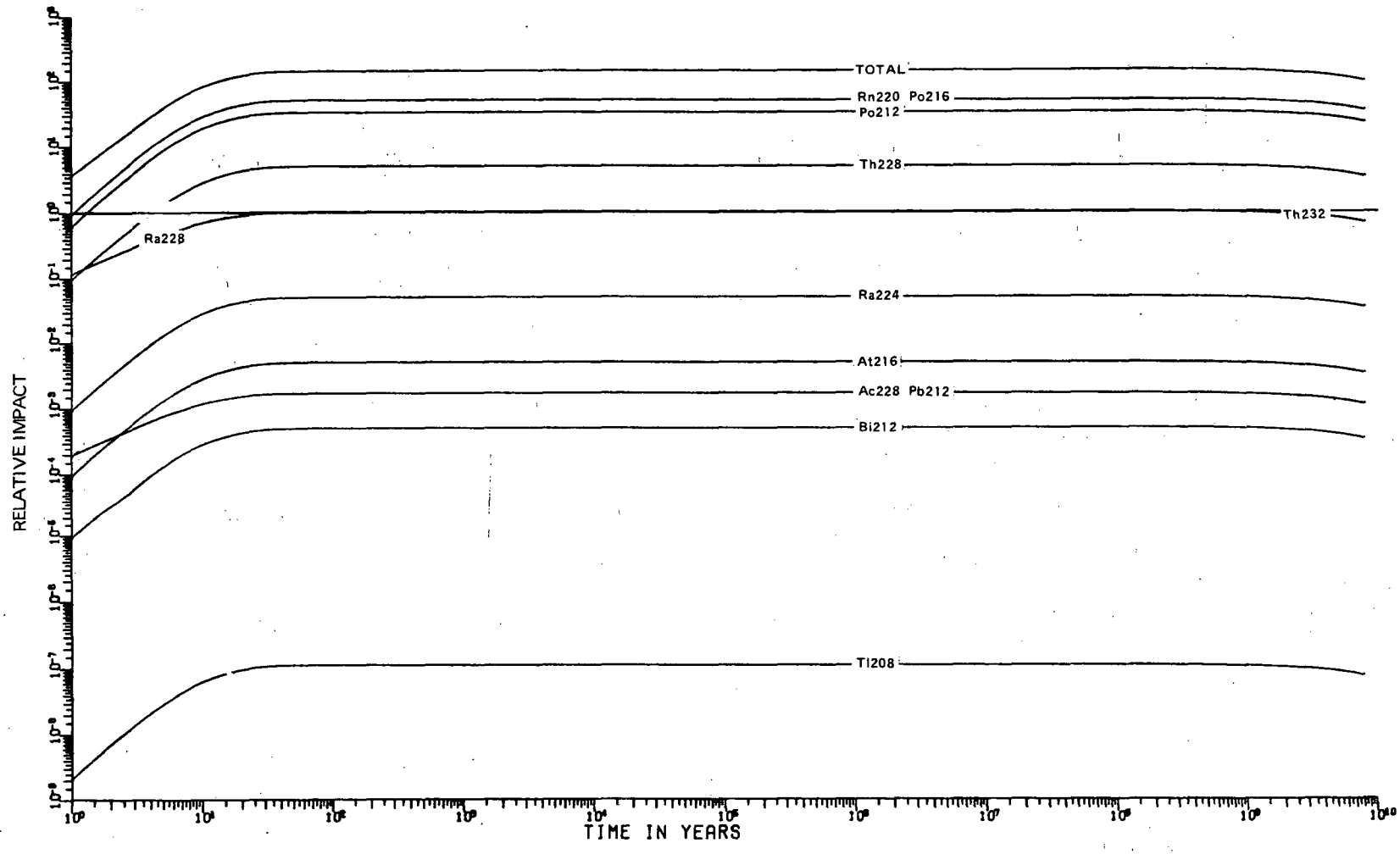

FIGURE 2 RELATIVE IMPACT OF THE MEMBERS OF THE ^{232}Th DECAY CHAIN

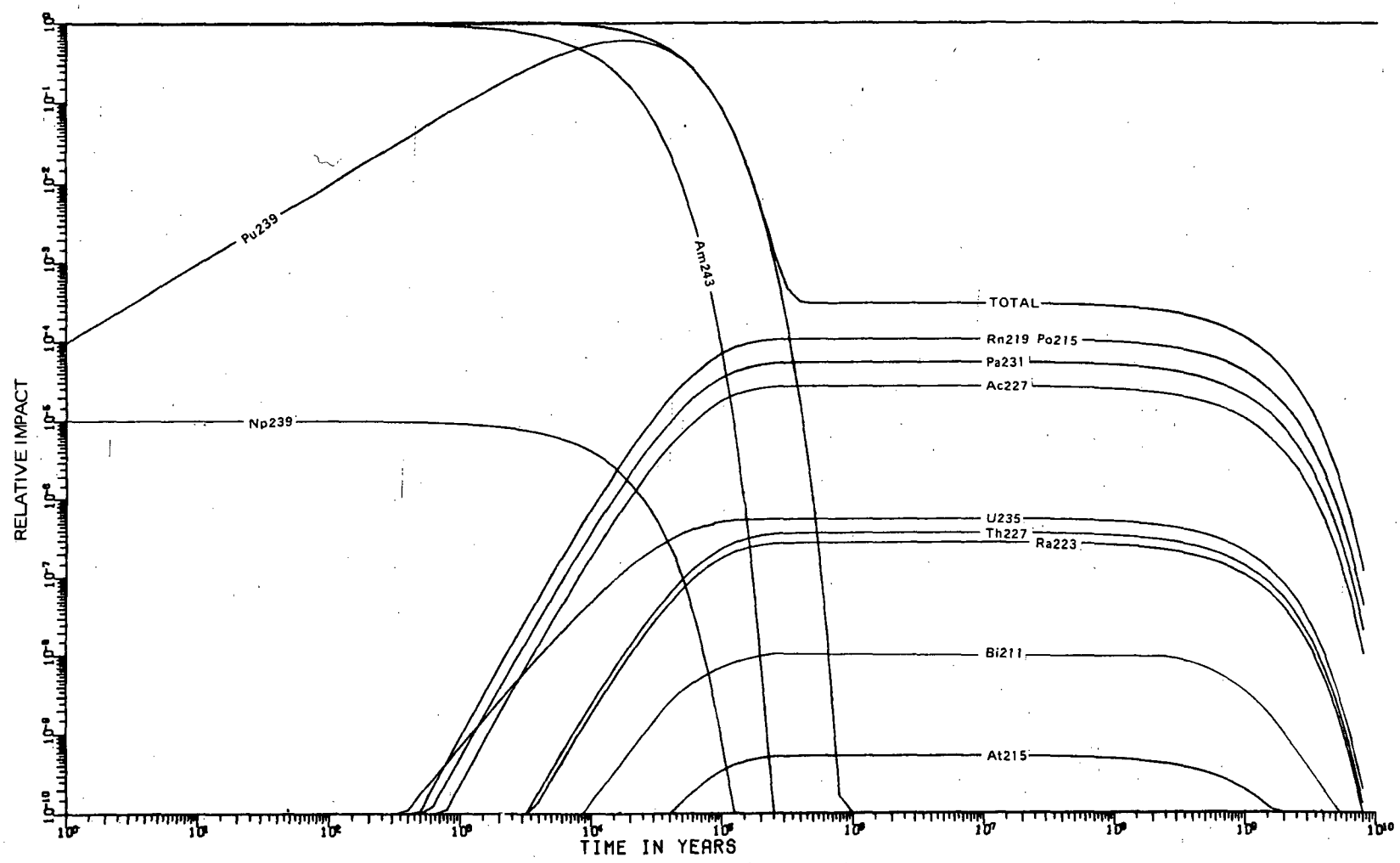

FIGURE 3 RELATIVE IMPACT OF THE MEMBERS OF THE ^{243}Am DECAY CHAIN

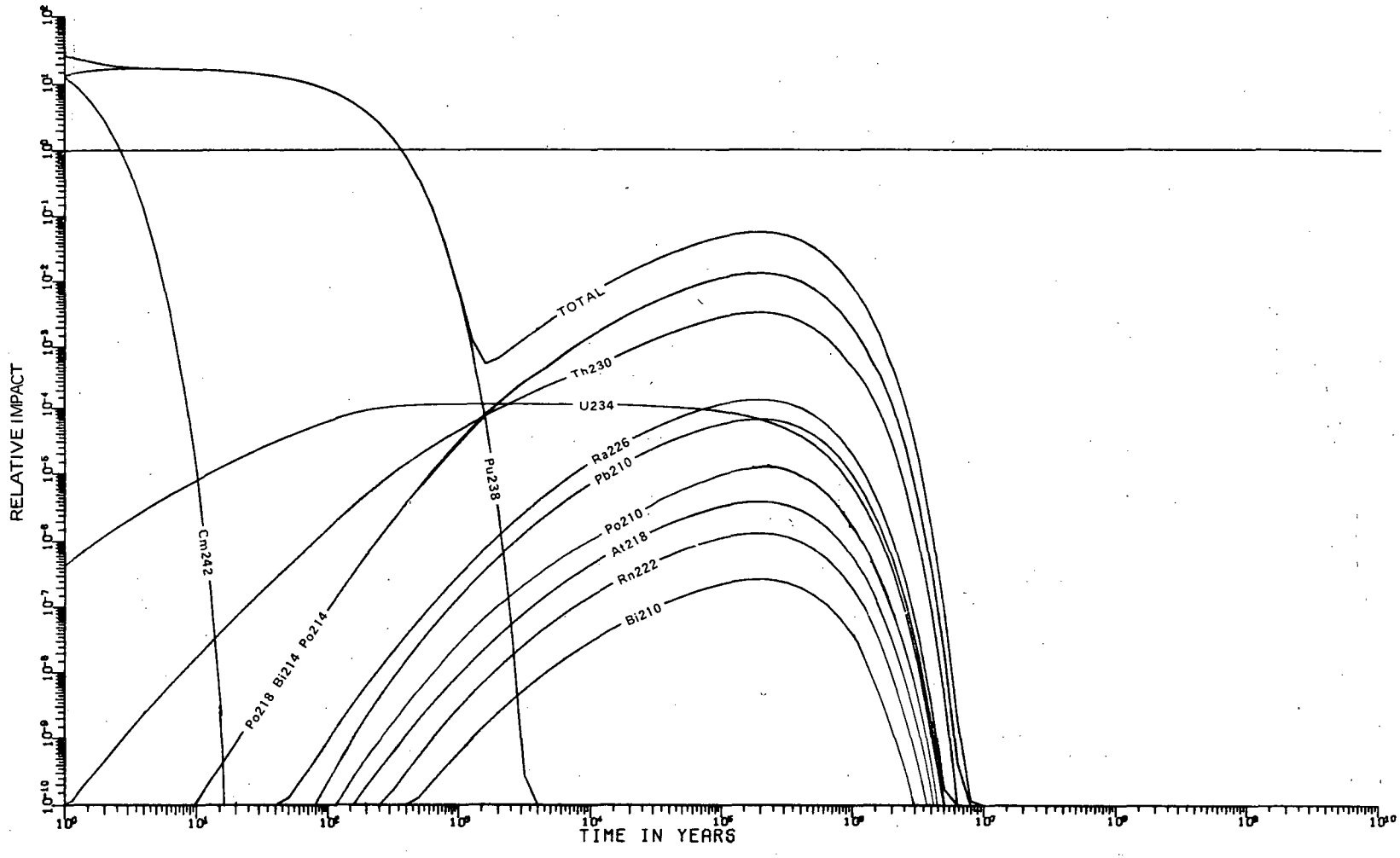

FIGURE 5 RELATIVE IMPACT OF THE MEMBERS OF THE ^{242}Cm DECAY CHAIN

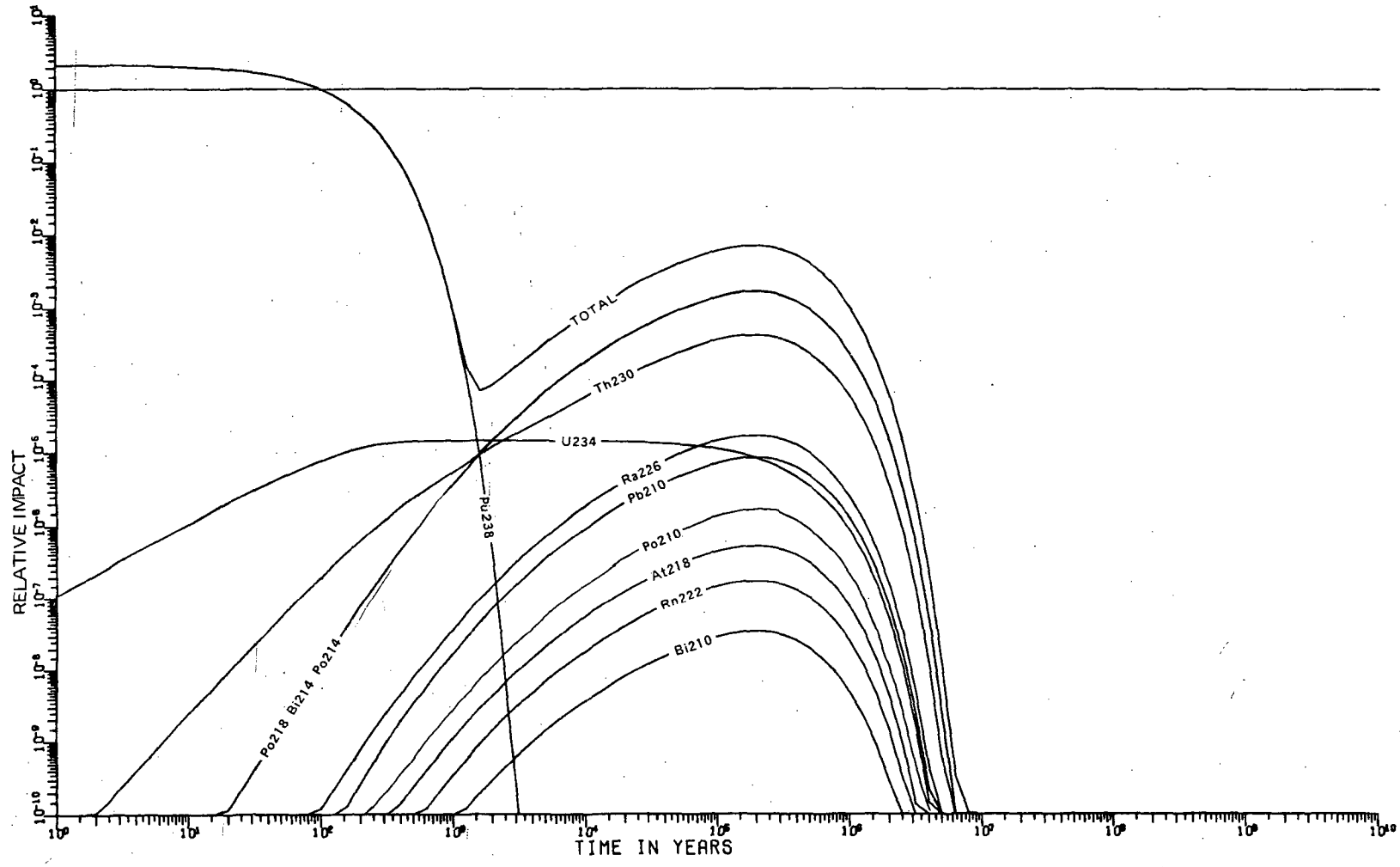

FIGURE 6 RELATIVE IMPACT OF THE MEMBERS OF THE ^{238}Pu DECAY CHAIN

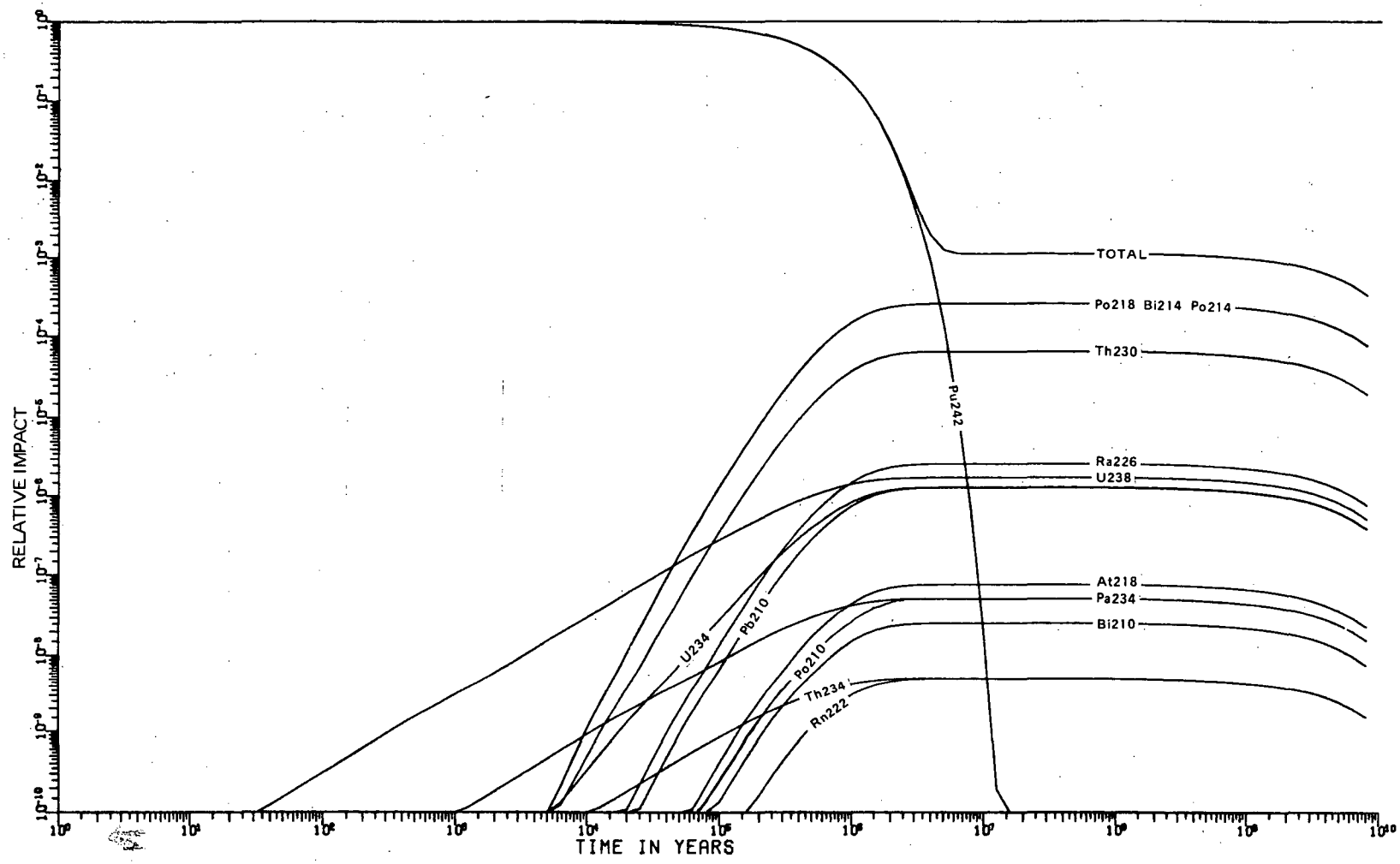

FIGURE 7 RELATIVE IMPACT OF THE MEMBERS OF THE ^{242}Pu DECAY CHAIN

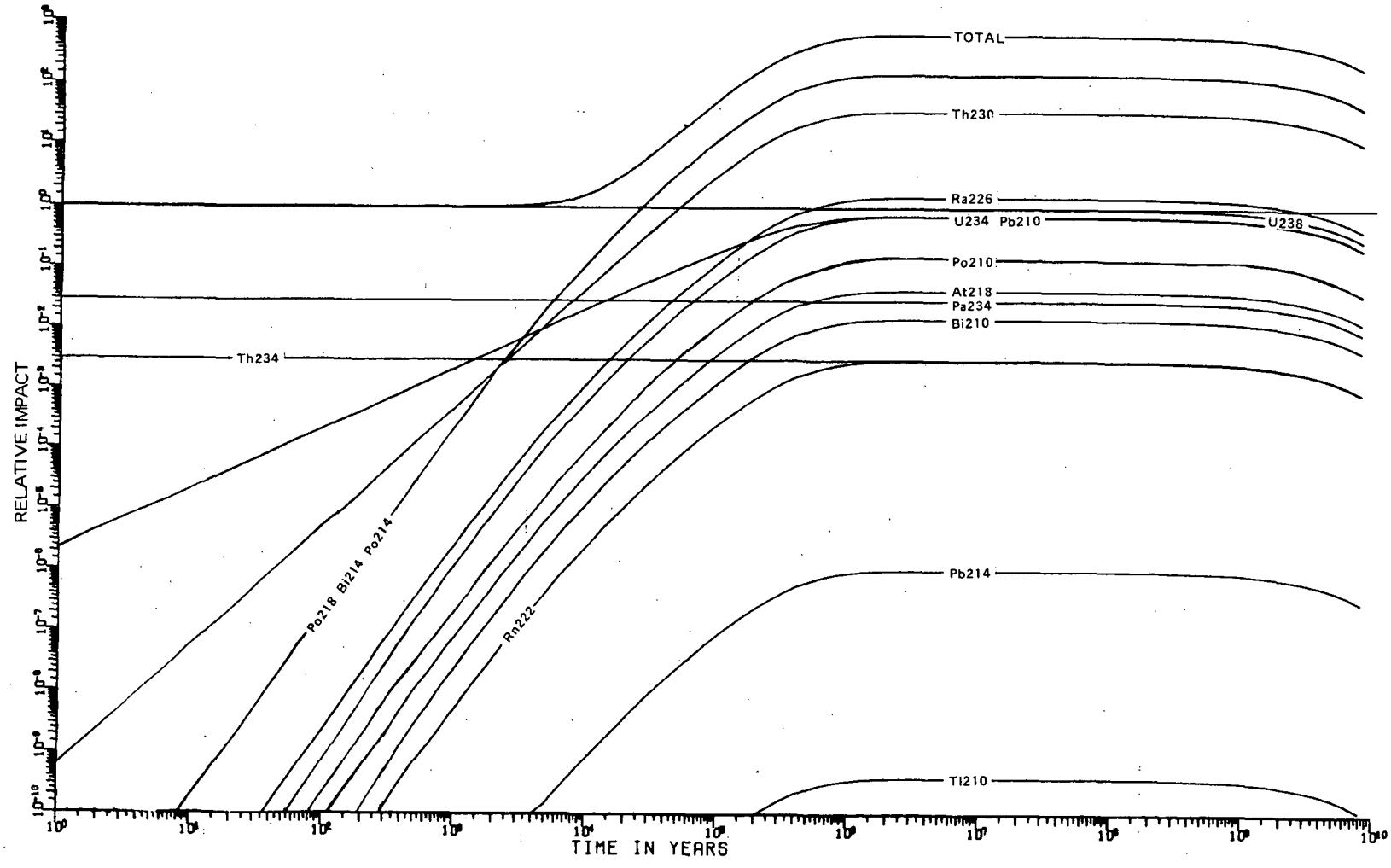

FIGURE 8 RELATIVE IMPACT OF THE MEMBERS OF THE ^{238}U DECAY CHAIN

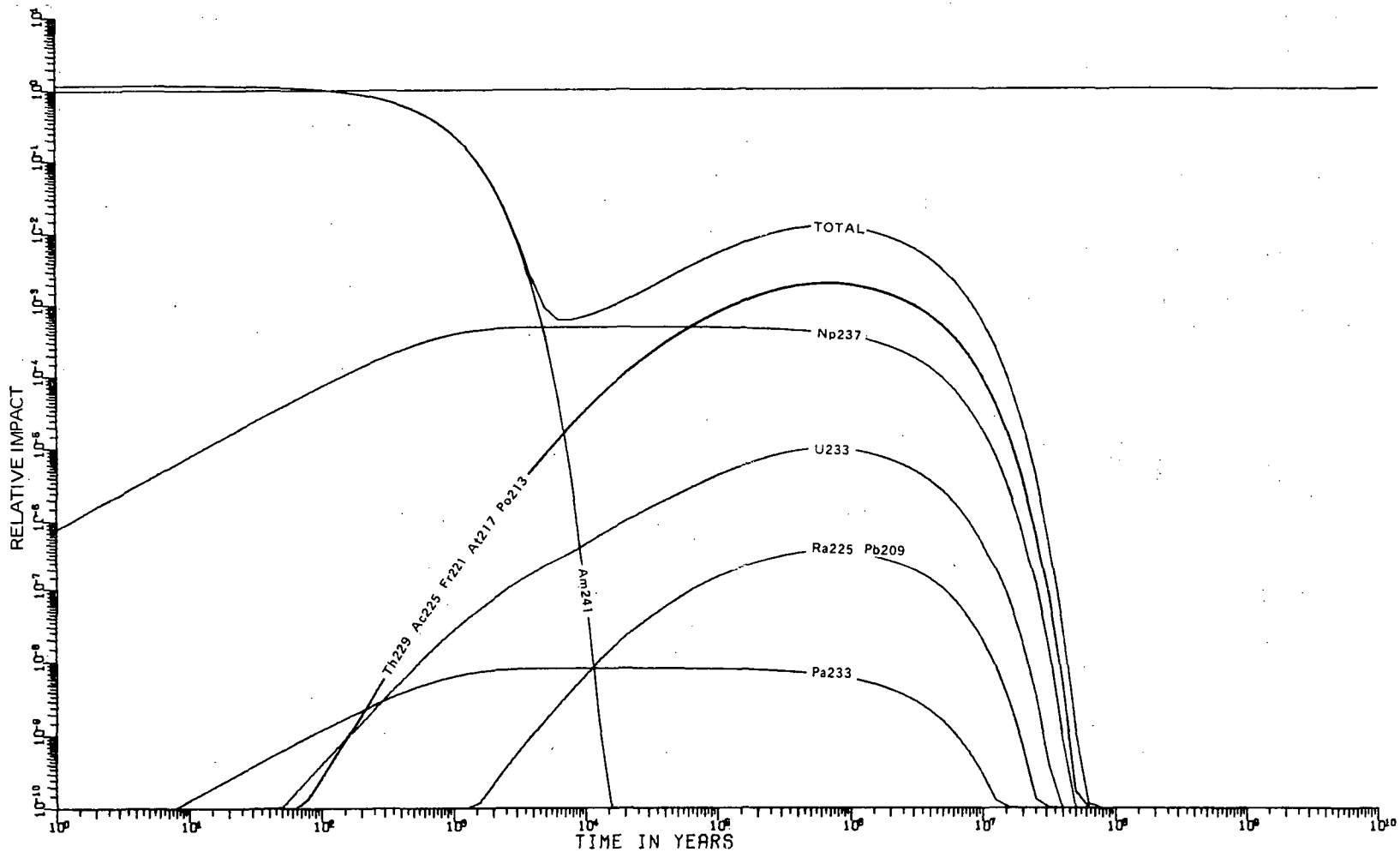

FIGURE 10 RELATIVE IMPACT OF THE MEMBERS OF THE ^{241}Am DECAY CHAIN

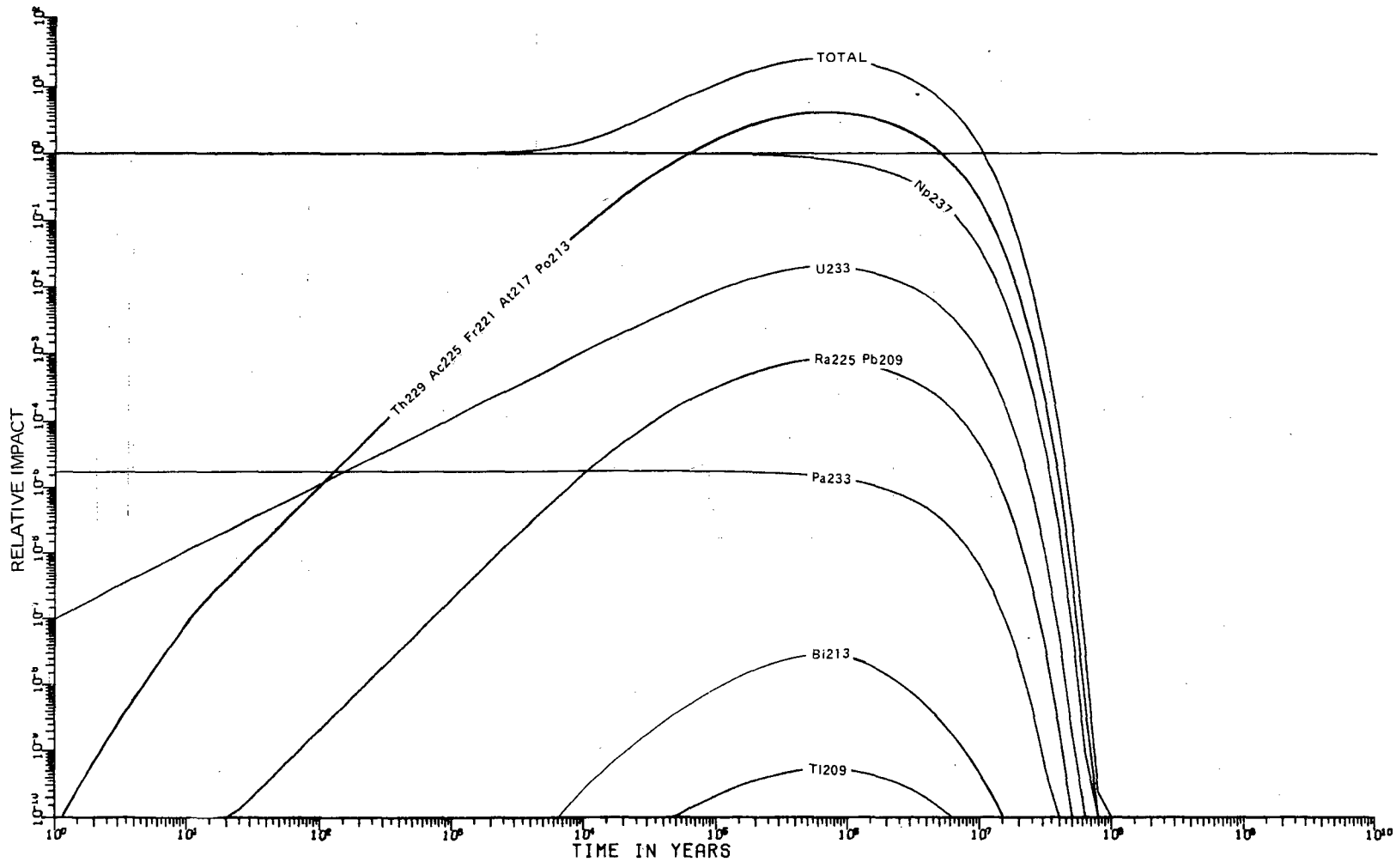

FIGURE 11 RELATIVE IMPACT OF THE MEMBERS OF THE ^{237}Np DECAY CHAIN

APPENDIX D

ENGINEERING EVALUATION OF THE LATTY AVENUE SITE, HAZLEWOOD, MISSOURI (FORMERLY LEASED BY COTTER CORPORATION)

June 1978

PREFACE

This Appendix, which is a detailed evaluation of the Latty Avenue Site, illustrates the application of the RWDCS to a specific site. The purpose was to demonstrate the ability of the RWDCS to bridge the gap between the generic study and a detailed engineering evaluation. Although the value of this effort in developing the RWDCS data base may not be readily apparent, this effort did have a significant impact on the RWDCS. It is also informative to compare the study guidelines with the Remedial Action Criteria (Appendix D.2) which are used in the absence of an accepted RWDCS.

It is also intended that this Appendix will provide a technical basis for any corrective action which may be required at the Latty Avenue Site.

ABSTRACT

Ford, Bacon & Davis Utah Inc. has performed an engineering evaluation of the problems resulting from the previous storage of radioactive uranium processing wastes at 9200 Latty Avenue, Hazelwood, Missouri. The site was formerly leased by Cotter Corporation for processing uranium containing residues prior to shipment to Canon City, Colorado. A radiological survey of the site was performed by Oak Ridge National Laboratory in mid-1977. This evaluation used those survey data, supplemented by FB&DU measurements, to determine areas and volumes of radium- and thorium-contaminated materials (including buildings), the evaluation of resulting radiation exposures of nearby workers and residents, the investigation of site hydrology and meteorology, and the evaluation and costing of alternative corrective actions.

Radon gas release from the 18,300 yd^3 of contaminated material at the Latty Avenue site, constitutes the most significant environmental impact. Limited windblown tailings, external gamma radiation, and localized contamination of surface waters are other environmental factors. The five remedial action alternatives presented range from cleanup and temporary storage of contaminated materials on site ($256,550), the removal of all contaminated materials and two buildings from the site and disposal at an airport storage site ($531,350), disposal in bulk at the Weldon Spring Chemical Plant ($315,350), disposal at a DOE facility such as the Nevada Test Site ($3,384,000), or disposal at a commercial nuclear waste disposal site ($3,964,350).

CONTENTS

Chapter	Title	Page
	Preface	i
	Abstract	ii
	List of Figures	viii
	List of Tables	x
	Glossary	xi
1	SUMMARY	
	1.1 Introduction	1-1
	1.1.1 Background	1-1
	1.1.2 Scope of Engineering Evaluation	1-1
	1.2 Site Description	1-1
	1.2.1 Location and Topography	1-1
	1.2.2 Ownership and History of Processing Operations	1-1
	1.2.3 Present Condition of the Site	1-2
	1.2.4 Soils and Geology	1-2
	1.2.5 Surface and Ground Waters	1-2
	1.2.6 Meteorology	1-3
	1.3 Radioactivity and Pollutant Impact on the Environment	1-3
	1.3.1 Radiation Exposure Pathways, Contamination Mechanisms, and Background Levels	1-3
	1.3.1.1 Radon Gas Measurements and Diffusion	1-4
	1.3.1.2 Radiation Measurements Inside Structures	1-4
	1.3.1.3 Direct Gamma Radiation On Site	1-4
	1.3.1.4 Soil Contamination	1-5
	1.3.1.5 Windblown Contamination	1-5
	1.3.1.6 Surface Water Contamination	1-5
	1.3.2 Remedial Action Criteria	1-5
	1.3.3 Potential Health Impact	1-7
	1.4 Population and Land Use	1-7
	1.4.1 Population and Employment Projections	1-7
	1.4.2 Land Use and Values	1-8

CONTENTS (Cont)

Chapter	Title	Page
	1.5 Stabilization Cover for Low-Level Radioactive Materials	1-8
	1.6 Long-Term Storage (Disposal) Site Selection	1-9
	1.7 Remedial Action Alternatives	1-9
	Chapter 1 References	1-13
2	SITE DESCRIPTION	
	2.1 Location	2-1
	2.2 Topography	2-1
	2.3 Ownership	2-1
	2.4 History of Processing Operations at the Site	2-1
	2.5 Present Physical Condition of the Site	2-3
	2.6 Soils	2-6
	2.7 Geology	2-6
	2.8 Surface Waters	2-7
	2.9 Ground Water	2-8
	2.10 Meteorology	2-8
	Chapter 2 References	2-20
3	RADIOACTIVITY AND POLLUTANT IMPACT ON THE ENVIRONMENT	
	3.1 Radioactive Material Characteristics	3-1
	3.2 Radiation Effects	3-2
	3.3 Natural Background Radiation	3-3
	3.4 Radiation Exposure Pathways and Contamination Mechanisms	3-3
	3.4.1 Radon Gas Measurements and Diffusion	3-4
	3.4.2 Radiation Measurements Inside Structures	3-6
	3.4.3 Direct Gamma Radiation On Site	3-6
	3.4.4 Soil Contamination	3-7
	3.4.5 Windblown Contamination	3-7
	3.4.6 Surface Water Contamination	3-8
	3.5 Remedial Action Criteria	3-8
	3.6 Potential Health Impact	3-9
	3.6.1 Assumptions and Uncertainties in Estimating Health Effects	3-11
	3.6.2 Health Effects	3-12

CONTENTS (Cont)

Chapter	Title	Page
	Chapter 3 References	3-42
4	POPULATION AND LAND USE	
	4.1 Recent History of the Area	4-1
	4.2 Population and Employment Projections . . .	4-1
	4.3 Land Use	4-2
	4.4 Impact of the Site on Land Values	4-3
	Chapter 4 References	4-8
5	STABILIZATION COVER FOR LOW-LEVEL RADIOACTIVE MATERIALS	
	5.1 Prevention of Wind and Water Erosion . . .	5-1
	5.1.1 Chemical Stabilization of the Surface	5-1
	5.1.2 Complete Chemical Stabilization . .	5-2
	5.1.3 Physical Stabilization	5-3
	5.1.4 Vegetative Stabilization	5-4
	5.2 Prevention of Leaching	5-4
	5.3 Reduction of Radon Exhalation	5-5
	5.4 Reduction of Gamma Radiation	5-5
	5.5 Assessment of Applicability	5-5
	Chapter 5 References	5-8
6	LONG-TERM STORAGE (DISPOSAL) SITE SELECTION	
	6.1 Airport Fill Site (Alternative II)	6-1
	6.2 Weldon Spring Chemical Plant, Pit No. 4 Disposal Site (Alternative III)	6-2
	6.3 Barnwell, South Carolina Privately Owned Disposal Site (Alternative V)	6-3
	6.4 Removal of the Radioactive Materials to a U.S. Government Storage Facility, Such as at the NTS, Nevada	6-4
	6.5 Sites Considered but not Included as Alternatives	6-4
	6.5.1 Westlake Landfill, St. Louis	6-4
	6.5.2 The "Quarry" Site at Weldon Spring .	6-4
	Chapter 6 References	6-6

CONTENTS (Cont)

Chapter	Title	Page
7	REMEDIAL ACTION ALTERNATIVES	
	7.1 Short-Term Storage Site Developed On Site (Alternative I)	7-1
	7.1.1 Security and Maintenance	7-4
	7.1.2 Resulting Impacts	7-4
	7.1.3 Costs	7-4
	7.2 Decontamination and the Material Removed to the Airport Fill Site (Alternative II) . .	7-5
	7.2.1 Security and Maintenance	7-6
	7.2.2 Resulting Impacts	7-7
	7.2.3 Costs	7-7
	7.3 Decontamination and the Material Removed to Weldon Spring (DOE) Site (Alternative III - A & B)	7-8
	7.3.1 Bulk Shipment and Storage (Alternative III - A)	7-8
	7.3.1.1 Security and Maintenance .	7-8
	7.3.1.2 Resulting Impacts	7-8
	7.3.1.3 Costs	7-9
	7.3.2 Packaged Shipment and Storage, (Alternative III - B)	7-9
	7.3.2.1 Costs	7-10
	7.4 Decontamination and the Material Removed to a Privately Owned Disposal Site: Barnwell, South Carolina (Alternative IV) .	7-11
	7.4.1 Costs	7-11
	7.5 Decontamination and the Material Removed to a U.S. Government Storage Facility (Alternative V)	7-12
	7.5.1 Costs	7-12

CONTENTS (Cont)

Appendix	Title	Page
D.1 (A)	DECONTAMINATION AND COST ANALYSES FOR FOUR BUILDING STRUCTURES AT 9200 LATTY AVENUE, HAZELWOOD, MISSOURI	
	A. Contamination Measurements	A-1
	B. Building Surveys, Decontamination Methods and Cost Analysis	A-2
	C. Summary	A-9
D.2 (B)	REMEDIAL ACTION CRITERIA	
	B.1 Surgeon General's Guidelines	B-1
	B.2 Radiological Criteria for Decontamination of Inactive Uranium Mill Sites	B-6
	B.3 Grand Junction Remedial Action Criteria (10CFR712)	B-9
	B.4 Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material	B-12
D.3 (C)	GEOLOGIC AND HYDROLOGIC CONDITIONS OF RAFFINATE PITS AT WELDON SPRING SITE	

LIST OF FIGURES

Number	Title	Page
2-1	Site Location	2-11
2-2	Site Description and Topography	2-12
2-3	Site Ownership	2-13
2-4	Northeast Inside Corner of Building 1, Taken October 13, 1977	2-14
2-5	Simplified Geologic Cross-Section of the Florissant Basin	2-15
2-6	Simplified Stratigraphic Column St. Louis County, Missouri	2-16
2-7	Structure Map of St. Louis Area, Missouri . .	2-17
2-8	Surface Drainage Patterns	2-18
3-1	Radioactive Decay Chain of Uranium 238	3-14
3-2	Charcoal Canister Measurement of Radon Flux .	3-15
3-3	Radon Concentration in Vicinity of Site . . .	3-16
3-4	Reduction of Outdoor 222Rn Concentration with Distance from the Site	3-17
3-5	External Gamma Radiation in and Around Building No. 1	3-18
3-6	External Gamma Radiation in Building No. 2 . .	3-19
3-7	External Gamma Radiation in and Around Building 3	3-20
3-8	External Gamma Radiation in and Around Building 4	3-21
3-9	Maximum Beta-Gamma Dose Rate and Direct Alpha Readings, Building 1	3-22
3-10	Maximum Beta-Gamma Dose Rate and Direct Alpha Readings, Building 2	3-23
3-11	Maximum Beta-Gamma Dose Rate and Direct Alpha Readings, Building 3	3-24
3-12	Direct Alpha Readings, Building 4	3-25

LIST OF FIGURES (Cont)

Number	Title	Page
3-13	External Gamma Radiation on Site	3-26
3-14	Beta-Gamma Dose Rates on Site	3-27
3-15	Surface Soil Sample Locations	3-28
3-16	Core Hole Locations	3-29
3-17	External Gamma Radiation Off Site	3-30
3-18	Aerial Gamma Radiation Survey	3-31
3-19	Water and Sediment Sample Locations	3-32
3-20	Lung Cancer Risk from Continuous Exposure to Radon Diffusion	3-33
4-1	Political Jurisdictions and Transportation Routes	4-5
4-2	Population and Employment Outlook for Area Within 0.75 Mi of Site	4-6
4-3	Generalized Land Use Map	4-7
5-1	Reduction of Radon Exhalation Flux with Depth of Cover	5-6
5-2	Reduction of Gamma Exposure Rate Resulting From Packed Earth Shielding	5-7
6-1	Weldon Spring Storage Site (Alternative III) .	6-5
7-1	Decontamination Plan	7-14
7-2	Schematic of Airport Disposal Area (Alternative II)	7-15

LIST OF TABLES

Number	Title	Page
1-1	Summary of Remedial Alternatives and Effects	1-11
2-1	Mean Meteorological Data in St. Louis County--1941-1970	2-19
3-1	Notations and Abbreviations Used in Chapter 3	3-34
3-2	Analyses of Surface Soil Samples	3-36
3-3	Analyses of Subsurface Soil Samples	3-38
3-4	Concentrates of ^{210}Pb, ^{226}Ra, and ^{230}Th in Water and Sediment Samples	3-40
3-5	Estimated Health Impact from Latty Avenue Site in an Area 0-0.75 Miles From Site Edge .	3-41
7-1	Contaminated Materials on the Site	7-16
7-2	Cost Summary of Alternatives	7-17

CHAPTER 1

INTRODUCTION AND SUMMARY

1.1 INTRODUCTION

This appendix documents the activities performed by Ford, Bacon and Davis Utah in describing the physical, radiological and land-use characteristics of the site on Latty Avenue in Hazelwood, Missouri in which uranium ore residues and radioactive process wastes were stored. It also contains suggested alternatives and costs for decontaminating the site. Although the purposes for performing this work are many fold, it is a part of Compilation of the Radioactive Waste Disposal Classification System Data Base project and is investigated as a site which was never intended to be a disposal facility.

This appendix contains the specific site information used in the analysis.

1.2 SITE DESCRIPTION

1.2.1 Location and Topography

The Latty Avenue site is an 11-acre tract located within the city of Hazlewood in north St. Louis County, Missouri The site is at 9200 Latty Avenue and its relationship to the surrounding area is shown in Figure 2-1, Chapter 2. The site is located in low rolling hill terrain at an altitude of approximately 520 ft above sea level.

1.2.2 Ownership and History of Processing Operations

The site is in two ownerships. The westerly 3.498 acres are owned by Mr. Dean Jarboe of St. Louis, and the easterly $\pm$ 7.5 acres are owned by the Norfolk and Western Railroad. Beginning in 1966, the Latty Avenue site was used by the Continental Mining and Milling Company of Chicago as a process drying facility for ore residues and uranium- and radium-bearing processed wastes. These wastes had been stored at the St. Louis Airport fill site, shown in Figure 2-1, Chapter 2, and were generated by the Mallinckrodt Chemical Works at St. Louis during the period 1942 through the late 1950's. Much of the material at the Latty Avenue site was shipped to the Cotter Corporation facilities in Colorado by the Commercial Discount Corporation of Chicago, which had purchased the remaining source material; approximately a year later, it shipped more material to its Colorado facility.

In late 1973, some 8,700 tons of leached barium sulfate and an estimated 39,000 tons of contaminated soil were removed from the Latty Avenue site in a decontamination effort. This material was hauled to the Westlake landfill in St. Louis County and buried under 3 ft of soil. Analyses of soil samples taken during an NRC investigation of the site in 1976 indicated the presence of ura-

nium- and thorium-bearing residues, and direct readings of radiation on the site exceeded criteria established by the NRC for release of the site for unrestricted use. Late in the summer of 1977 the Health and Safety Research Division of the Oak Ridge National Laboratory made a radiological survey of the site.

1.2.3 Present Condition of the Site

The eastern portion of the site (7.5 acres) is vacant. The western ± 3.5-acre portion has 4 structures, along with a railroad spur. Most of the site is weed-covered and there is some debris on the site. The site is only partly fenced. Two of the buildings are steel-framed, with ribbed-steel siding and roofing. The combined areas of the two buildings total 15,000 ft^2. They have eave heights of 25 ft and 30 ft and are rusted, with doors, floors, and partitions missing. The walls and roofs have been punctured by vandals and many of the structural steel columns are bent. The third structure is a "Butler" type metal building of 1,176 ft^2 with a concrete floor, and is in good condition. The other structure is a 1,120-ft^2 one-story masonry office building which was extensively damaged by a fire in 1974, but has been partially reconstructed. There is some paving, with curbes and landscaping, on the west portion of the site; the paving services the office and other buildings. The structures are vacant and are not being used for any purpose at present.

1.2.4 Soils and Geology

The Latty Avenue site is located in the Florissant Basin, a shallow oval-shaped depression. The soils in the area of the site consist of a veneer of alluvium, a few feet of dark silt loam, up to 10 ft of silty fine sands and clays (i.e. loess) of glacial origin underlain by 35 to 85 ft of blue and gray clays, silts, and some sands of glacial lacustrine origin. The bedrock under the site is Mississippian age limestone, which is a unit in a sequence of almost horizontal Ordivician, Mississippian, and Pennsylvanian sedimentary formations.

1.2.5 Surface and Ground Waters

The Latty Avenue site is on the floodplain of Coldwater Creek, a tributary of the Missouri River. There is an abundance of surface water on or near the site, including ponded water during and after precipitation, a sewer system, septic tank(s), a waterline, several ditches, a storm drainage ditch, and a tributary to Coldwater Creek. There is very little flow of off-site waters onto the site. Water does flow off-site to the north into a ditch and to the south into a stream, both of which drain into Coldwater Creek.

Ground waters in the area are contained largely in unconsolidated deposits and to a lesser extent in bedrock. The many silt and clay layers act to slow movement of the waters and provide ample opportunity for natural filtration and purification. Consequently, there is very little chance of contamination of ground waters from on-site existing contamination. Surface waters are, however, the primary source of water usage in the St. Louis area.

1.2.6 Meteorology

Normal annual precipitation for the St. Louis area for the period 1941-1970 has been a little over 35 in. Variations rarely exceed 15 in. above or below this amount. Snowfall has averaged less than 20 in./yr since 1930, ranging from less than 1 in. in 1931-32 to 42.4 in. in 1973-74. Winters are brisk, but seldom severe. Temperatures drop to 0° F or below only 2 or 3 days a year and are 32° F or lower less than 20 to 25 days a year. Maximum temperatures of 90° F or higher occur on an average of 35 to 40 days per year. Winds are predominently from the south with a mean speed of 9.5 mi/hr. The next most prominent direction is from the northwest, which occurs mainly during the winter months.

1.3 RADIOACTIVITY AND POLLUTANT IMPACTS ON THE ENVIRONMENT

About 85% of the total radioactivity originally in uranium ore remains in the processing wastes after removal of the uranium because the radium and thorium, principal contributors to radioactive emissions, were not normally removed from the uranium ores during milling. The principal environmental radiological impact and associated health effects arise from the ^{230}Th, ^{226}Ra, ^{222}Rn, and ^{222}Rn daughters contained in the waste materials. Other isotopes of uranium and thorium and their daughter products may also be present depending upon the type of ore present. Although these radionuclides occur in nature, their concentrations in tailings material are several orders of magnitude greater than their average concentrations in the earth's crust.

1.3.1 Radiation Exposure Pathways, Contamination Mechanisms, and Background Levels

The major potential environmental routes of exposure to man are:

(a) Inhalation of ^{222}Rn and its daughter products resulting from the continuous radioactive decay of ^{226}Ra in the radioactive materials. Radon is a gas which diffuses from the site. The principal exposure results from inhalation of the ^{222}Rn and Rn daughters. This exposure affects the lungs. For this assessment, no criteria have been established for radon concentrations in air. However, the pathway for radon and radon daughters accounts for the major portion of the exposure to the population.

(b) External whole-body gamma exposure directly from radionuclides in the piles.

(c) Inhalation and ingestion of windblown materials. The primary health effect relates to the alpha emitters ^{230}Th and ^{226}Ra, each of which causes exposure to the bones and lungs.

(d) Ingestion of ground and surface water contaminated with radioactive elements (primarily ^{226}Ra) and other toxic materials.

(e) Contamination of food through uptake and concentration of radioactive elements by plants and animals is another pathway which can occur; however, this pathway was not considered further in this study.

1.3.1.1 Radon Gas Measurements and Diffusion

Radon gas concentration measurements at three background locations averaged 0.8 ± 0.2 pCi/l. The ^{222}Rn measurements were made with continuous radon monitors for 24 hr each. One measurement on the site and two measurements at 0.1 and 0.2 mi from the site were all within the background range of 0.6 to 1.1 pCi/l.

Radon flux measurements performed with charcoal canisters at four locations on the site ranged from 2.3 pCi/m^2-s (about twice background) to 55 pCi/m^2-s.

In Buildings 1, 2, and 4, average ^{222}Rn concentrations were greater than the 3 pCi/l concentration limit in 10CFR712 during the 24-hr measurement periods.[1] It is probable that the concentration of ^{219}Rn (actinon) produces virtually all of the alpha activity in Building 1 and most of the activity in Buildings 2 and 3.[1]

1.3.1.2 Radiation Measurements Inside Structures

Gamma radiation rate measurements were made in the four buildings at 1 m above the ground or floors at grid intersections. Beta-gamma measurements and alpha disintegration rates were performed at grid locations in the buildings at 1 cm above the surfaces of the floor and lower walls.[1]

Decontamination tests were also performed on selected surfaces inside the buildings to determine how much of the radioactivity could be removed. Comparison of these test results with the NRC guideline for release limits of the various isotopes present on the site indicated that Buildings 1 and 2 require extensive decontamination before they could be used. Portions of Building 3 require decontamination, and Building 4 has contamination mostly on the floors that can be removed readily.

1.3.1.3 Direct Gamma Radiation On Site

External gamma radiation rates were measured at grid intersections over the entire site at 1 m above ground.[1] The highest measurement was 500 μR/hr at the northern end of the site.

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[1] See end of chapter for references.

Other areas of high radiation rates (300-400 μR/hr) were found around Buildings 1 and 2. The gamma radiation rates at four background locations were from 7 to 9 μR/hr.

Beta-gamma rates were also measured at the grid locations 1 cm above ground. At several locations, the radiation rates reached 1 mR/hr.[1]

1.3.1.4 Soil Contamination

Surface and subsurface soil samples were taken for analyses of radioactive contamination on the site and from the dirt floors of Buildings 1 and 2. It was found that ^{238}U, ^{235}U, and ^{232}Th and their daughter products were present on the site.[1]

Surface contamination is present over a large portion of the site ranging up to 2700 pCi/g of ^{226}Ra. In isolated areas, the contamination penetrated to depths of 2 to 3 ft before reaching background concentration levels. The average concentration of ^{226}Ra in four background soil samples was 1.3 pCi/g.[1]

1.3.1.5 Windblown Contamination

Radioactive material was found at several locations off the site, mostly within 20 ft of the site boundary. Wind- or water-eroded radioactive material might have traveled as much as 100 ft from the site in a few directions and remedial action will be required. An aerial survey of the site vicinity was performed in the fall of 1977. No areas of significant contamination were detected off the site.

1.3.1.6 Surface Water Contamination

Water samples from drainage ditches north and south of the site contained concentrations of ^{230}Th, ^{226}Ra, and ^{210}Pb well below maximum permissable concentrations found in 10CFR712, but elevated levels of these isotopes were found in sediments filtered from the water samples. Both ditches drain into Coldwater Creek. A sample from this creek contained only background levels of these isotopes (less than 1 pCi/l).

1.3.2 Remedial Action Criteria

Radiological criteria established for the Phase II - Title I Engineering Assessment of Inactive Uranium Mill Tailings Sites could also be utilized for this engineering evaluation. The criteria of the Phase II program are divided into two general categories:

(a) Criteria applicable to structures with tailings underneath them or within 10 ft

(b) Criteria pertaining to the mill tailings site and open land

The criteria utilized for habitable structures are the guidelines published by the Surgeon General of the United States for use in the Grand Junction, Colorado, remedial program. These guidelines recommend graded levels for remedial action in terms of the external gamma radiation (EGR) levels and of the indoor radon daughter concentration (RDC) levels above background found within dwellings constructed on or near uranium mill tailings. (In this usage, the word "external" refers to gamma radiation from sources outside the human body to which an individual may be exposed.)

The recommended graded levels are as follows:

EGR	RDC[a]	Recommendations
Greater than 0.1 mR/hr[b]	Greater than 0.05 WL[c]	Remedial action indicated
From 0.05 to 0.1 mR/hr	From 0.01 to 0.05 WL	Remedial action may be suggested
Less than 0.05 mR/hr	Less than 0.01 WL	No remedial action indicated

[a]Based upon yearly average values from 6 air samples of at least 100-hr duration taken at a minimum of 4-wk intervals throughout the year

[b]mR/hr = milliroentgen per hour, a measure of gamma radiation, 1 mR/hr = 1,000 μR/hr

[c]WL = working level, a measure of alpha radiation from short-lived radon daughter elements

The criteria for land decontamination have the objective of reducing residual gamma radiation to levels which are as low as practicable. However, topographic and economic considerations frequently preclude complete decontamination. A provisional maximum of 40 μR/hr above background is used in such circumstances. Background in the St. Louis area was determined in this study to be 7 to 9 μR/hr. As a guideline for the land beyond the site, if residual gamma levels are less than 10 μR/hr above background, the land may be released for unrestricted use. Where cleanup is necessary the radium content of the soil should be reduced to no more than twice the radium background in the area. If the radioactive tailings material is stabilized in place, the same criteria apply but control of gamma radiation would be by an earth covering. However, the area should be designated a controlled area, be fenced to limit access, and be restricted as to human occupancy. The numerical guidelines provide a basis for the engineering evaluation, but are subject to review based on the overall findings of the evaluation.

The NRC has published guidelines for decontamination of facilities and equipment for unrestricted use. Since ^{226}Ra,

^{230}Th, and ^{227}Ac were found on site, the most restrictive guidelines for acceptable surface contamination levels apply to this site. These guidelines allow a maximum of 100 dpm/100 cm^2 average activity and a limit for removable contamination of 20 dpm/100 cm^2.

Complete descriptions of remedial action criteria and decontamination guidelines are included in Appendix B.

The radium and gross alpha content of ground and surface water should meet applicable state and federal standards.

1.3.3 Potential Health Impact

Radon gas exhalation from the piles and the subsequent inhalation of radon daughters account for most of the total dose to employees and residents near the Latty Avenue site under present conditions. The gamma radiation exposure from the site is essentially zero since only a few workers would approach the area within 200 ft of the site where gamma radiation is above background.

Health effects, in this case lung cancer, were calculated as described in Paragraph 3.6, Chapter 3, using the absolute risk estimator from the National Academy of Sciences Advisory Committee on the Biological Effects of Ionizing Radiation (BEIR report)[2]. The estimated health impact within 0.75 mi from the site is approximately 1% of the health effects from background radon daughter concentration. The 25-yr cumulative health effect is far less than 1 health effect (0.05) for the total number of inhabitants and employees in the area within 0.75 mi of the site.

Radon daughter concentrations in three of the buildings on the site were above allowable limits for unrestricted use and health effects were not calculated for on-site employees.

1.4 POPULATION AND LAND USE

The Latty Avenue site is in Hazelwood, Missouri. Large estates in Hazelwood of the 1800's were subdivided in the early 1900's, and subsequently have become commercial and industrial centers. The area is still slowly becoming more industrialized. There are large residential areas approximately 1 mi east and north of the site.

1.4.1 Population and Employment Projections

The area within 0.75 mi of the site is largely industrial and commercial with a few residential neighborhoods. The residential population was estimated in 1970 to be 660 people living in 60 single-family houses and 25 apartment buildings. Between 0.75 mi and 1 mi from the site there are several high-density residential areas containing approximately 690 dwelling units housing around 2,300 people. Since 1970, residential population within this area has changed very little.

Employment figures depend largely on how far boundaries are extended and which companies are included. There are about 22,000 employees within a 0.75-mi distance of the site. If boundaries are extended to 1 mi from the site, McDonnell Douglas which employs 25,000 people is included.

Planning assumptions for the area include little development of residential areas within the 0.75-mi boundary. It, therefore, has been projected that within the next 25 yr, the residential population will increase by no more than 165 and shrink by no more than 70, making the population somewhere between 590 and 825 for the next 25 yr.

Employment is harder to project since even one of the large industrial complexes in the area can have a large effect one way or the other. Assuming complete use of all available lands at a density of 30 people/acre, the total workers within 0.75 mi of the site could reach 34,000. For a lower limit, a depression affecting 10% of the workers in the area was assumed, reducing the worker count to 20,000.

1.4.2 Land Use and Values

Most of the land near the Latty Avenue site is zoned industrial/commercial. Although the site is in the city of Hazelwood, the property across the street and immediately east of the site is in the city of Berkeley. Seventy-five percent of the land within 0.75 mi of the site is being used as industrial/commercial, 13% as transportation corridors, 10% is vacant, and the remaining 2% is residential or recreational.

Vacant land in the area has a market value of from \$0.75 to \$1.00/ft^2. The ± 3.5-acre west portion of the site was sold in July 1976 for \$0.492/ft^2. The presence of radioactive materials on the site has had no bearing on the demand for, nor the market value of, surrounding properties.

1.5 STABILIZATION COVER FOR LOW-LEVEL RADIOACTIVE MATERIALS

Investigation of present practices and technology of uranium mill tailings stabilization indicates that much research and development remains to be performed before complete and permanent stabilization of radioactive mill waste materials can be realized.

Reasonably effective means of wind and water erosion control are available, although they involve continual maintenance costs. Lining of containment areas or chemical solidification of wastes are possible methods for control of leaching.

Until recently, no attempt had been made to contain radon in a tailings pile. NRC performance objectives for tailings management after conclusion of mill operations include reduction of radon exhalation from the tailings to twice background radon

ıalation in the vicinity. Uranium well operators have proposed ıpacted clay pond liners and clay caps on the tailings plus 'eral feet of cover material to meet this objective. Research ›grams are under way to determine the effectiveness of various 'er materials in reducing radon flux from tailings.

One of the remedial action alternatives for disposal of the :ty Avenue radioactive materials includes construction of a ;posal site with a clay liner and a clay cap.

6 LONG-TERM STORAGE (DISPOSAL) SITE SELECTION

In all of the alternatives, it is proposed that the contami- ted materials be placed in disposal areas. One area is an terim disposal location on the Latty Avenue site, another is a ng-term storage pit at the St. Louis Lambert-International rport fill site, which would be especially constructed to meet iteria established for the containment of radioactive materials.

In one alternative, the material would be deposited in pit at the DOE facility at Weldon Spring, Missouri, 23 mi west of ıe Latty Avenue site. Disposal here is proposed either in .lk or in packaged containers (55-gal. drums).

The privately owned and NRC-licensed disposal site at Barnwell, uth Carolina is another alternative disposal site. The owners this facility would take responsibility for the material after has been loaded at Latty Avenue, and take care of hauling and sposal.

The last alternative would be to haul the materials to a S. Government (DOE) facility such as the one at the Nevada Test te (NTS) northwest of Las Vegas, Nevada. The material would be ipped to as near the site as possible in railroad cars, then be ansferred into trucks and hauled to the disposal area.

7 REMEDIAL ACTION ALTERNATIVES

In all of the alternatives, the Latty Avenue site would be ;contaminated, two of the buildings would be demolished and ;sposed of along with the contaminated material, and two of the ıildings would be decontaminated and remain on the site. The vo structures to be demolished are large steel-framed buildings. : would be less expensive to tear these down and replace them ı kind rather than to leave them and decontaminate them. The ›posite was true with the two smaller structures. All of the ›ntaminated soil, building rubble, debris, etc., that are now . the site would be gathered. In the first alternative, these terials would be put into a pit designed to provide short-term orage on the site. The advantages of this alternative are at its cost is only $457,000, and that it could be accomplished ickly. (See Table 1-1 for a cost summary.) Disadvantages are ıt some of the site (about 14%) would be used for the storage

pit, and that the contaminated material would need to be relocated to a permanent disposal location at some later time.

In Alternative II the contaminated material would be relocated to the St. Louis Lambert-International Airport fill site, 2.5 road miles from the Latty Avenue site. Here, a specially designed and constructed disposal pit would be constructed for the material so as to provide and meet long-term storage criteria as established by the NRC. The estimated cost is $730,000. Advantages are that the Latty Avenue site would be completely free of contaminated material, and that the material would be placed in an already contaminated location. Disadvantages are that it would require the establishment of a perpetual care fund, and that it would be in an area of heavy industrial use.

In Alternative III the materials would be relocated to the DOE storage facility (pit 4) at Weldon Spring Army Ordinance site; and in Alternative II-A the materials would be hauled and stored in bulk, and in Alternative III-B the materials would be hauled and stored in 55-gal. drum containers.

Advantages of III-A would be that disposal would be in an ideal, already established location, isolated from the populace with no additional maintenance or monitoring required. Disadvantages are the increased exhalation of radon from pit 4 (assuming no stabilization cover), and the possibility of relocating all of the materials at the Weldon Spring site at some subsequent time. Cost of III-A is $513,000. The estimated cost of III-B is $1,867,000, with the advantage of having the contaminated material in metal containers that easily could be relocated. Also, the radon exhalation and gamma radiation would be controlled. Disadvantages are the costs and the relatively short-term life of the containers.

In Alternative IV the materials would be disposed of in a privately owned facility in Barnwell, South Carolina at a cost of $4,131,000. Advantages would be that NRC criteria for disposing of radioactive materials would be met from the time of leaving the site and the material no longer would be a governmental concern. The disadvantage is the cost for hauling and disposal.

In Alternative V the materials would be shipped by rail to a DOE facility such as the Nevada Test Site northwest of Las Vegas. The estimated cost is $3,552,000. Advantages are that all of the NRC objectives for disposal would be met and that no new areas for disposal of such material would be created. Disadvantages are the costs and the unknown maintenance costs.

TABLE 1-1

SUMMARY OF REMEDIAL ALTERNATIVES AND EFFECTS

Alternative Number	Cost ($000)	Description	Advantages	Disadvantages
I	457	On- and off-site decontamination of earth surfaces, removal for storage of Bldgs 1 & 2, decontamination of Bldgs 3 & 4, interim storage of all contaminated materials in a 1.5-acre pit constructed on the site.	A,C,D	U,W,X
II	730	Same as I, except contaminated materials all removed from the site and stored in a specially built long-term storage pit on the St. Louis Airport fill site.	B,C,F,H	V,W,X
III-A	513	Same as II, except the material would be hauled in bulk by truck and stored in Pit No. 4 at the DOE facility at Weldon Spring Chemical Plant site.	B,C,D,F,H	V,X
III-B	1,867	Same as III-A, except material packaged, shipped, and stored in metal containers (55-gal drums).	B,E,F,H	V,X,Y,Z
IV	4,131	Same as II, except material hauled to commercial storage site at Barnwell, South Carolina.	B,F,G,H,I	Y,Z
V	3,552	Same as II, except that material hauled to government storage site at DOE Nevada Test Site.	B,F,G,H,I	Y,Z

TABLE 1-1 (Cont)

Notes

1. All costs are in 1978 dollar value and cost projections.
2. For Alternative III-A no stabilization cover is provided for the 1.5 acres utilized. If cover is desired, it is estimated to cost $10,000/ft.

Definition of Advantages

A. Buildings and most of Latty Avenue site made usable

B. Buildings and all of Latty Avenue site made usable

C. Speed in which work could be accomplished

D. Relatively inexpensive

E. Material packaged for easy storage or re-shipment

F. No more land would be contaminated as being used for a storage facility

G. No more re-handling necessary--permanent storage

H. No additional maintenance or monitoring programs need be established

I. All NRC objectives for storage are met

Definition of Disadvantages

U. Material as stored does not meet NRC storage objectives

V. Material as stored meets only some of NRC storage objectives

W. Monitoring and maintenance program would be required

X. Material may need to be relocated at some subsequent date

Y. Cost

Z. Time of cleanup is more than other alternatives

CHAPTER 1 REFERENCES

1. F. F. Haywood, et al; "Interim Report of Radiological Survey of the Property at 9200 Latty Avenue, Hazelwood, Missouri"; ORNL; Oak Ridge, Tennessee; Sep 1977.

2. "The Effects on Populations of Exposure to Low Levels of Ionizing Radiation"; Report Advisory Committee on Biological Effects of Ionizing Radiation; NAS, National Research Council; Nov 1972.

CHAPTER 2

SITE DESCRIPTION

The purpose of this chapter is to describe the Latty Avenue site at Hazelwood, Missouri, the characteristics of the contaminated soils and structures present on the site, and the local geology, hydrology and meteorology.

2.1 LOCATION

The site is located in north St. Louis County within the corporate limits of the city of Hazelwood, Missouri. It is on the south side, and at the western end of Latty Avenue, with a street address of 9200. The site is 2 mi (in a straight line) north and slightly east of the control tower of the Lambert-St. Louis International Airport. The site is part of Lots 11 and 12 of Hazelwood Farm, a subdivision in U.S. Surveys 1 and 2, Township 47 North, Range 6 East, St. Louis County, Missouri. (See Figure 2-1.)

2.2 TOPOGRAPHY

The total site, comprising approximately 11 acres, is in low rolling hill terrain at approximately 520 ft above sea level. It is in the drainage basin of Coldwater Creek, which discharges some 12 mi downstream into the Missouri River. The variation in elevation on the gently rolling site is approximately 10 ft. The site is separated from Coldwater Creek by Right-of-Way lands of the Norfolk and Western Railroad Company. Figure 2-2 is a descriptive map of the site vicinity and includes topography of the site.

2.3 OWNERSHIP

The site presently is under two ownerships. The westerly 3.498 acres are owned by Mr. Dean Jarboe of St. Louis; he purchased the property in June 1977 from the Bayliss Company, which in July of 1976 purchased it from Associate Commercial Corporation, formerly Commercial Discount Corporation. The remainder of the site (approximately 7.5 acres) is owned by the Norfolk and Western Railroad. (See Figure 2-3.)

2.4 HISTORY OF PROCESSING OPERATIONS AT THE SITE[1]

In early 1966, ore residues and uranium- and radium-bearing processed wastes which had been stored at the St. Louis Airport fill site, shown in Figure 2-1, were moved by the Continental

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[1] See end of chapter for references.

Mining and Milling Company of Chicago, Illinois to the Latty Avenue site. These wastes had been generated by Mallinckrodt Chemical Works of St. Louis during the period 1942 through the late 1950's. The Commercial Discount Corporation of Chicago, Illinois purchased the residues in January 1967; much of the material then was dried and shipped to the Cotter Corporation facilities in Canon City, Colorado. The source material remaining at the Latty Avenue site was sold to the Cotter Corporation in December 1969. Records indicate that residues remaining on the site at that time included 74,000 tons of Belgian Congo pitchblende raffinate containing about 113 tons of uranium, 32,500 tons of Colorado raffinate containing about 48 tons of uranium, and 8,700 tons of leached barium sulfate containing about 7 tons of uranium. During the period August through November 1970, Cotter Corporation dried some of the remaining residues at the site and shipped them to its mill in Canon City, Colorado. By December 1970, an estimated 10,000 tons of Colorado raffinate and 8,700 tons of leached barium sulfate remained at the Latty Avenue site.

In April 1974, an NRC inspector was informed that the remaining Colorado raffinate had been shipped in mid-1973 to Canon City without drying and that the leached barium sulfate had been transported to a landfill area in St. Louis County. A reported 12 to 18 in. of topsoil had been removed with the leached barium sulfate. However, analyses of soil samples taken during an NRC investigation of the Latty Avenue site in 1976 indicated the presence of uranium- and thorium-bearing residues; furthermore, at some points on the site, direct readings of radiation exceeded criteria established by the NRC for decontamination of land areas prior to release for unrestricted use.

On May 30 and June 1, 1976 articles appeared in the St. Louis Post-Dispatch which indicated that there were some discrepancies in the official records as to how much contaminated material from the Latty Avenue site had been deposited where. By letter of June 2, 1976 the Director, Division of Environmental Quality, Missouri Department of Natural Resources, inquired of the NRC about this matter. During the June 22-24 and August 11, 1976 periods, Region III of the NRC investigated the alleged discrepancies and reached the following conclusions[2]:

(a) About 8,700 tons of leached barium sulfate and almost 39,000 tons of soil were removed from the former licensee's Latty Avenue site and buried under 3 ft of other soil at the West Lake Landfill in St. Louis County during the period July-October 1973.

(b) The material present at the West Lake Landfill does not present an immediate radiological health hazard to the public.

(c) It is estimated that the concentration of natural

uranium in the West Lake Landfill could result in concentrations of radon and radon progeny near occupational limits in housing structures if constructed directly on or in the disposed material.

(d) Considering the potential radon and radon progeny build-up in structures plus the direct radiation through basement walls and floors from the buried material, estimated doses warranting avoidance may be hypothesized.

(e) Environmental soil samples indicate the presence of uranium ore process residues at the Latty Avenue site. Beta-gamma surveys performed at that site on August 11, 1976 indicate levels of radiation exceeding the criteria established by the NRC for decontamination of land areas prior to release for unrestricted use.

In the history of the Latty Avenue site as contained in references 1 and 2, there is no mention of the disposal of the equipment in Buildings 1 and 2 nor of where the concrete floors were deposited. There is speculation that some of this material was deposited in what is known as the "airport fill site" located between Brown Road and the railroad line which runs along the north edge of the Lambert-St. Louis International Airport. (See Figure 2-1.)

The West Lake Landfill is located just northwest of the intersection of Taussig Road and St. Charles Road, some 9.5 road miles east of the Latty Avenue site. (See Figure 2-1.)

Late in the summer of 1977 a radiological survey of the Latty Avenue site was made by the Health and Safety Research Division of the Oak Ridge National Laboratory. (See Chapter 3 for a discussion of this effort.)

2.5 PRESENT PHYSICAL CONDITION OF THE SITE

The eastern 7.5-acre area of the site is vacant land and apparently always has been void of any structures. From the Latty Avenue frontage on the north (the lowest elevation area of the site) the land rises towards the south for 2/3 of its length, then slopes to the south and west. On-site inspections have revealed evidence that some of the topsoil, particularly along the Latty Avenue frontage, has been removed from the site. Runoff from storms has left erosion streaks in the soil as it leaves the property. Precipitation drains to the north off the property through an erosion-caused channel in a berm which runs along the northern edge of the site. The precipitation then enters into a drainage ditch which runs west between the north boundary of the site and the south side of Latty Avenue, then crosses beneath Latty Avenue in two 24-in.-diameter concrete culverts, and

eventually empties into Coldwater Creek. This 7.5-acre portion of the site is about 75% covered with thick growths up to 5 ft tall.

The western 3.5-acre area of the site has 4 structures scattered in a north-south direction. Building 1 is a 12,000-ft^2 steel-framed continuous-beam structure with a 30-ft eave height. Siding and roofing are built of 36-in.-wide ribbed steel siding, probably 26 gauge, continuous from floor to eave. The building has metal gutters and downspouts. All of the doors have been removed, as has some of the siding on the southwest and northeast corners. The siding has been penetrated in many locations by vandals, by gunshot holes, and other sharp objects. Near the base (foundation) the siding is bent in many places and very rusty. The steel frame and purlins (which support the siding and roofing) were never painted and are extremely rusted, and in several locations bent or knocked loose from the foundation. (See Figure 2-4.) A layer of 1-in.-thick alluminum-backed insulation was installed between the purlins and the siding and roofing. This material has been torn, pulled and broken in many places. Most of the steel horizontal-projection casement windows remain. The concrete floor that was once a part of the structure has been removed, although large blocks of concrete remain. The reinforcing rod and concrete tie beams which once were an integral part of the structural system have been broken and, for the most part, removed. Under extreme roof loading conditions, such as heavy snows, this "break" in the structural system could mean roof failure; however, because of the center support posts and the lack of any deadweight being supported from the roof, structural failure of the roof appears unlikely. All partitions have been removed.

Building 2 is a 50- x 60-ft (3,000-ft^2) structure located just north of Building 1. It is similar in construction to Building I, except that it has a 25-ft eave height and has a rigid frame construction (no center supports). Its physical condition is similar to that described for Building 1. The major difference is that Building 2 is in much greater danger of failure because of the removal of its in-concrete tie beams. An extreme roof-loading condition brought about by heavy snow could cause the structure to fail.

Both Buildings 1 and 2 were made under the trade name "MES-TEX" which no longer is available in the St. Louis area. Apparently the doors and missing siding from these structures were removed to facilitate the removal of the concrete floors.

Building 3 is northeast of Building 2 and is a 28- x 42-ft (1,176-ft^2) prefabricated "Butler" type building with a 12-ft eave height. The structure has a concrete floor, a full-height sliding door on the north side and regular-size door on the northwest corner. In a few places on the inside walls some 1-in.-thick rigid foam insulation has been installed. The structure is in good physical condition. There is a 9-ft-wide concrete apron

apron in the front (north side) of the building.

Building 4 is a 1,120-ft^2, single-story masonry office building located on the north edge of the site, with paved access from Latty Avenue. In 1974 it was partially damaged by fire, and later was partially re-constructed. The building has a concrete floor with asphalt tile surfacing. Sewage disposal was by means of a septic tank located on its west side. It is not known what systems handled sewage generated from occupants in Buildings 1, 2 and 3, but probably there is a buried septic tank south of Building 1; although this was not verified since no building records could be located for the four structures.

The highest point on this western 3.5-acre portion of the site is the floor of Building 1. From this point the land slopes in all four directions. As with the 7.5-acre east portion of the site, some of the topsoil on the west area has been removed, and the same types of weeds cover the vacant land of the east portion.

A north-south spur of the Norfolk and Western Railroad runs parallel to the western boundary of the site. From this spur, another spur enters the site near its southern boundary and runs parallel to the western boundary inside the west property line. This spur runs alongside the west side of Building 1, then dead-ends near the south edge of Building 4. Weeds have overgrown the track.

There is a small amount of debris on the site, including a pile of assorted junk in the extreme southwest corner, and some junk from the fire which occurred in Building 4 and which is just south of that building. Twenty rolls of 6-in. x 6-in. 10-gauge wire mesh lie adjacent to Building 3. These were to be used by Mr. Jarboe in replacing the concrete floors in Buildings 1 and 2.

Fencing around the site is incomplete. A variety of fence types have been used. A good portion of the fencing is missing, with just metal posts remaining. Chainlink fencing, the predominant material, is in poor condition or on the ground in many locations. There is some fencing which separates the east and west portions of the site.

Along the eastern edge of the property owned by Norfolk and Western is a 10-ft easement given to the Metropolitan St. Louis Sewer District. Within that right-of-way there is a 10-in. vitrified clay sewerline. Two manholes on the site allow access to the sewerline. Grading operations apparently have exposed the northern-most manhole, which now sits some 3 ft avove the ground level. The other manhole surface is flush with the ground.

A high-voltage transmission line with four major crossarms traverses the site near its north boundary in a generally east-west direction. This line passes just north of Building 4.

The entire 11-acre site is within the city limits of Hazelwood. The property immediately east of the site and that across Latty Avenue to the north are in the city limits of Berkeley, Missouri.

2.6 SOILS

The Latty Avenue site is located in a shallow oval-shaped depression known as the Florissant Basin. The slope of the land is generally less than 2% and minor drainages are typically broad and shallow. During glacial times drainage from the area was blocked, and the Florissant Basin was a lake in which over a 100-ft thickness of silts, clays, and sands were deposited. These lacustrine soils have a very high water content and are more compressible than most alluvial soils. The glacial lake bed sediments subsequently were covered by wind-laid glacial material known as "loess". As the waters dried up a rich prairie flora developed across the basin. Since then only a minimum of the rich alluvium has been deposited by waters of Coldwater Creek and its tributaries over certain sections of the basin. A simplified cross-section of the Florissant Basin is shown in Figure 2-5.

The soil materials underlying the site reflect this history. The dominant surface material is a dark silt loam, underlain by loess, which in turn is underlain by lacustrine sediments. The entire sequence is approximately 100 ft thick. The general characteristics of the upper soil types include shallow slopes, seasonally high water table, marginal to slow percolation rates, poor bearing capacity, high compressibility, very high available water capacity, poor surface and internal drainage, high organic matter, high potassium content, and medium phosphorous content. The soils are dark-colored silty loams or silty clay loams.

2.7 GEOLOGY

The unconsolidated materials at the site reflect its relatively recent geologic history of glaciation and post-glacial environments as discussed in paragraph 2.6. Underlying these deposits are limestone strata of Mississippian age. These sedimentary strata are the middle unit of a sequence of Ordivician, Mississippian, and Pennsylvanian strata that outcrop over sections of St. Louis County. Figure 2-6 is a simplified stratigraphic column showing these rock units. These strata are almost flat-lying with a slight dip to the northeast. This northeasterly dip is modified by gentle anticlines, synclines, and by other structural features such as the Florissant Dome northeast of the site. It has been suggested that the major northwest-southeast trending faults in the county still may act as zones of weakness and be subject to earth movement. Figure 2-7 is a map of the area showing regional structural features.

2.8 SURFACE WATERS

There are abundant surface waters in the area: two major rivers, one lesser river, several streams, numerous intermittent drainages, wide flood plains, lakes, impoundments, ditches, and standing water. Surface waters at or near the site consist of: ponded water on the site during and for considerable time after precipitation occurs; on-site man-made water systems such as the sewer system along the eastern margin of the site, septic tank(s) on site, and a waterline onto the site; the Latty Avenue ditch which runs westward along the northern margin of the site to an unnamed drainage leading to Coldwater Creek; a storm drainage on the northern side of Latty Avenue which also empties into this same drainage; an unnamed intermittent but perhaps perennial tributary to Coldwater Creek which runs northwestward along the southern boundary of the property; a short ditch along the southwestern section of the property which carries drainage from the site into this unnamed tributary of Coldwater Creek; a dry ditch running parallel to the western margin of the site along the railroad embankment; and evidence of on-site and off-site flow of waters during and after rainstorms. These features are shown in Figure 2-8.

The contour of the site results in very little flow of off-site waters onto the site. The higher elevation along the eastern fence is a barrier to most flows of off-site water onto the site, as is the railroad grade to the west of the site. Water flows off-site to the north into the Latty Avenue ditch and to the south and southwest into the unnamed tributary of Coldwater Creek. Percolation of ponded waters is evident, but the rate appears to be relatively slow as evidenced by dessication cracks and algal mats over portions of the lower sections of the property. Ongoing erosion of soil material is evident over barren areas of the site. Deposition of such material is evident in other areas of the property as well as erosion off-site to the north into the Latty Avenue ditch.

To the west of the site Coldwater Creek, a tributary of the Missouri River, flows northwestward. The site is within the creek's flood plain and was innundated by the equivalent of a 25- to 35-yr flood in 1957. Projections of a 100-yr flood or greater have not been made for the reach containing the site; consequently it is not known to what depth or at what velocity flood waters would flow across the site.

Surface water quality of Coldwater Creek as well as the water quality of the Missouri and Mississippi Rivers have been influenced adversely by extensive human settlements. Coldwater Creek water quality decreased with the rapid development of the area from 1930 to 1960. A sewage plant 8 mi downstream from the site discharges 25 million gallons per day of mechanically and biologically treated waters into Coldwater Creek. Immediately upstream from the site industrial treated waste water and storm water run-

off drain into Coldwater Creek. There is no evidence that the Latty Avenue site contributes ongoing contamination to Coldwater Creek, but no long-term monitoring program has been undertaken to verify this conclusion. There is some evidence that the Latty Avenue site contributes radioactive materials to the ditches along the northern and southern boundaries of the site. Water samples taken in the course of the field surveys of the site are described in Chapter 3, paragraph 3.4.6.

2.9 GROUND WATER

Ground water in St. Louis County occurs in unconsolidated deposits and in bedrock aquifers. Confined ground water systems in the limestone formations occur in limited quantities in fractures and solution channels. Yields are variable and unpredictable. At depths below 100 ft above sea level the water is saline and unfit for most domestic and industrial purposes. The St. Peters sandstone unit is an aquifer in other sections of the state but is not tapped in the St. Louis area. Some limestone units are recharged in the area downstream from the site. Because of the open nature of solution channels in limestone, there is little opportunity for natural filtration and purification. Also, it is unlikely that contaminants are carried off the Latty Avenue site and it is virtually impossible for such contaminants to percolate through the 100 ft of clays and silts of the lacustrine sediments underlying the site; nevertheless, should such contamination reach the recharge areas of the limestone units, it easily would enter these underground systems.

Excellent aquifers exist in the unconsolidated materials of the flood plains of the Missouri, Mississippi and Meramec rivers. Within the Florissant Basin the water content of the subsoils is very high but the percolation rates are low, and thus the area has virtually no potential for ground water development from shallow wells. Shallow ground water flows would flow towards Coldwater Creek. Because of clay and silt barrier layers, such movement would be slow and lateral along soil units and not downward. The ground water level varies, but water usually is found within 20 ft of the surface and during wet seasons is found within 5 ft to 10 ft of the surface.

The Missouri, the Mississippi, and the Meramec Rivers furnish nearly all of the water used in St. Louis County, St. Charles County, and Jefferson County. Ground water could be developed, but the abundance of surface waters has not encouraged such development. It is unlikely that unconfined or confined ground water systems would be developed in the Florissant Basin. Thus the potential for contamination from the site to these aquifers, and the potential usage of this water are unlikely.

2.10 METEOROLOGY

The Latty Avenue site is approximately 2 mi from the tower

at the Lambert-St. Louis International Airport, where weather records have been kept since 1929. The following data obtained from the National Weather Service at that location are applicable also to the Latty Avenue site.

St. Louis is located near the geographical center of the United States. Thus, it has a somewhat modified four-season continental climate. That is, the climate is without prolonged periods of extreme cold, extreme heat, or high humidity. To the south is the warm, moist air of the Gulf of Mexico, and to the north in Canada is a region of cold air masses. The alternate invasion of St. Louis by air masses from these sources, and the conflict along the frontal zones where they come together, produce a variety of weather conditions.

Winters are brisk but seldom severe. Records, since 1871, show that temperatures drop to zero or below an average of two or three days per year. Maximum temperatures remain as cold as 32° F or lower less than 20 to 25 days in most years. The record low temperature recorded by the National Weather Service was -22° F on January 5, 1884, but the all-time record low for St. Louis was a -23° F in 1864.

Snowfall has averaged less than 20 in. per winter season since 1930, and has varied from 0.7 in. in 1931-32 to 42.4 in. in 1973-74. Snowfall of 1 in. or more is received between 5 to 10 days in most winters; however, there have been seasons when less than an inch fell, as in 1931-32, and other years when there were 15 days or more with an inch or more snowfall. Snowfall of an inch or more has occurred as late as May 2 (depth in 1929 was 3 in.), and as early as November 5. It was November 5-6, 1951, that the fourth heaviest snowfall of record at St. Louis occurred at the airport, with a fall of 10.3 in. The greatest snowfall at the airport was 12.0 in. received on December 19, 1973. The winter of 1911-12 had the greatest total snowfall of record at St. Louis with 67.6 in.

The long-term record (since 1871) for St. Louis indicates that maximum temperatures of 90° F or higher occur an average of 35 to 40 days per year. Extremely hot days of 100° F or more are expected on no more than about 5 days per year. The highest temperature on record is 115° F, recorded at the airport on July 14, 1954.

The last temperature as low as 32° F in the spring at St. Louis has occurred as early as March 8 and as late as May 10, while the first occurrence of a freezing temperature in the fall has been as early as September 28 and as late as November 27. There is an average of approximately 190 days between the last freezing temperature in the spring and first such temperature in the fall, but this can vary from 150 days to around 230 days. The average date of the last freeze in the spring is April 15, and the average date of the first freeze in the fall is October 20.

Normal annual precipitation for the St. Louis area, based on the average for the period 1941-1970, is a little over 35 in.; but 68.83 in. was recorded in 1858, and as little as 20.59 in. in 1953. The three winter months are the driest, with an average total of about 6 in. of precipitation. The spring months of April to June are normally the wettest with normal total precipitation of nearly 12 in From the middle of summer in July into the fall, it is not unusual to have extended periods of 1 to 2 weeks or more without appreciable rainfall.

In the St. Louis area thunderstorms occur on the average between 40 to 50 days per year. During any year there are usually a few of these that can be classified as severe storms with hail and damaging winds. During the entire period of record there have been only four tornadoes which produced extensive damage and loss of life in St. Louis: 27 May 1896, 29 September 1927, 10 February 1959, and 24 January 1967.

Winds are predominantly from the south with a mean speed of 9.5 mi/hr. The next most prominent wind direction is from the northwest, which occurs mainly during the winter months. (See Table 2-1 for weather data summary.)

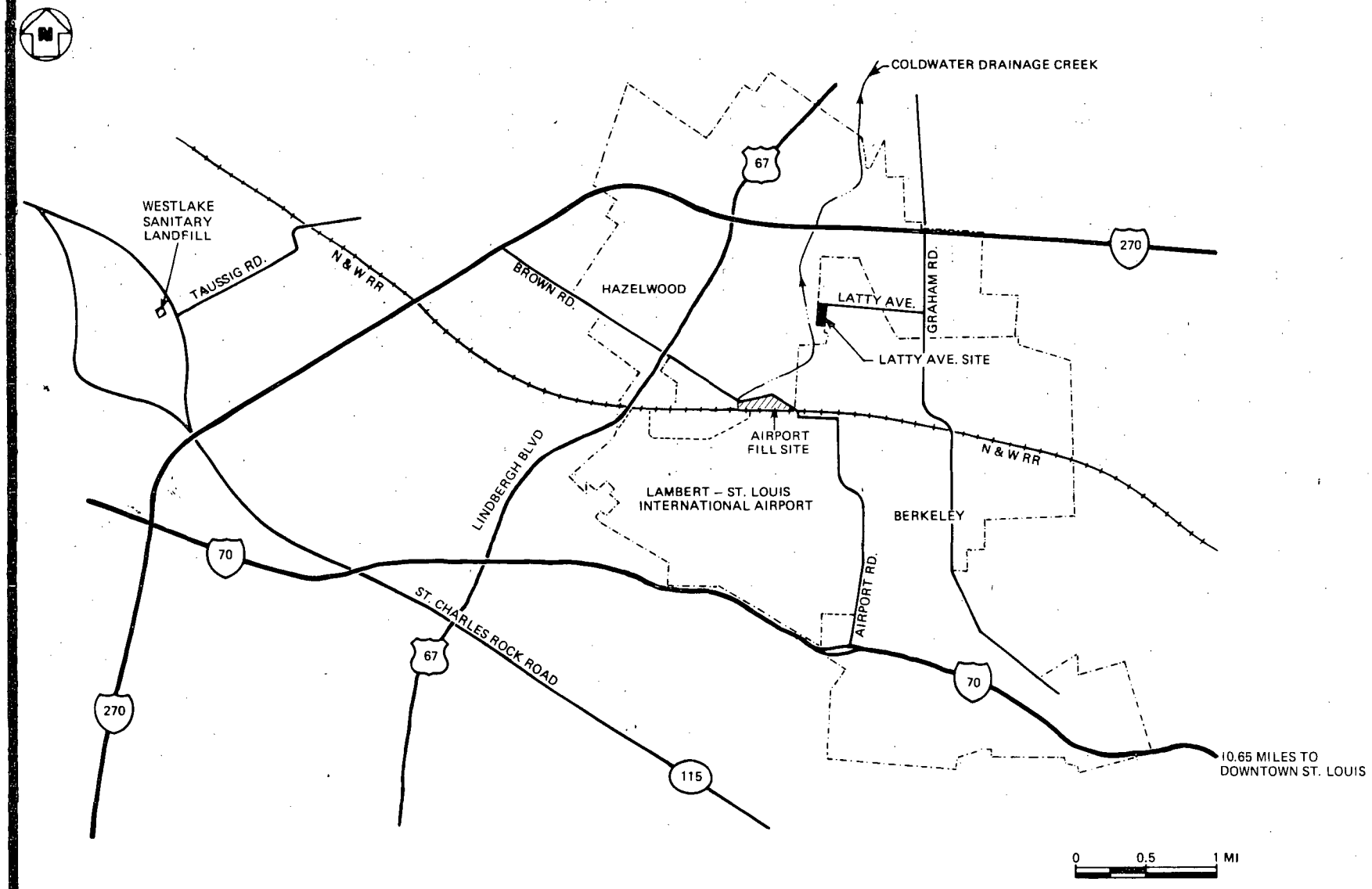


FIGURE 2-1. SITE LOCATION

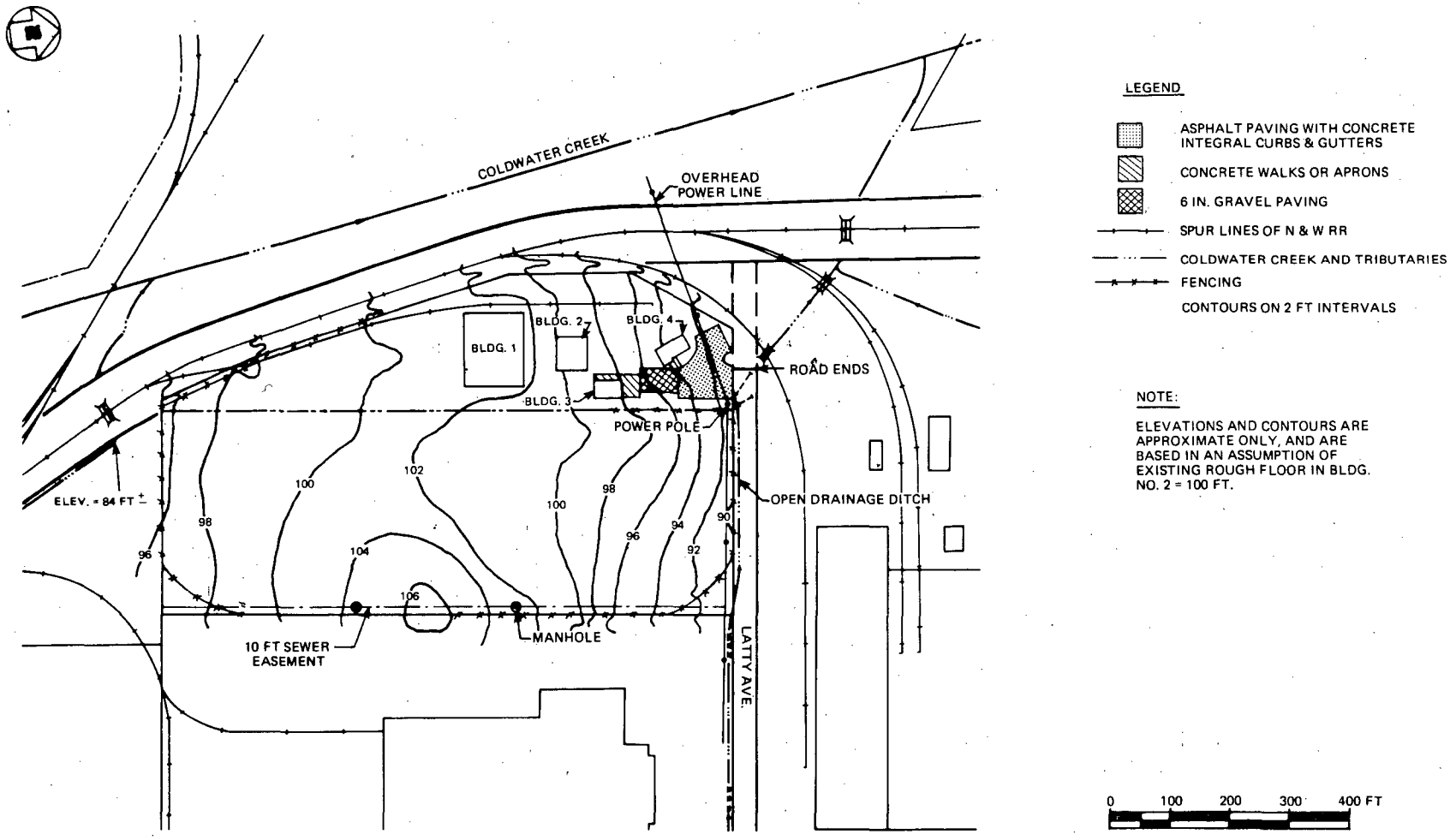


FIGURE 2-2. SITE DESCRIPTION AND TOPOGRAPHY

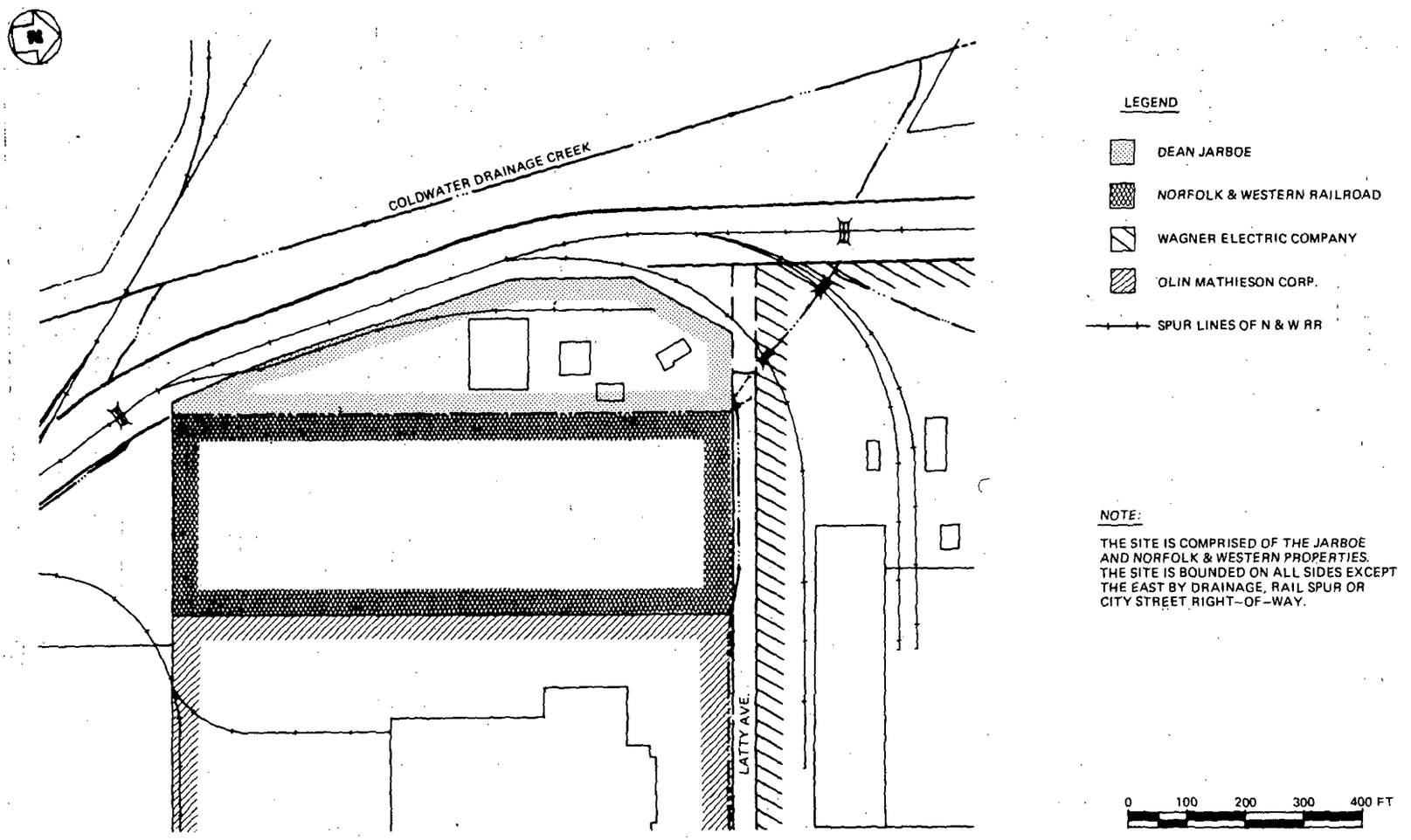
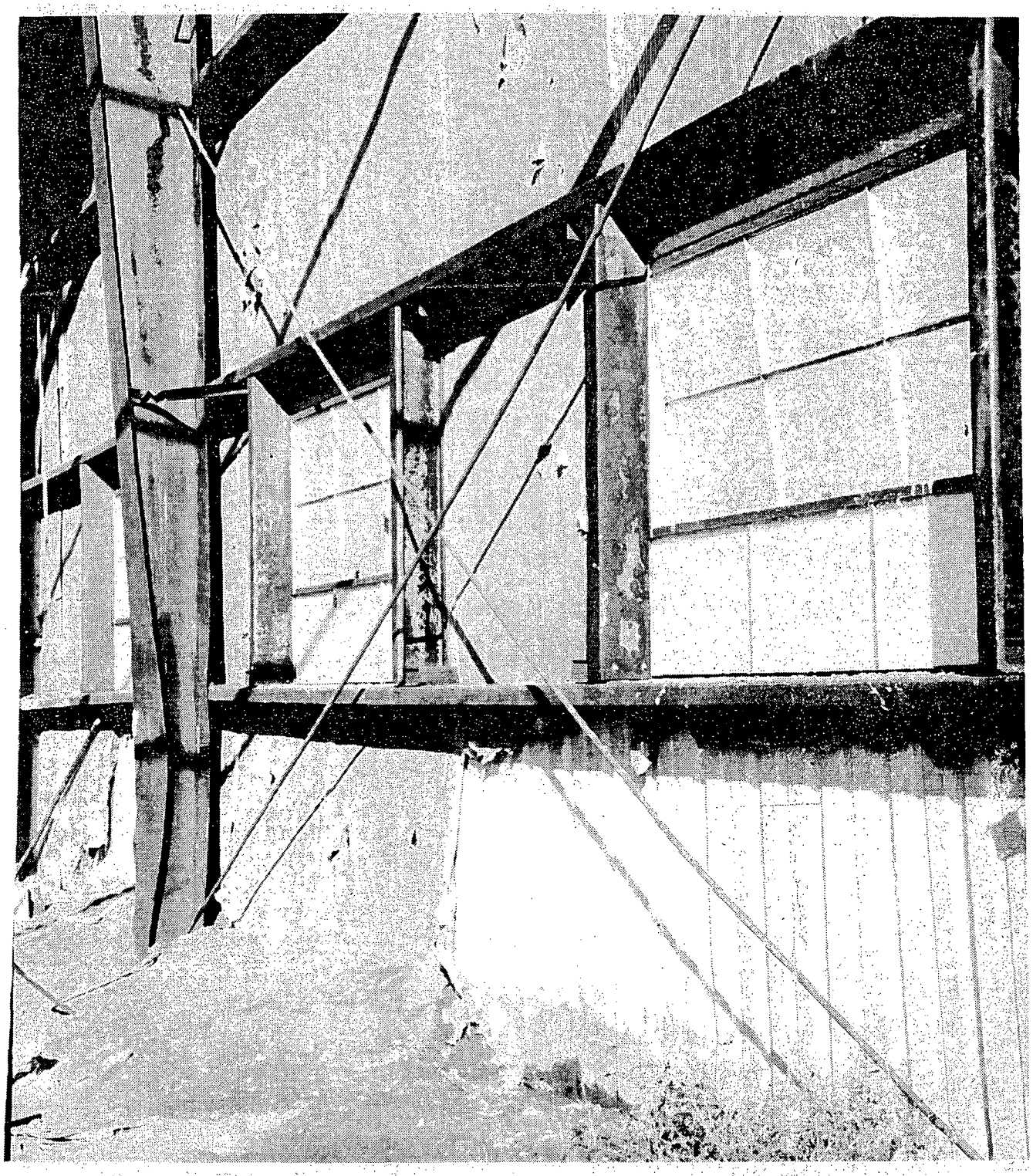
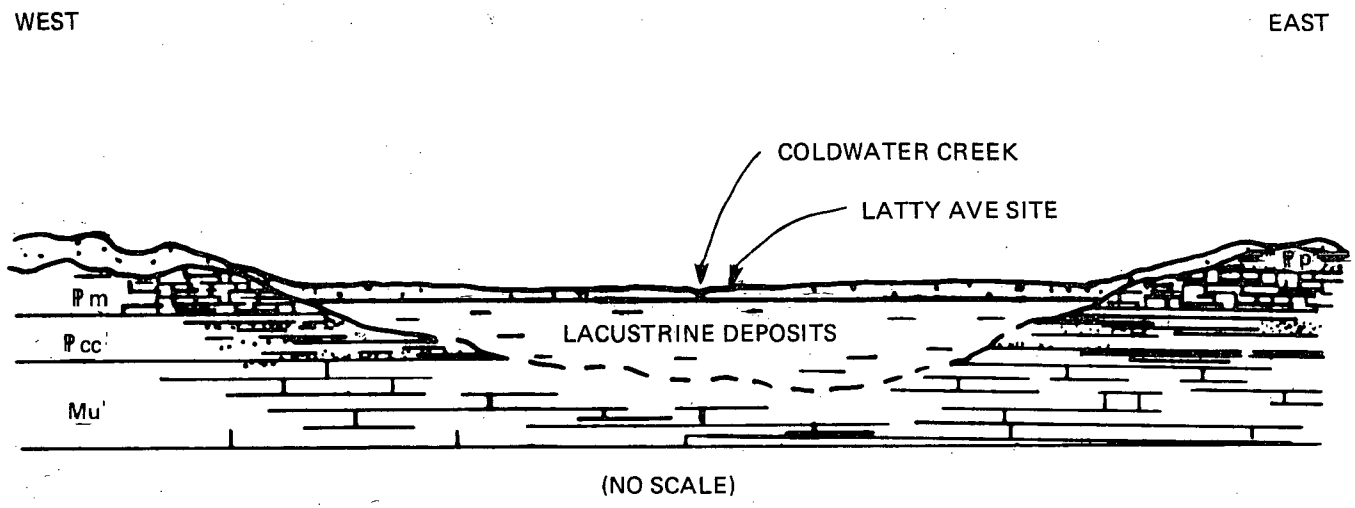


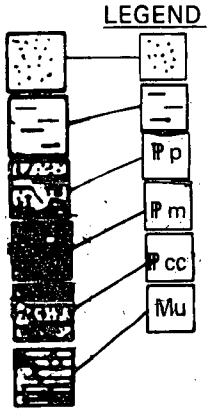
FIGURE 2-3. SITE OWNERSHIP



UC 22

FIGURE 2-4. NORTH-EAST INSIDE CORNER OF BUILDING 1, TAKEN OCTOBER 13, 197





LOESS

LACUSTRINE DEPOSITS OF GLACIAL AGE

PENNSYLVANIAN STRATA PRINCIPALLY SHALE AND SANDSTONE

PENNSYLVANIAN STRATA CYCLIC DEPOSITS WITH PROMINENT LIMESTONE UNITS

PENNSYLVANIAN STRATA CYCLIC DEPOSITS WITH PROMINENT SHALES AND SANDSTONES

MISSIPPIAN LIMESTONES (UNDIFFERENTIATED)

FIGURE 2-5. SIMPLIFIED GEOLOGIC CROSS-SECTION OF THE FLORISSANT BASIN

SYSTEM	FORMATION		THICK-NESS (FT)	CHARACTER
QUATERNARY	ALLUVIUM		0-150	SANDS, GRAVELS, SILTS AND CLAYS; A MAJOR AQUIFER
	GLACIAL DEPOSITS	LOESS	0-110	WIND DEPOSITS; SILTS, FINE SANDS; NOT AN AQUIFER
		LUCUSTRINE DEPOSITS	0-150	LAKE DEPOSITS; CLAYS, SILTS, SOME SANDS; AQUICLUDE
		TILL	0-55	GLACIER DEPOSITS; NON-SORTED, NON-STRATIFIED PEBBLY CLAYS AND SILTS; NOT AN AQUIFER
PENNSYLVANIAN	SEVERAL FORMATIONS		0-350	SHALES, SILTSTONES, SANDSTONES, SOME COAL BEDS AND THIN LIMESTONE UNITS; YIELDS VERY SMALL QUANTITIES OF WATER
MISSISSIPPIAN	SEVERAL FORMATIONS		0-900	LIMESTONES, OCCASIONAL SHALES; LOCALLY YIELDS SMALL TO MODERATE QUANTITIES OF WATER
DEVONIAN	3 FORMATIONS		0-300	SANDSTONES, LIMESTONES, SHALES; LOCALLY YIELDS SMALL QUANTITIES OF WATER
SILURIAN	UNDIFFERENTIATED		0-200	CHERTY LIMESTONES; NOT AN AQUIFER
ORDIVICIAN	SEVERAL FORMATIONS		UP TO 2200	MANY UNITS; SILTSTONES, SHALES, LIMESTONES, SANDSTONES, DOLOMITES; 3 UNITS ARE POTENTIAL MODERATE TO LARGE YIELD AQUIFERS
OLDER ROCK FORMATIONS				

FIGURE 2-6. SIMPLIFIED STRATIGRAPHIC COLUMN ST. LOUIS COUNTY, MISSOURI

UC-225

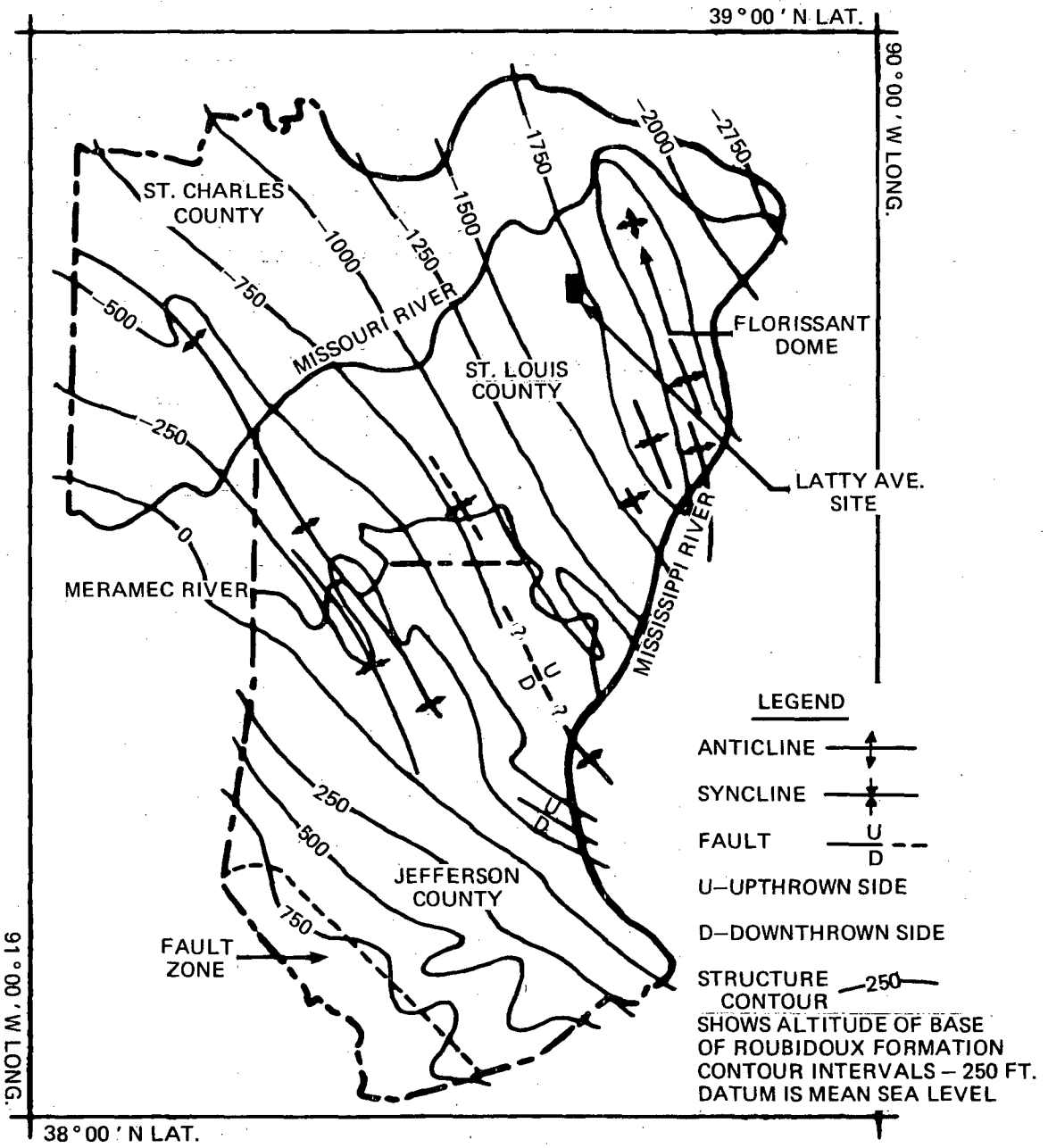


FIGURE 2-7. STRUCTURE MAP OF ST. LOUIS AREA, MISSOURI

UC-225

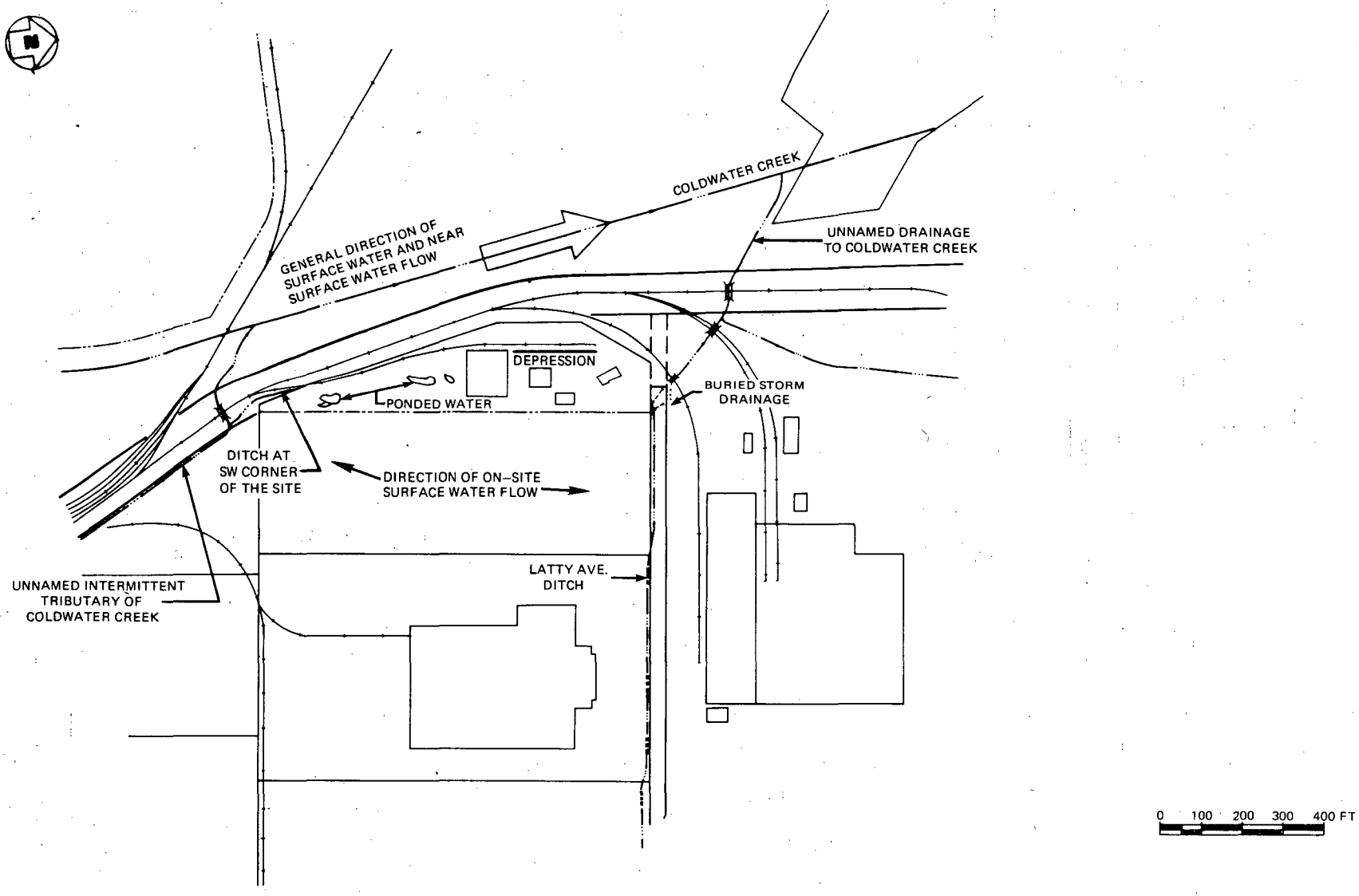


FIGURE 2-8. SURFACE DRAINAGE PATTERNS

Table 2-1

MEAN METEOROLOGICAL DATA IN ST. LOUIS COUNTY - 1941-1970
(Station: Lambert Field, St. Louis, Elevation: 535 ft)

	Monthly Temperature (deg F)			Precipitation (in.)		Relative Humidity (%)		
	Highest	Lowest	Average	Total	Snow, Ice Pellets	6 a.m.	12 p.m.	6 p.m.
January	76	-11	34.6	1.86	4.1	82	64	69
February	85	- 2	35.1	2.06	4.0	80	61	64
March	88	3	43.3	3.03	4.5	81	57	58
April	92	22	56.5	3.92	0.3	78	54	53
May	92	31	65.8	3.86	Trace	82	56	55
June	98	43	74.9	4.42	0	84	57	56
July	106	51	78.6	3.69	0	86	57	56
August	105	47	77.2	2.87	0	89	57	58
September	100	36	69.6	2.89	0	91	61	64
October	94	25	59.1	2.79	Trace	84	55	60
November	81	1	45.0	2.47	1.4	84	62	68
December	76	- 6	54.6	2.04	3.5	85	70	75
Year	106	-11	55.9	35.89	17.8	84	59	61

WIND

	Mean Speed (mi/hr)	Prevailing Direction	Percent of Possible Sunshine
January	10.3	NW	52
February	10.8	NW	51
March	11.8	WNW	54
April	11.4	WNW	56
May	9.4	S	62
June	8.6	S	69
July	7.6	S	71
August	7.4	S	66
September	7.9	S	63
October	8.5	S	62
November	9.9	S	49
December	10.2	WNW	41
Year	9.5	S	58

SOURCE: U.S. Department of Commerce, National Oceanic and Atmospheric Administration, National Weather Service.

CHAPTER 2 REFERENCES

1. "Interim Report of Radiological Survey of the Property at 9200 Latty Avenue, Hazelwood, Missouri"; ORNL; Oak Ridge, Tennessee; Sep 1977.

2. "Allegations Regarding the Disposal of Uranium Ore Residues"; draft report of NRC Office of Inspection and Enforcement, Region III, License No. SUB-1022 (terminated); Hazelwood, Missouri; June 22-24 and Aug 11, 1976.

CHAPTER 3

RADIOACTIVITY AND POLLUTANT IMPACT ON THE ENVIRONMENT

The principal objective of the assessment in this chapter is to determine the magnitude and characteristics of the radiation emitted from the Latty Avenue site and the resulting potential exposure to the population residing and working in the vicinity of Hazelwood and Berkley, Missouri. In addition, this chapter describes briefly the potential radioactive pollutants and their pathways in the environment. The notations and abbreviations used are given in Table 3-1.

Most of the radiological measurements discussed in this chapter were made by Oak Ridge National Laboratory and are reported in a separate document[1]. FB&DU performed radon and gamma measurements off the site, radon flux measurements on site, decontamination tests and contamination measurements in the buildings, and collected additional water and soil samples to aid in determining depth and extent of contamination.

3.1 RADIOACTIVE MATERIAL CHARACTERISTICS

Many elements spontaneously emit subatomic particles; therefore, these elements are radioactive. For example, when the most abundant uranium isotope, ^{238}U undergoes radioactive decay, it emits a subatomic particle called an alpha particle; the ^{238}U after undergoing decay becomes ^{234}Th, which is also radioactive; and ^{234}Th subsequently emits a beta particle and becomes ^{234}Pa. As shown in Figure 3-1, this process continues with either alpha or beta particles being emitted, and the affected nucleus thereby evolves from one element into another. As noted in Figure 3-1, ^{230}Th decays to ^{226}Ra, which then decays to ^{222}Rn, an isotope of radon. Radon, a noble gas, does not react chemically. The final product in the chain is ^{206}Pb, a stable isosope that gradually accumulates in ores containing uranium. Uranium ore contains ^{226}Ra and the other daughter products of the uranium decay chain. One of the daughters of ^{226}Ra is the isotope ^{214}Bi, which emits a significant amount of electromagnetic radiation known as gamma radiation. Gamma rays are very similar to X-rays, only more penetrating. The ^{214}Bi is the principal contributor to the gamma radiation exposure in the uranium-radium decay chain.

Besides knowing the radioactive elements in the decay chain, it is also important to know the rate at which they decay. This decay rate, or activity, is expressed in curies (Ci) or picocuries (pCi), where 1 pCi equals 10^{-12} Ci or 3.7×10^{-2} disintegrations per second. The picocurie often is used as a unit of measure of

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[1] See end of chapter for references.

the quantity of a radioactive element present in soil, air, and water.

Another important parameter used in characterizing radioactive decay is known as the "half life", $T_{1/2}$. This is the time that it takes for half of any initial quantity of the radioactive atoms to decay to a different isotope. For example, it takes 4.5 x 10^9 yr for half the ^{238}U atoms to decay to ^{234}Th. Similarly, half of a given number of ^{222}Rn atoms will decay in 3.8 days.

The activity and the total number of radioactive atoms of a particular type depend upon their creation rates as well as their half life for decay. If left undisturbed, the radioactive components of the decay chain shown in Figure 3-1 all reach the same level of activity, matching that of the longest-lived initiating isotope. This condition is known as secular equilibrium. When the uranium is removed in the milling process, ^{230}Th, which is not removed, becomes the controlling isotope. After processing the ore for uranium, the thorium, radium, and other members of the decay chain remain in the spent ore solids in the form of a waste slurry called tailings.

At the Latty Avenue site, ore residues, process wastes and liquid raffinates, all containing uranium, thorium, and radium, were present during the period that the facility was in operation. Although these materials were removed from the site, the radioactive contamination that remains contains radium and thorium and their daughters that are of concern because of possible health effects.

3.2 RADIATION EFFECTS

The radioactive exposure encountered with uranium processing wastes occurs from the absorption within the body of the emitted alpha and beta particles, and gamma radiation. The range of alpha particles is very short; they mainly affect an individual when the alpha emitter is taken internally. Beta particles have a much lighter mass than alphas, and have a longer range; but they still cause damage mainly to the skin or internal tissues when taken internally. Gamma rays, however, are more penetrating than X-rays and can interact with all of the tissue of an individual near a gamma-emitting material.

The biological effects of radiation are related to the energy of the radiation; therefore, exposure to radiation is measured in terms of the energy deposited per unit mass of a given material. In the case of radon and its daughter products, the principal effect is from alpha particles emitted after the radon and its daughter products are inhaled.

The basic units of measurement for the alpha particles from short-lived radon daughters are the working level (WL) and the working level month (WLM). The working level is defined as any combination of the short-lived radon daughters in a liter of air

that will result in the ultimate emission of 1.3×10^5 MeV of alpha energy. The working level is so defined because it is a single unit of measure, taking into account the relative concentrations of radon daughter products which vary according to factors such as ventilation. One WLM results from exposure to air containing a radon daughter concentration (RDC) of 1 WL for a duration of 170 hr.

The basic units of measurement for gamma radiation exposure and absorption are the roentgen (R) and the rad. One R is equal to an energy deposition of 88 ergs/g of dry air, and 1 rad is the dose that corresponds to the absorption of 100 ergs/g of material. The numerical difference between the magnitude of the two units is often less than the uncertainty of the measurements, so that exposure of 1 R is often assumed equivalent to an absorbed dose of 1 rad or a gamma dose of 1 rem.

3.3 NATURAL BACKGROUND RADIATION

There are several sources of radiation that occur naturally in the environment. Natural soils contain trace amounts of uranium, thorium, and radium that give rise to radon gas and to alpha, beta, and gamma radiation. The average background value of ^{226}Ra concentration measured in 4 off-site soil samples was 1.3 pCi/g.[1] The samples were taken within a 4-mi radius of the site. Another natural source of radiation in the environment arises from the decay of ^{232}Th, the predominant thorium isotope. The half-life of ^{232}Th is 1.4×10^{10} yr. It is also the parent of a decay chain containing isotopes of radium and radon. The average background concentration of ^{232}Th in the same off-site soil samples was 1.2 pCi/g. The average concentration of ^{238}U in these samples was 1.2 pCi/g.

Background values of ^{222}Rn concentrations were measured at 3 locations between 0.4 and 6 mi from the site using continuous radon monitors.[2] An average background value of 0.8 ± 0.2 pCi/l was obtained from the 24-hr samples. However, the range of the measurements extended from 0.6 to 1.1 pCi/l.

Background gamma ray levels, as measured 1 m above the ground, also were determined at the 4 background locations within 4 mi of the site by using an energy-compensated Geiger Mueller detector. The measurements ranged from 7 to 9 μR/hr. Cosmic rays are part of the measured background radiation levels. The contribution from cosmic rays is generally dependent upon the altitude and is approximately 5 μR/hr in the St. Louis area,[3] or approximately 60% of the measured average background value.

3.4 RADIATION EXPOSURE PATHWAYS AND CONTAMINATION MECHANISMS

As noted previously, the principal environmental radiological implications and associated health effects of uranium mill wastes are related to radionuclides of the ^{238}U decay chain: primarily ^{230}Th, ^{226}Ra, ^{222}Rn, and ^{222}Rn daughters. Depending upon

the source and type of uranium ore, ^{232}Th and ^{235}U and their daughters also may be present. Although these radionuclides occur in nature, their concentrations in mill waste materials are several orders of magnitude greater than in average natural soils and rocks. The major potential routes of exposure to man are:

(a) Inhalation of the ^{222}Rn daughters, from decay of ^{222}Rn escaping from radioactive waste material at the site; the principal exposure hazard is to the lungs.

(b) External whole-body gamma exposure directly from the radionuclides in the radioactive waste materials on site (primarily from ^{214}Bi) and in the general vicinity of the site.

(c) Inhalation of windblown radionuclides from the site; the primary hazard relates to the alpha emitters ^{230}Th and ^{226}Ra, each of which causes exposure to the bones and the lungs.

(d) Ingestion by man of ground or surface water contaminated from either radioactivity (primarily from ^{226}Ra) leached from the radioactive waste materials or from solids physically transported into surface water.

(e) Erosion and removal of radioactive waste materials from the site by flood waters or heavy rainfall; this can create additional contaminated locations with the same problems as the original site storage area.

(f) Physical removal of radioactive waste materials from the site storage area also provides a mechanism for contamination of other locations.

(g) Contamination of food through uptake and concentration of radioactive elements by plants and animals is another pathway which can occur; however, this pathway was not considered in this assessment.

The extent of radiation and pollution transport from the site into the environment is discussed in the following paragraphs.

3.4.1 Radon Gas Measurements and Diffusion

Measurements of radon flux were made at four locations on the site using the charcoal canister technique.[4] The locations and flux values are shown in Figure 3-2. In general, reported values of radon flux vary considerably from time to time at a single sampling location. This variation is due in part to dif-

fering moisture, soil, and climatological factors, and to the difficulty of performing the measurements.

Radon flux depends principally upon the radium content of the soil or waste material. Background values of flux are generally in the range of 1 to 2 pCi/m^2-s. The four measurements ranged from near background to 55 pCi/m^2-s. There was no rainfall during the measurement period, but 0.18 in. of rain was recorded the preceding day.

Radon gas concentrations were measured for 24 hr each at the 3 background locations mentioned previously, at 2 other off-site locations, and at 1 on-site location using the continuous radon monitors. Figure 3-3 shows the measurement locations and lists the 24-hr average values of ^{226}Ra concentration. All outside measurements were within the background range of 0.6 to 1.1 pCi/l.

The radon concentration measurements are plotted in Figure 3-4 as a function of distance from the edge of the site. Also shown in the figure are the FB&DU model results. Model calculations were performed with annual meteorology data to provide an additional estimate of the radon concentration in the vicinity of the site. The FB&DU model first determines radon flux and the total radon releases from the site with diffusion theory using radium soil concentrations and site configurations deduced from the drilling and survey data. Then, the radon transport off-site is calculated by Gaussian diffusion[5] plus wind drift conditions. The model curve was used to calculate potential health effects resulting from radon diffusing from the site.

Radon concentrations were measured in the four buildings on the site.[1] In Building 1, measurements at 4 locations were averaged over a 24-hr period. The radon concentrations ranged from 5.8 to 20.3 pCi/l, with an average of 15.4 pCi/l. In Building 2, a 24-hr average measurement yielded a concentration of 2.7 pCi/l. In Building 3, 0.5 pCi/l of radon was measured. Two 24-hr average measurements in Building 4 gave radon concentrations of 1.5 and 1.9 pCi/l. Maximum concentrations during the 24-hr measurement period were about 3 times the average values and above the 3 pCi/l concentration limit of 10CFR20 in Buildings 1, 2, and 4.

Sample analyses of radioactive materials from the Latty Avenue site showed the presence of ^{238}U, ^{235}U, and ^{232}Th.[1] These three isotopes produce ^{222}Rn (radon), ^{219}Rn (actinon), and ^{220}Rn (thoron). The latter two isotopes have half-lives that are short compared to that of ^{222}Rn; and because the continuous radon monitors operate on the diffusion principle, the radon concentrations measured with those units are due principally to ^{222}Rn. The presence of the other two isotopes of radon and their daughters does create interferences that affect the working level (WL) measurements of ^{222}Rn daughters. Spectral counting of filter samples reveals the presence of these other isotopes and allows one to evaluate the data properly.

Analysis of filter samples led to the conclusion that virtually all of the activity in Building 1 was due to actinium daughters in the ^{235}U decay chain. In Buildings 2 and 3 most of the activity is from actinium daughters, and in Building 4, the predominant activity is due to ^{222}Rn daughters.[1] In Building 4, the maximum observed radon daughter concentration was 0.005 WL and the average concentration was 0.003 WL.[1]

3.4.2 Radiation Measurements Inside Structures

The ORNL performed gamma radiation rate measurements on a grid pattern at 1 m above the surface in Buildings 1, 2, and 3 and in several offices in Building 4. These measurements are plotted in Figures 3-5 through 3-8.[1]

Beta-gamma dose rates at 1 cm above the surface and alpha contamination also were measured inside the buildings by ORNL. These measurements are shown in Figures 3-9 through 3-12. Analyses of the contamination shows the presence of ^{226}Ra, ^{230}Th, and ^{227}Ac, so that the strictest NRC guidelines apply to cleanup for release of the structures for unrestricted use.[1] Removable contamination was measured at several locations in the buildings by ORNL. During this survey, additional measurements of alpha contamination were made in all four buildings before and after obtaining wipes to determine smearable contamination and to determine the effectiveness of several decontamination techniques. These tests were necessary to compare costs of decontamination with demolishing the buildings, hauling and burying the contaminated material, and building new structures. A discussion of the measurements, decontamination tests and agents, and detailed cost breakdowns for the decontamination efforts are included in Appendix A. These results were factored into costs of the remedial action alternatives in Chapter 7.

Comparison of contamination levels with the remedial action criteria discussed in paragraph 3.5 shows that extensive decontamination of Buildings 1 and 2 and portions of 3 will be required. Building 4 has contamination on the floors that could be removed economically.

3.4.3 Direct Gamma Radiation On Site

The external gamma radiation (EGR) levels measured 1 m above ground at grid points on the site are shown in Figure 3-13. These measurements include background and were taken with energy-compensated Geiger-Mueller detectors.[1] The highest gamma radiation rate on the site (500 μR/hr) was measured at the northern edge of the site where runoff from precipitation collects.

Other areas of high gamma rates are adjacent to the northwest and northeast corners of Building 1. One of these areas is at the end of the railroad spur.

Beta-gamma dose rates at the same grid points, but 1 cm above ground, are shown in Figure 3-14. The radiation rates reach up to 1 mR/hr at the northern edge of the site and near the end of the railroad spur.

3.4.4 Soil Contamination

Surface and subsurface soil samples were taken on the site and inside Buildings 1 and 2. Locations of these samples are given in Figures 3-15 and 3-16. Values of ^{226}Ra, ^{238}U, ^{227}Ac, and ^{232}Th concentration in the soil samples are listed in Tables 3-2 and 3-3.

A large area in the center of the site shows only slight surface contamination, less than 6 in. deep. Other areas near Building 1 and a few isolated areas near the perimeter of the site have high levels of contamination at the surface; and 2 to 3 ft of contaminated soil must be removed to reach approximately twice the background radium content in the soil (1.3 pCi/g).

The FB&DU field crew augered several additional holes to aid in defining the amount of contaminated soil to be removed during decontamination operations. These hole locations have been added to the ORNL map of core hole locations shown in Figure 3-16.

Depth of contamination information was analyzed and a map of the site was drawn indicating the depth of soil to be removed. This map is shown in Figure 7-2, Chapter 7, where remedial action alternatives are discussed.

3.4.5 Windblown Contamination

FB&DU performed external gamma radiation rate measurements 10 to 20 ft outside the site boundaries to determine the extent of the spread of radioactive contamination from the site. Figure 3-17 shows the measurements around the site boundary. Two measurements of background gamma rates at radon measurement locations were 12 and 16 μR/hr. On a traverse to the east of the site, an 11 μR/hr reading was recorded within 100 ft east of the property line, and 15 μR/hr was measured at 200 ft south of the southeast corner of the site.

At each measurement point shown in Figure 3-17, a scintillator probe was used with and without a lead shield between the probe and the ground. The difference between the two readings (Δ) is an indication of the presence of radioactive materials on the ground at the measurement location. Even without windblown radioactive materials on the surface, there is an observable difference between the two readings. In the absence of surface soil sample analyses of radium content off the site, a judgement was made of the Δ value indicative of the presence of radioactive materials based upon previous work on uranium mill tailings. These locations where radioactive material is considered to be present on the surface are marked with an asterisk.

An aerial survey of the gamma radiation from the site was performed by EG&G in the fall of 1977. The survey results are shown in Figure 3-18. The smooth concentric isocontours indicate little off-site radioactivity, and the elevated levels off the site are primarily the result of "shine" from the site.

3.4.6 Surface Water Contamination

Water samples were collected from drainage ditches at the north edge of the site and at the southwest corner of the site, and from Coldwater Creek 1 mi downstream from the site. Concentrations of ^{230}Th, ^{226}Ra, and ^{210}Pb in the water samples were all well below maximum permissable concentration guides in 10CFR20, but elevated levels of these isotopes in sediments filtered from the water samples taken in the drainage ditch indicate that some radioactive material is being carried off the site by surface water runoff.[1] The water sample from Coldwater Creek contained only background values of the three isotopes. Figure 3-19 shows the locations of additional water and sediment samples from the site vicinity. Radiometric analyses are listed in Table 3-4.

3.5 REMEDIAL ACTION CRITERIA

Radiological criteria were established for the Phase II - Title I engineering assessment of inactive uranium mill tailings sites[6] for remedial action applicable to structures with tailings underneath them or within 10 ft,[7] and criteria pertaining to the mill tailings site and open land.[8] Copies of the complete documents establishing these criteria are presented in Appendix B. Also given in Appendix B are the Grand Junction remedial action criteria for structures (10CFR712). These criteria could also be applied to cleanup of the Latty Avenue site.

The criteria which apply to the structures are the guidelines published by the Surgeon General of the United States.[7] These guidelines recommend the following graded levels for remedial action in terms of the EGR levels and indoor RDC levels above background found within the dwellings constructed on or near uranium mill tailings:

EGR, mR/hr	RDC*, WL	Recommendation
Greater than 0.1	Greater than 0.05	Remedial action indicated
From 0.05 to 0.1	From 0.01 to 0.05	Remedial action may be suggested
Less than 0.05	Less than 0.01	No remedial action indicated

*Based upon yearly average values from 6 air samples of at least 100-hr duration taken at a minimum of 4-wk intervals throughout the year.

The radiological criteria for decontamination of inactive uranium millsites and for open areas are based upon EGR readings above background, measured 3 ft above ground.[8] Decontamination should result in residual exposures that are as low as practicable. For this assessment the following criteria could be applied:

(a) For the radioactive materials:

(1) Contaminated materials should be covered so that residual gamma ray levels do not exceed 0.040 μR/hr above background. The area also should be designated a control area with restricted access.

(2) Where the site is not considered suitable for long-term stabilization, remove radioactive materials so that residual radium concentration in the soil does not exceed twice background values.

(b) Windblown tailings in open land areas near to or adjacent to the site:

(1) If gamma levels are less than 0.010 mR/hr above background, the land may be released for unrestricted use.

(2) If gamma levels exceed 0.010 mR/hr above background, cleanup should reduce the radium soil concentration to no more than twice background.

(3) If removal of radioactive materials is not practicable, residual gamma levels should in any part of the area not exceed 0.040 mR/hr above background.

The NRC has guidelines in effect for decontamination of facilities and equipment prior to release for unrestricted use. These guidelines also are included in Appendix B. Since ^{226}Ra, ^{230}Th, and ^{227}Ac were found on site, the most restrictive guidelines for acceptable surface contamination levels apply to this site. These guidelines allow a maximum of 100 dpm/100 cm^2 average activity and a maximum of 300 dpm/100 cm^2 over an area of 100 cm^2. The limit for removable contamination on a surface wipe is 20 dpm/100 cm^2.

3.6 POTENTIAL HEALTH IMPACT

An assessment has been made of the potential health impact of the site. The six environmental pathways described in paragraph 3.4 were evaluated. A summary of the evaluation of each pathway is presented below:

(a) Radon Diffusion - inhalation of radon daughters from radon diffusion constitutes the most significant pathway, but results in only a small estimated population dose.

(b) External Gamma Radiation - gamma radiation above background is measurable to distances up to 200 ft from the site, an area with no inhabitants and only occasional occupancy by employees. People on site will receive some gamma exposure until the radioactive materials are removed or covered with sufficient material to reduce the gamma radiation. Exposure to employees in the vicinity of the site has been evaluated and found to have negligible health impact.

(c) Airborne Activity - the limited spread of windblown radioactive materials toward inhabited areas indicates that direct inhalation or ingestion of radioactive particles is a minor component of the total population dose. Remedial actions will eliminate any gradual accumulation of materials off the site.

(d) Water Contamination - although the sediment sample results indicate slight local water contamination in the ditch at the north end of the site, a sample of water from Coldwater Creek shows no indication of contamination from the site. Since no potable water is obtained from this creek, the present health impact from this pathway is negligible. Furthermore, all water sample analyses contained ^{226}Ra concentrations below EPA Interim Drinking Water Regulations.

(e) Subsoil Contamination - leaching of radioactive materials into the ground beneath the former storage areas and at some of the structures reaches 3 ft in some areas.

Only the potential health effects from the inhalation of radon daughters (pathway a) are estimated quantitatively in this assessment because this pathway constitutes the most significant pathway. Furthermore, the uncertainty in the estimates of the potential health effects from this pathway far exceeds the magnitude of the health effects from the other pathways.

It is extremely difficult to predict with an assurance that a specific health effect will be observed within a given time after chronic exposure to low doses of toxic material. Therefore, the usual approach to evaluation of the health impact of low-level radiation exposures is to make projections from observed effects of high exposures on the basis that the effects are linear, using the conservative assumption of no threshold for the effects. The resulting risk estimators also have associated uncertainties due to biological variability among individuals and to unknown contributions from other biological insults which may be present simultaneously with the insult of interest. No synergistic effects are considered explicitly in this analysis. For the purpose of this engineering study, lung cancer is the potential health effect considered for RDC. The health effects were estimated using both an absolute and a relative risk model.

3.6.1 Assumptions and Uncertainties in Estimating Health Effects

Since radiation exposure from ^{222}Rn daughters is expressed in terms of working levels (WL) and working level months (WLM), total population exposures as well as health risk estimates are based upon these units, i.e. person-WLM. Exposures and resulting health effects often are expressed in terms of rems; however, estimates of the WLM-to-rem conversion factor for internal lung exposure to alpha particles from ^{222}Rn daughters vary by over an order of magnitude. Presently, there are significant differences of opinion related to the choice of an appropriate conversion factor. Consequently, disagreements of calculated health effects from RDC occur when these effects are based on the rem.

The absolute risk estimator used in this assessment is that given in the report of the National Academy of Sciences Advisory Committee on the Biological Effects of Ionizing Radiation (BEIR report).[9] This report present risk estimators for lung cancer derived from epidemiological studies conducted on two groups of miners, namely:

3 cancers per year per 10^6 person-WLM exposure for uranium miners

8 cancers per year per 10^6 person-WLM exposure for fluorspar miners

Therefore, the average of these two values was chosen as the risk estimator for use in this study. This estimator then is:

6 cancers per year per 10^6 person-WLM exposure

A dose from a given ingestion or inhalation of radionuclides varies widely due to differences in age (infants-adults), physical size, etc. This and other components of natural biological variability which exist among members of any given population, as well as the differences between exposure conditions in residences and mines, give rise to an uncertainty on the order of a factor of 3 in this parameter.[10]

The commitment, then, of 6 cancers per year has a statistical basis and relates to a total population exposure of 10^6 person-WLM. If a cancer does occur, it likely will be evident during the 30-yr period following the initial exposure and latency period.[11] When the exposure is continual over an individual's lifetime, this commitment is cumulative and the risk per year increases to an ultimate value of 6 times 30, or:

180 effects per year for 30 x 10^6 person-WLM total cumulative exposure

This mathematical expression also can be interpreted in terms of the average annual risk to an individual per unit of exposure.

For example, an individual with a continuous exposure of 1 WLM annually has about a 2×10^{-4} probability each year of developing lung cancer from this exposure. Several investigations have been reported recently concerning the association between lung cancer incidence and RDC exposures in miners.[10,12,13] These investigations yielded risk estimator values consistent with the risk estimator used in the present assessment. The relative risk estimator can be several factors larger than the absolute risk estimator.[14]

For the purposes of this assessment, equivalent working levels inside structures are determined from the radon concentration assuming a 50% equilibrium condition. This yields the following conversion factor:

$$1 \text{ pCi/l of } {}^{222}\text{Rn} = 0.005 \text{ WL}$$

It is assumed that the component of indoor radon concentration due to radon exhaled from the piles is equal to the corresponding outdoor concentration component at that point. However, the concentration of radon daughters is higher indoors owing to reduced ventilation and to other sources of radon, such as building materials.

The exposure rate in terms of WLM/yr can be obtained from a continuous 0.005 WL concentration (equivalent to 1 pCi/l Rn concentration) as follows:

$$(0.005 \text{ WL})\ (8766\ \frac{\text{hr}}{\text{yr}}) \left[\frac{1 \text{ WLM}}{1 \text{ WL } (170 \text{ hr})}\right] = 0.25\ \frac{\text{WLM}}{\text{yr}}$$

The risk estimator[9] used for continual exposure to gamma radiation is:

100 effects per year for 10^6 person-rem continuous exposure

In this assessment it is assumed that a gamma exposure of 1 R in air is equivalent to a dose of 1 rem in soft tissue.

3.6.2 Health Effects

The model curve of radon concentration-versus-distance (Figure 3-4) is used to determine the health effects due to radon from the site. First, an indoor radon daughter concentration is deduced from the outdoor radon concentration curve using the conversion factor 1 pCi/l of ^{222}Rn outside equals 0.25 (WLM/yr) inside, then, the resulting RDC distribution is multiplied by the risk estimators given previously to yield the health effect risk per person as a function of distance from the site. The estimated annual radiation-induced lung cancer risk due to the site is given in Figure 3-20 as a function of distance from the edge of the site for prolonged continuous exposure. The curves shown in the figure represent the sum of the estimated annual radiation-induced risk from the site plus the average lung cancer risk per

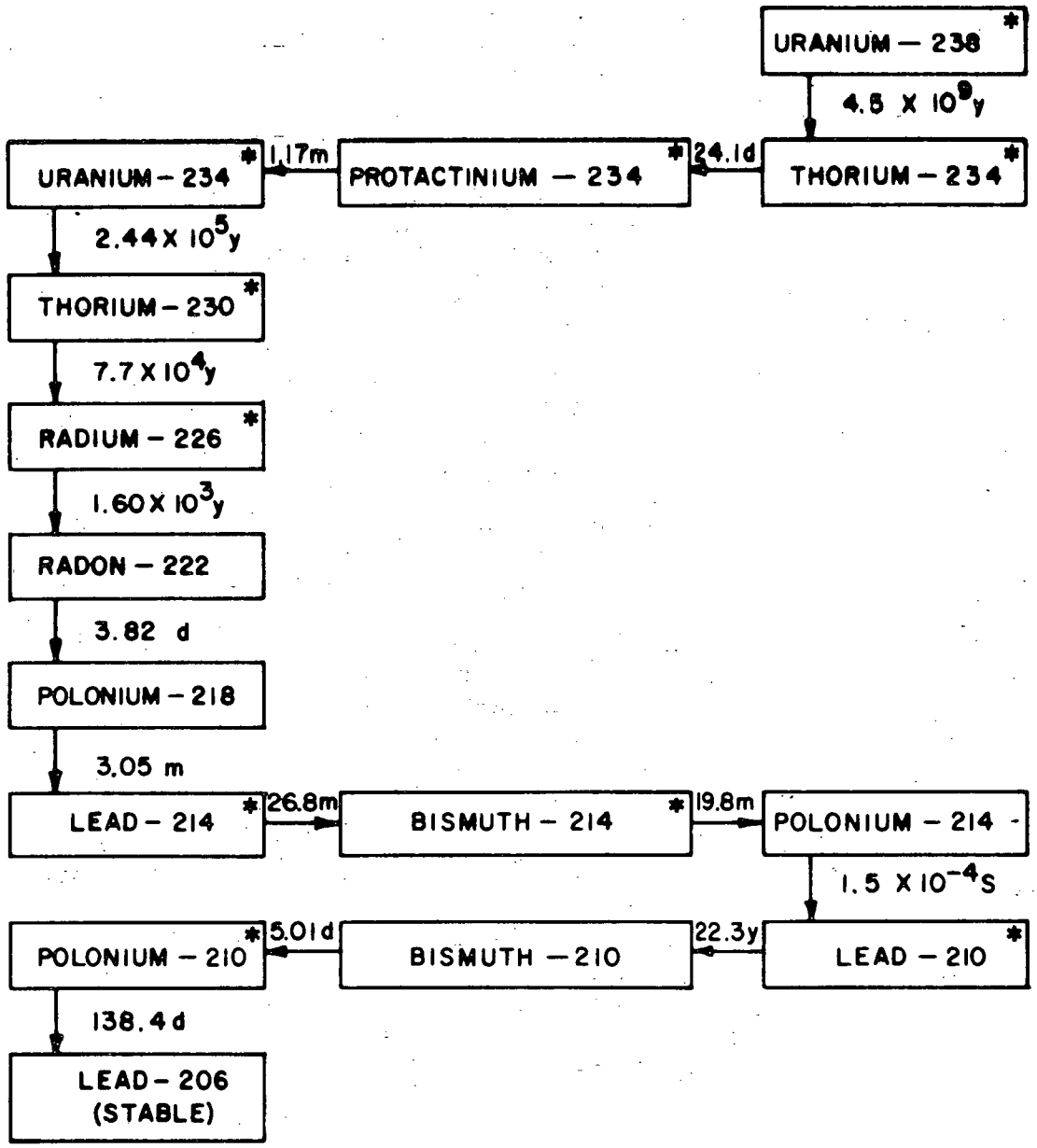
year from all causes for residents of the State of Missouri.[15] It is noted that the risk for developing lung cancer from site radon is only a small fraction of the natural occurrence risk at a distance of 0.1 mi from the edge of the site.

No health effects were attributed to gamma radiation from the site, because population gamma exposures are negligible within 200 ft of the site where the gamma radiation from the site is greater than the background range.

Health effects from total population RDC exposures for the area within 0.75 mi from the site perimeter are obtained by multiplying the health effect risk per person from the curves given in Figure 3-20, by the population distribution as a function of distance from the site. The results are given in Table 3-5. Beyond 0.75 mi, the site radon concentration is so low that the contribution to health effects is negligible. The population was estimated using 1970 census enumeration and projections by the St. Louis County Department of Planning. The population distribution as a function of distance and direction from the site was considered in the health effects calculations.

Also shown in the tables are health effects estimated from background radon concentrations. The site-induced radon daughter health effects are approximately 1% of background values for the area within 0.75 mi of the tailings. Table 3-5 also includes 25-yr cumulative health effects based on two growth rate projections.

If the relative risk estimator is used, the health effects estimates are correspondingly larger than the ones given in Table 3-5. The uncertainty in the health effects estimation is about a factor of 4.



NOTE:

VERTICAL DIRECTION REPRESENTS ALPHA DECAY, HORIZONTAL DIRECTION INDICATES BETA DECAY. TIMES SHOWN ARE HALF LIVES. ONLY THE DOMINANT DECAY MODE IS SHOWN.

* ALSO GAMMA EMITTERS

FIGURE 3-1. RADIOACTIVE DECAY CHAIN OF URANIUM 238

CANISTER NO.	FLUX pCi/m^2 –SEC
C–1	55.2 ± 2.4
C–2	4.9 ± 1.1
C–3	2.3 ± 1.1
C–4	17.3 ± 1.6

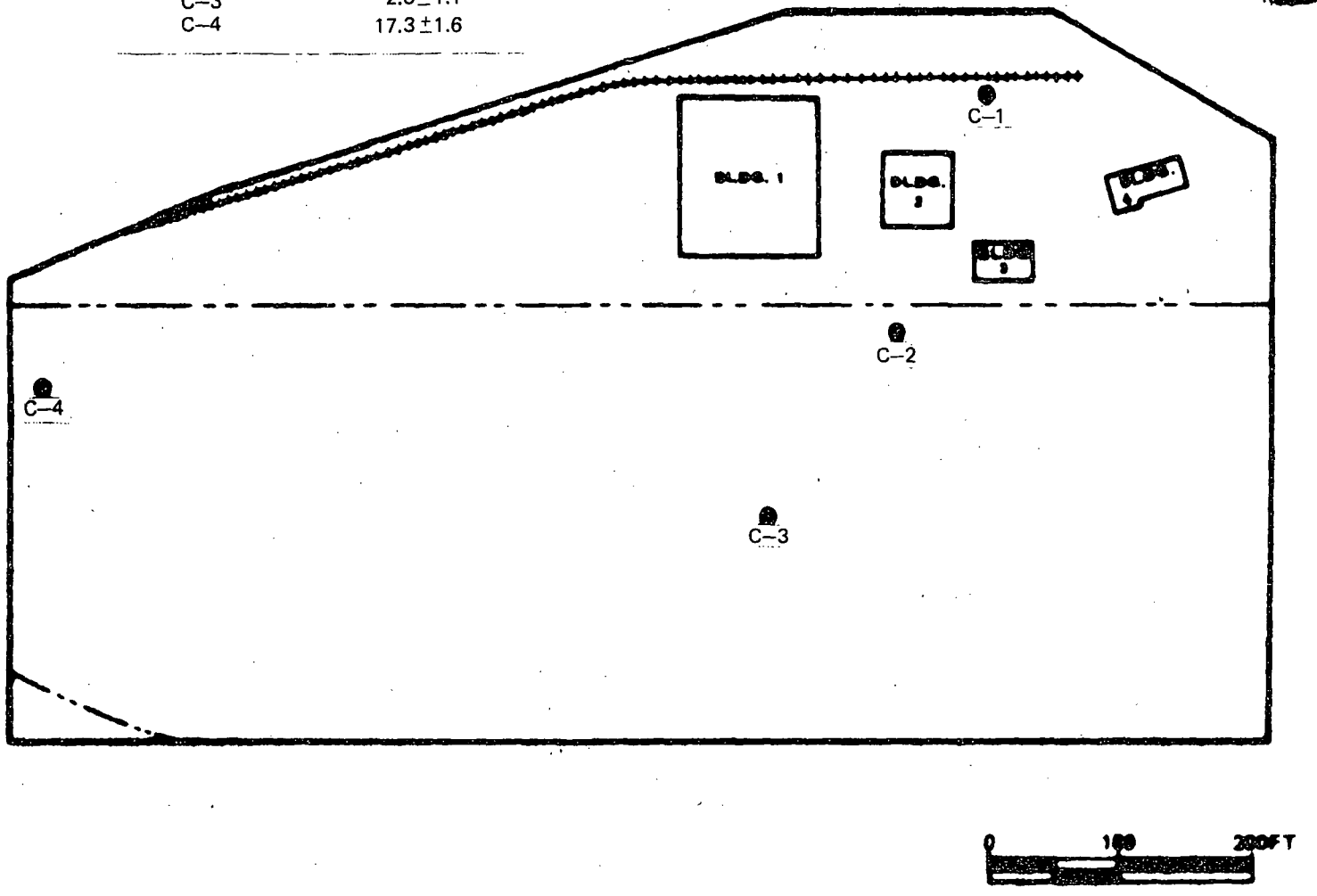
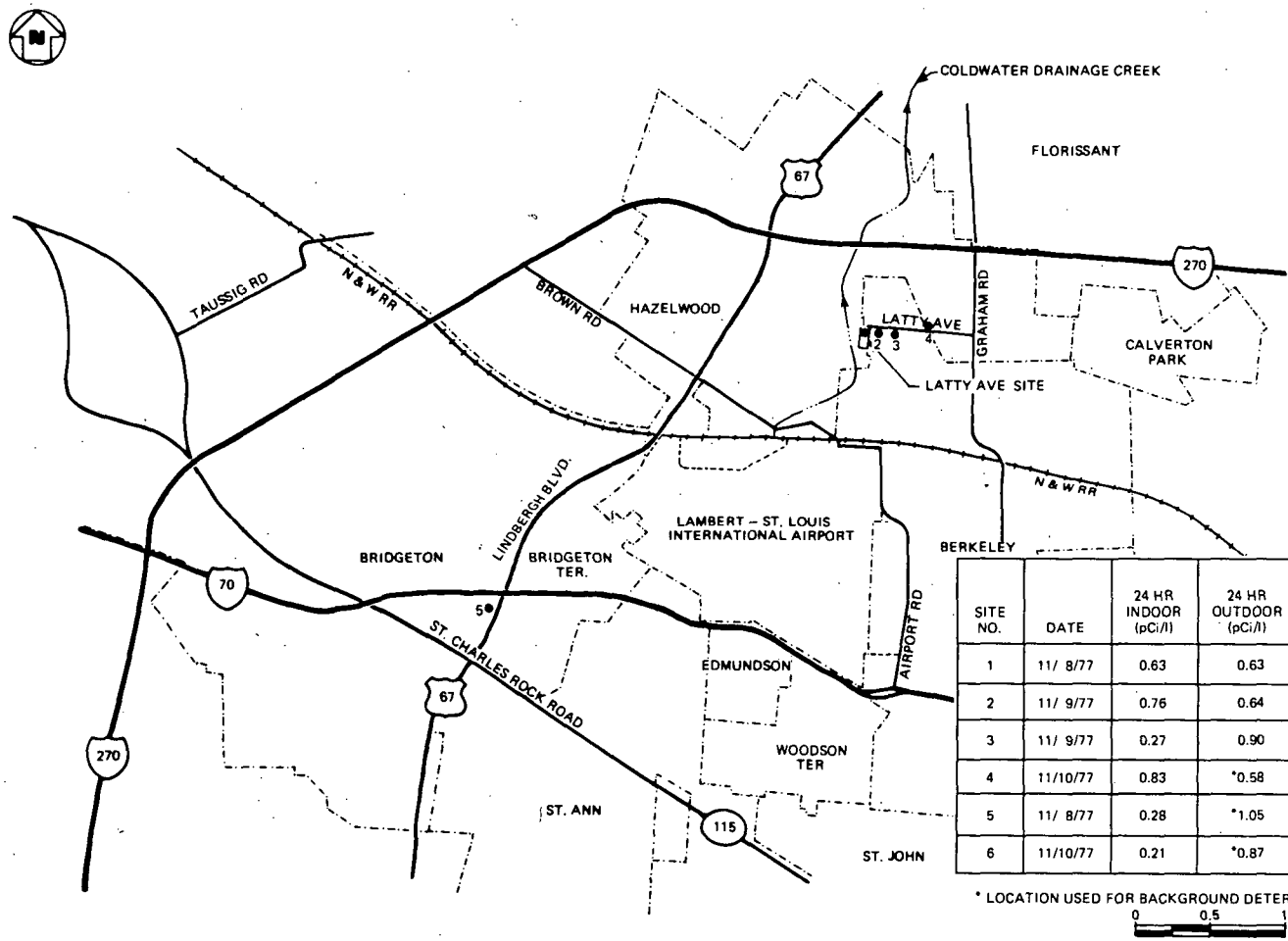


FIGURE 3-2. CHARCOAL CANISTER MEASUREMENT OF RADON FLUX

UC-225



SITE NO.	DATE	24 HR INDOOR (pCi/l)	24 HR OUTDOOR (pCi/l)	AVERAGE WIND SPEED (KNOTS)	AVERAGE WIND DIRECTION	LOCATION
1	11/ 8/77	0.63	0.63	12	S	BLDG 3 ON SITE
2	11/ 9/77	0.76	0.64	16	W	0.1 MI E
3	11/ 9/77	0.27	0.90	16	W	0.2 MI E
4	11/10/77	0.83	*0.58	14	NW	0.4 MI E
5	11/ 8/77	0.28	*1.05	14	S	3 MI SW
6	11/10/77	0.21	*0.87	14	NW	6 MI W

* LOCATION USED FOR BACKGROUND DETERMINATION

0 0.5 1 MI

FIGURE 3-3. RADON CONCENTRATION IN VICINITY OF SITE

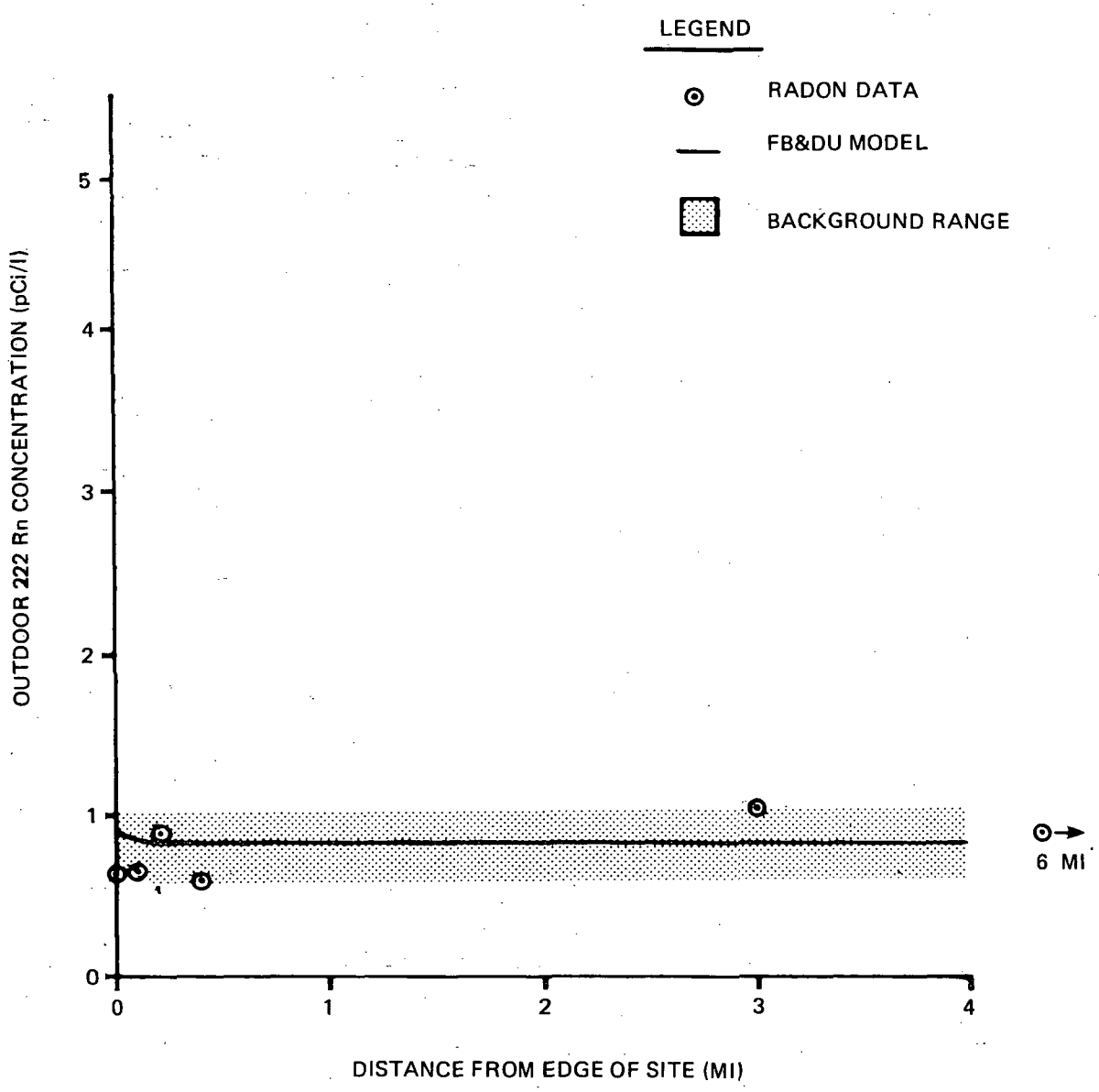


FIGURE 3-4. REDUCTION OF OUTDOOR 222Rn CONCENTRATION WITH DISTANCE FROM THE SITE

UC-228

NOTE: NUMBERS SHOWN ARE GROSS GAMMA RATES IN μR/HR, 1m ABOVE SURFACE

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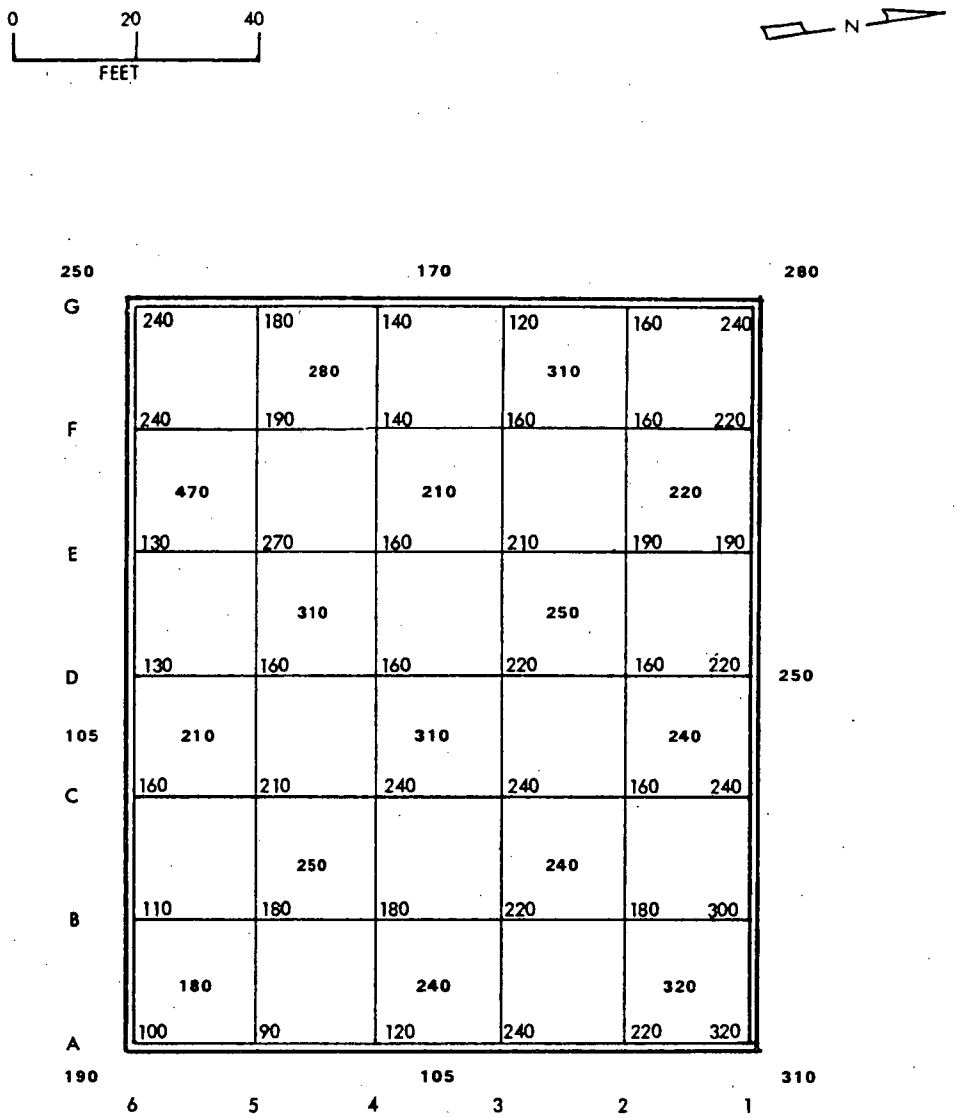


FIGURE 3-5. EXTERNAL GAMMA RADIATION IN AND AROUND BUILDING NO.1

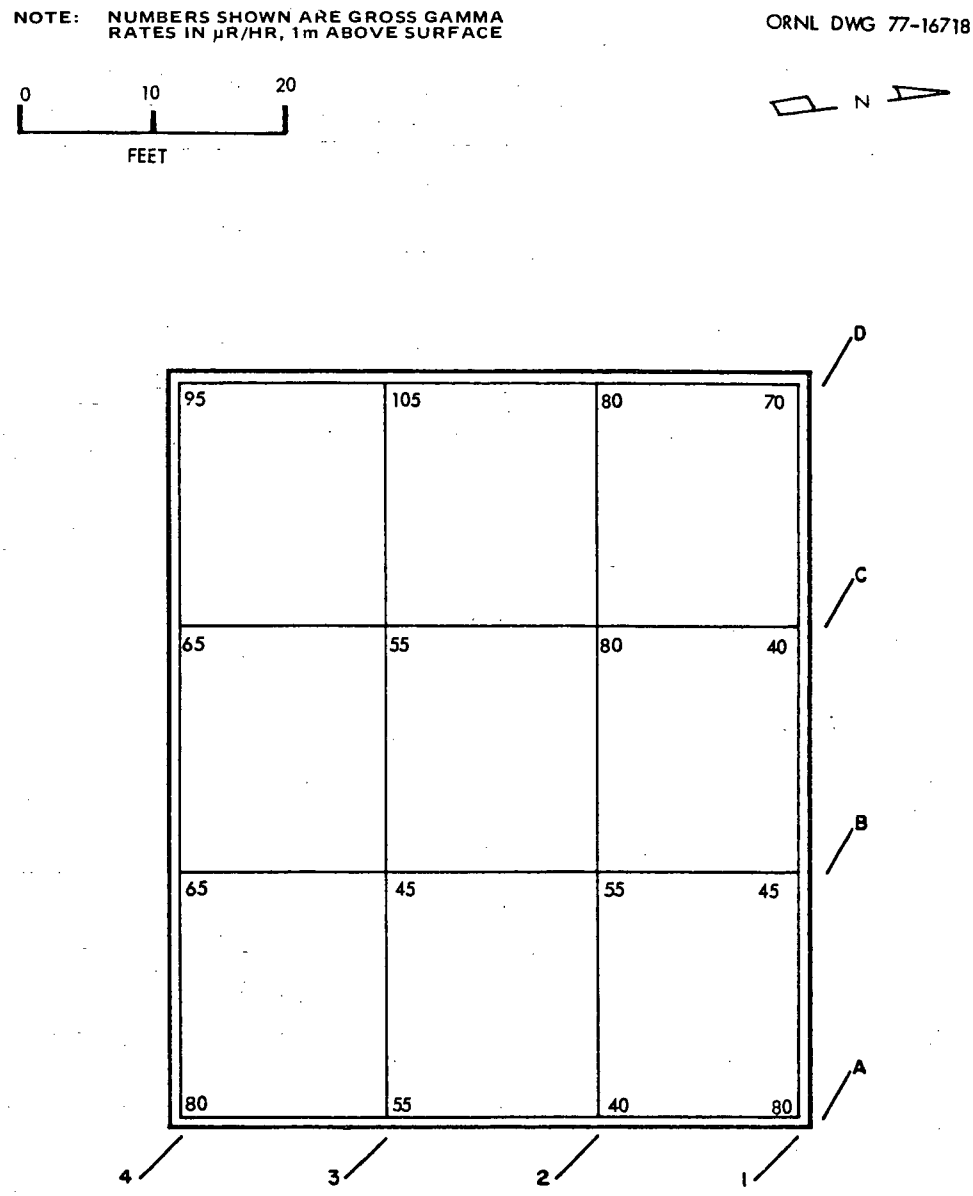


FIGURE 3-6. EXTERNAL GAMMA RADIATION IN BUILDING NO. 2

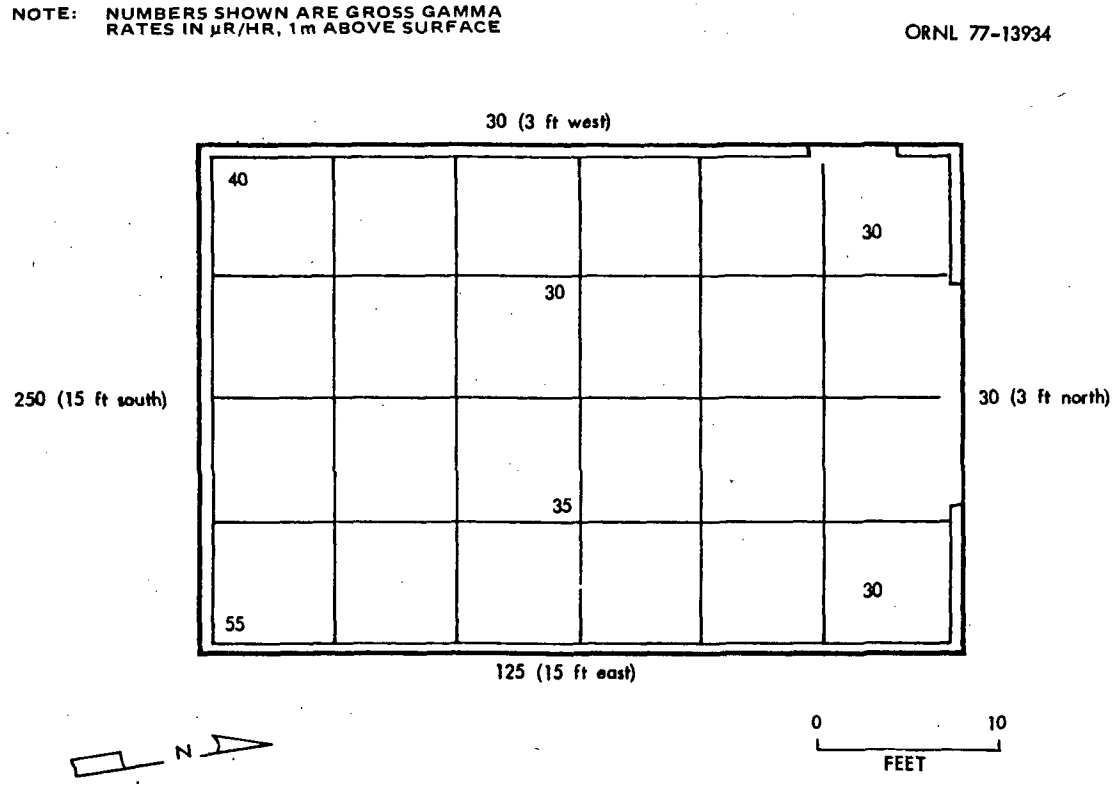


FIGURE 3-7. EXTERNAL GAMMA RADIATION IN AND AROUND BUILDING 3

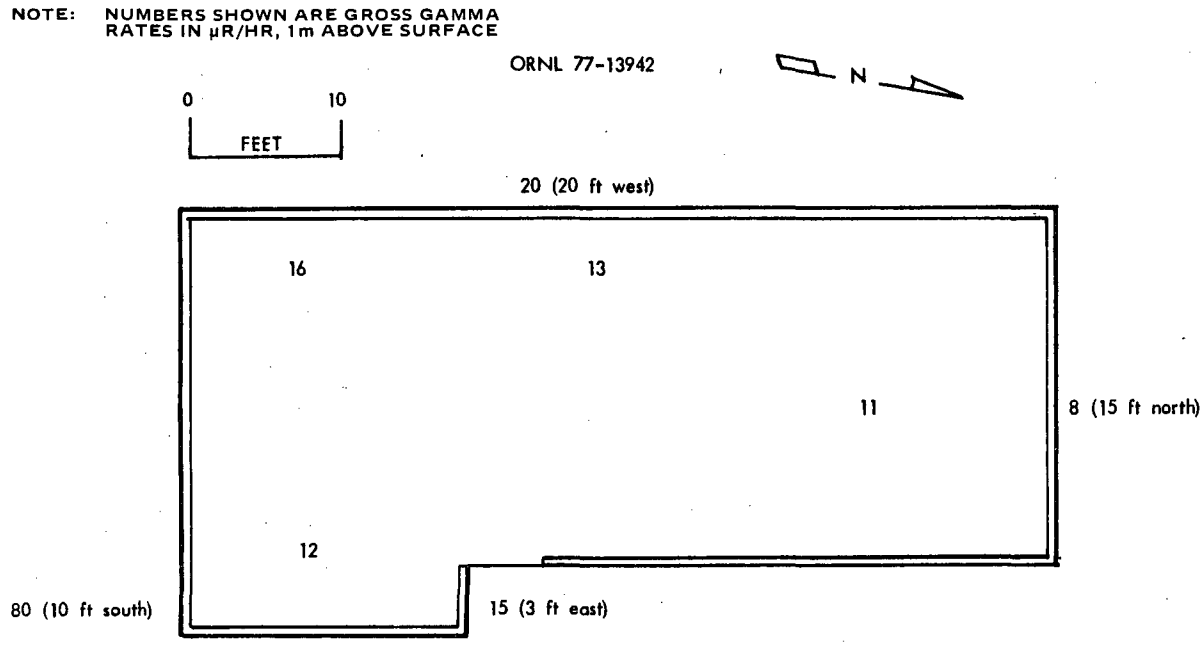


FIGURE 3-8. EXTERNAL GAMMA RADIATION IN AND AROUND BUILDING 4

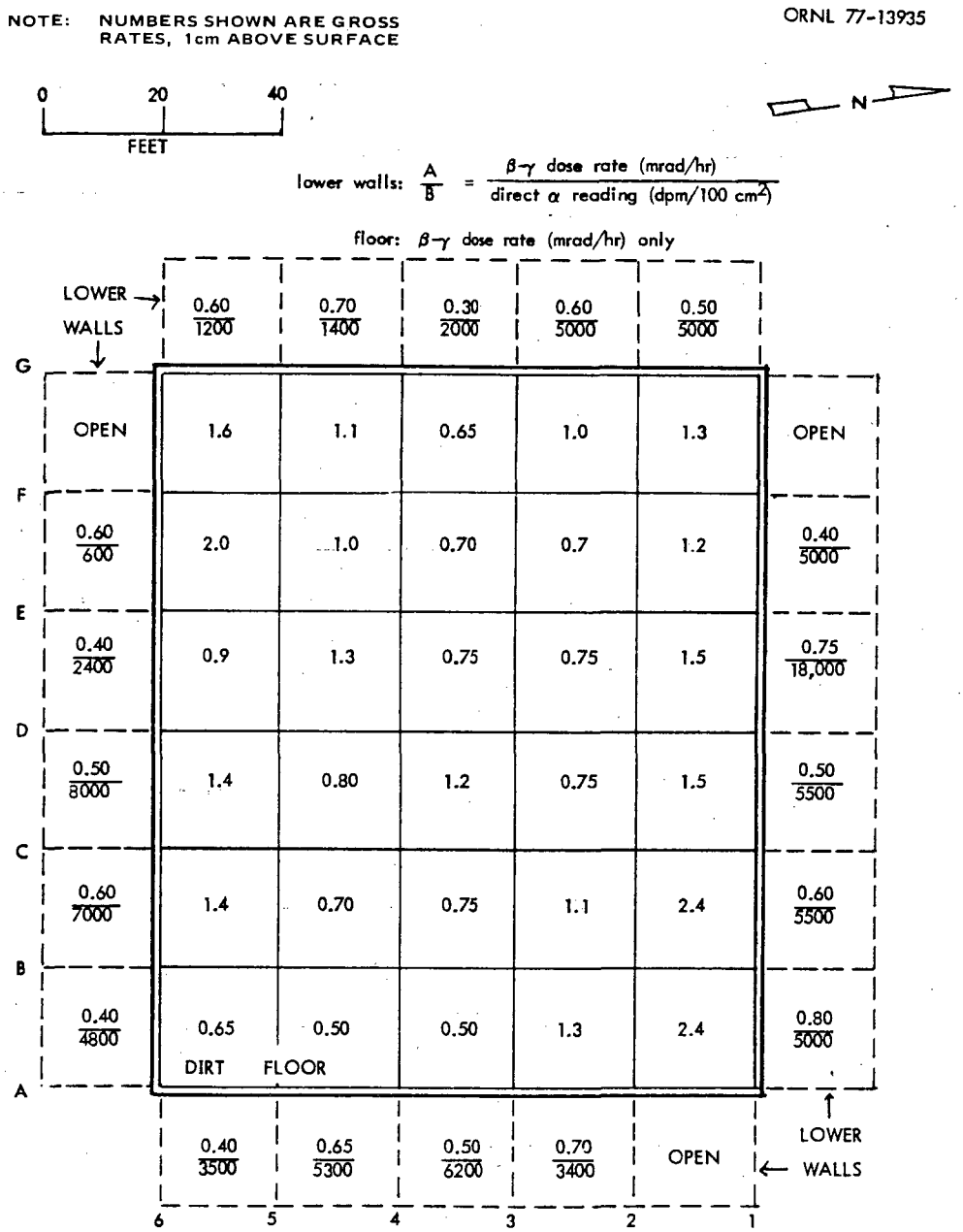


FIGURE 3-9. MAXIMUM BETA-GAMMA DOSE RATE AND DIRECT ALPHA READINGS, BUILDING 1

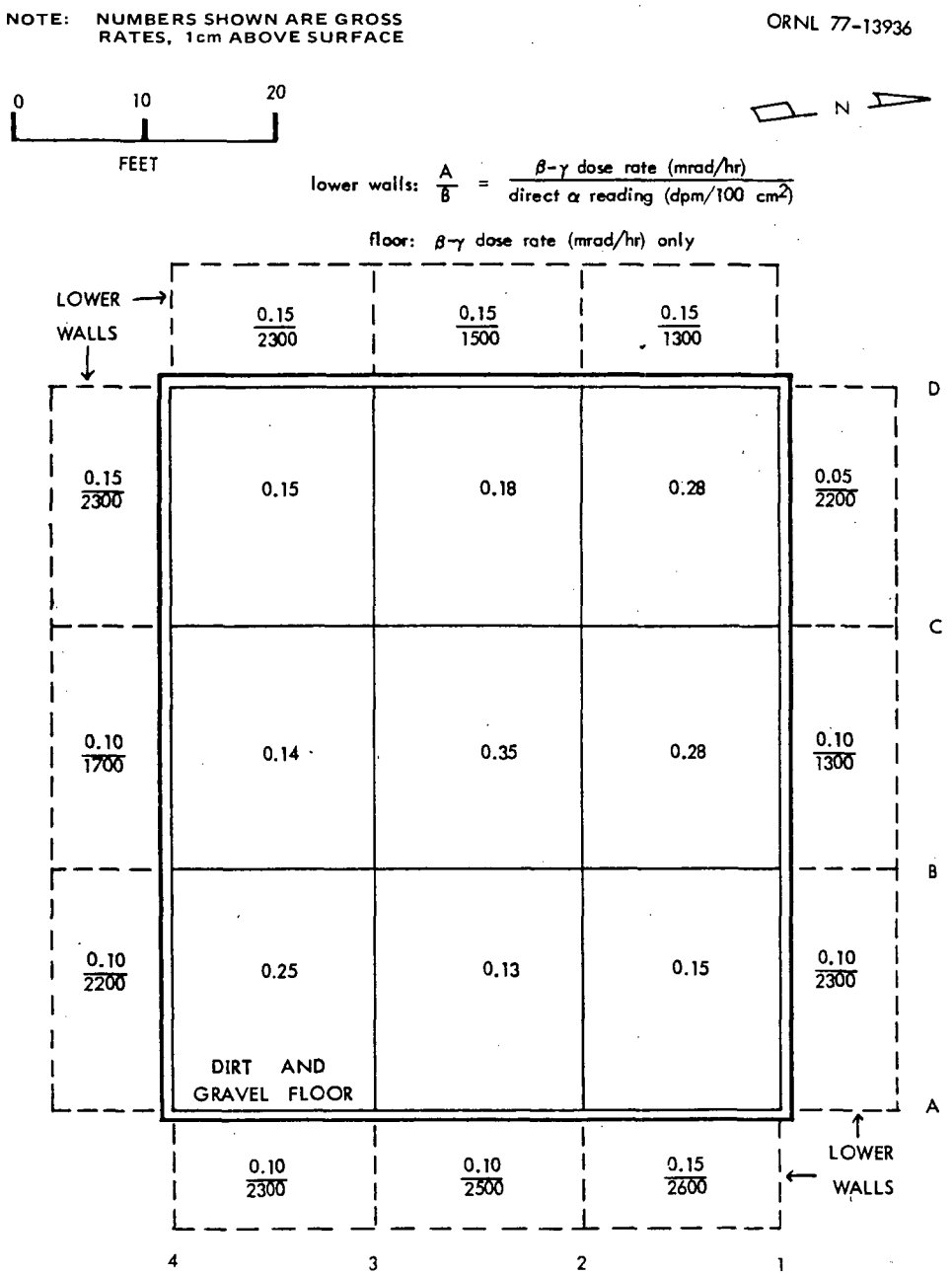


FIGURE 3-10. MAXIMUM BETA-GAMMA DOSE RATE AND DIRECT ALPHA READINGS, BUILDING 2

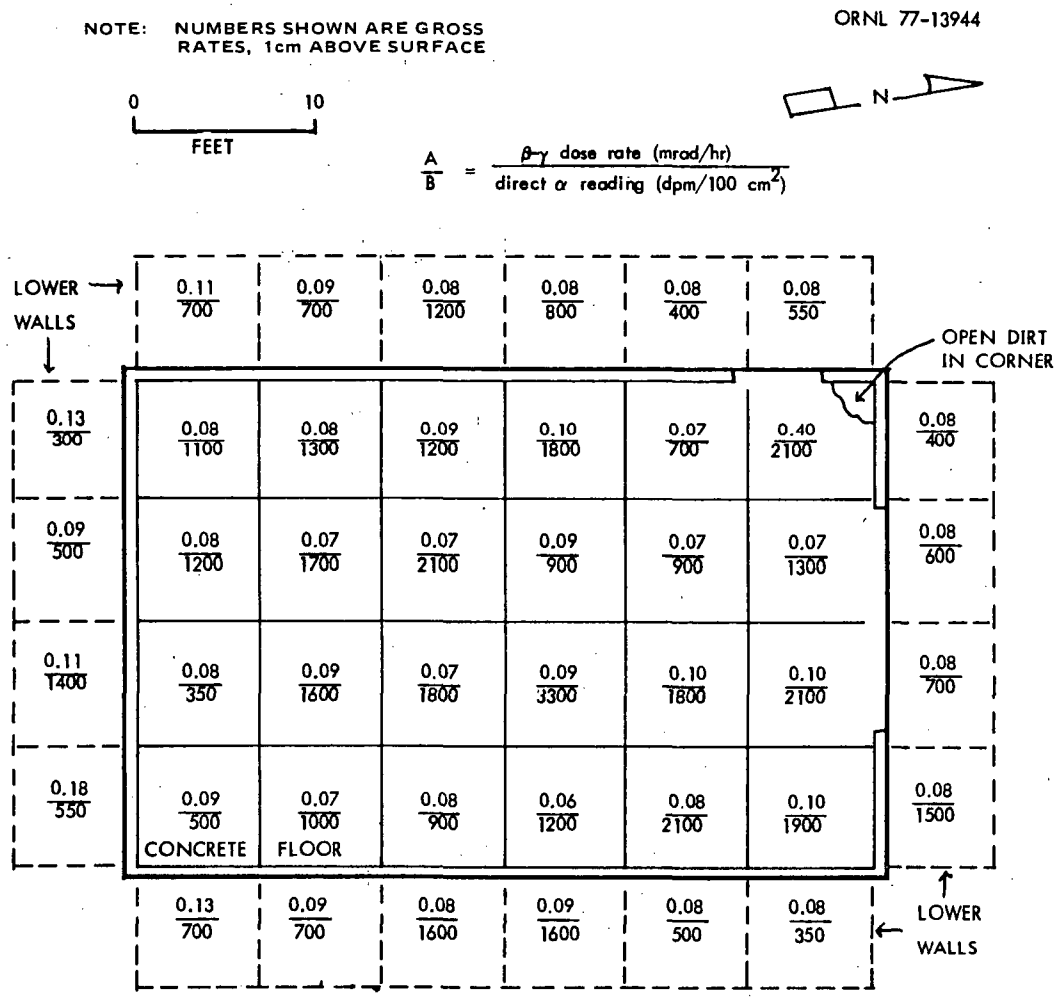


FIGURE 3-11. MAXIMUM BETA-GAMMA DOSE RATE AND DIRECT ALPHA READINGS, BUILDING 3

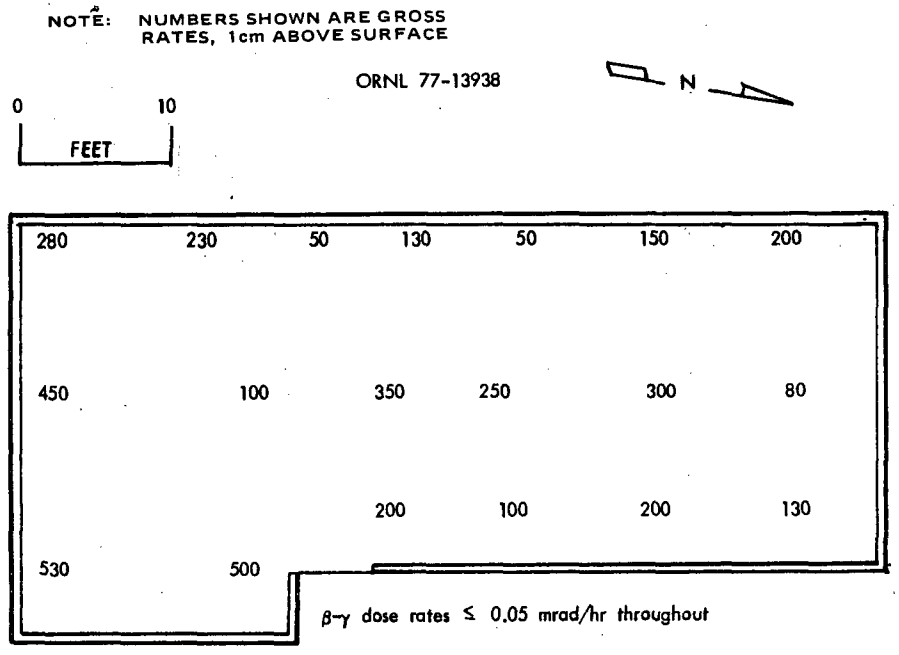


FIGURE 3-12. DIRECT ALPHA READINGS, BUILDING 4

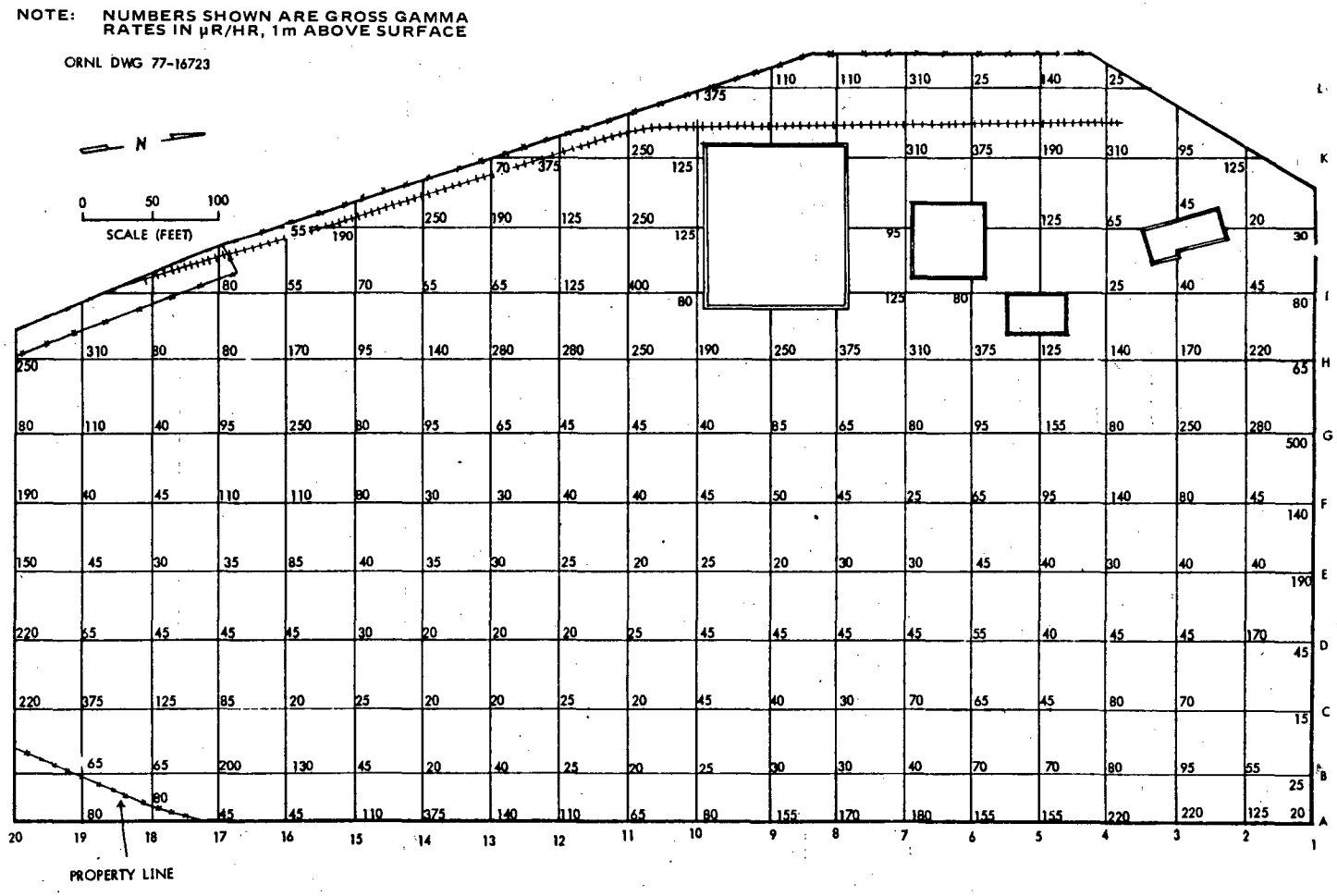


FIGURE 3-13. EXTERNAL GAMMA RADIATION ON SITE

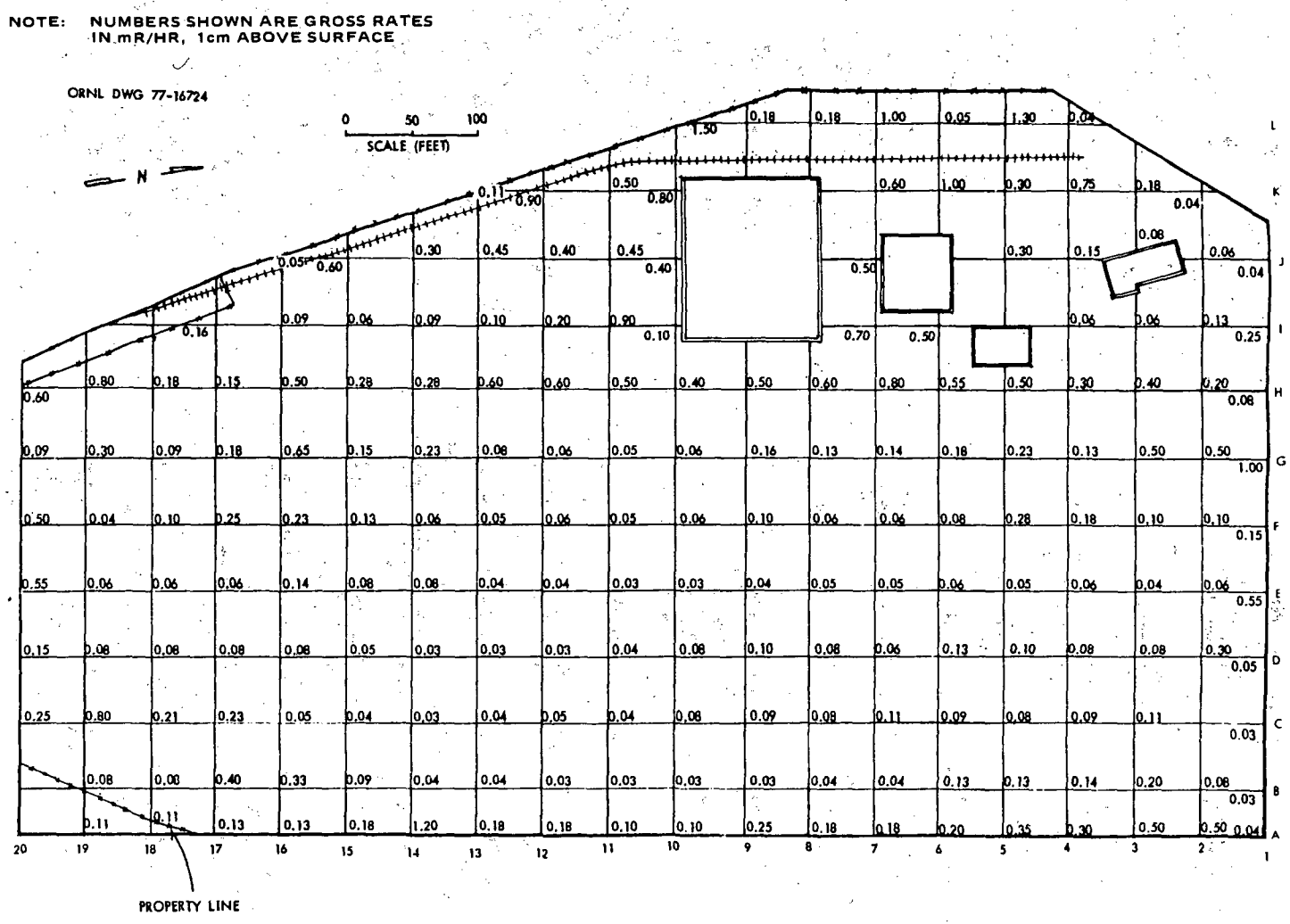


FIGURE 3-14. BETA-GAMMA DOSE RATES ON SITE

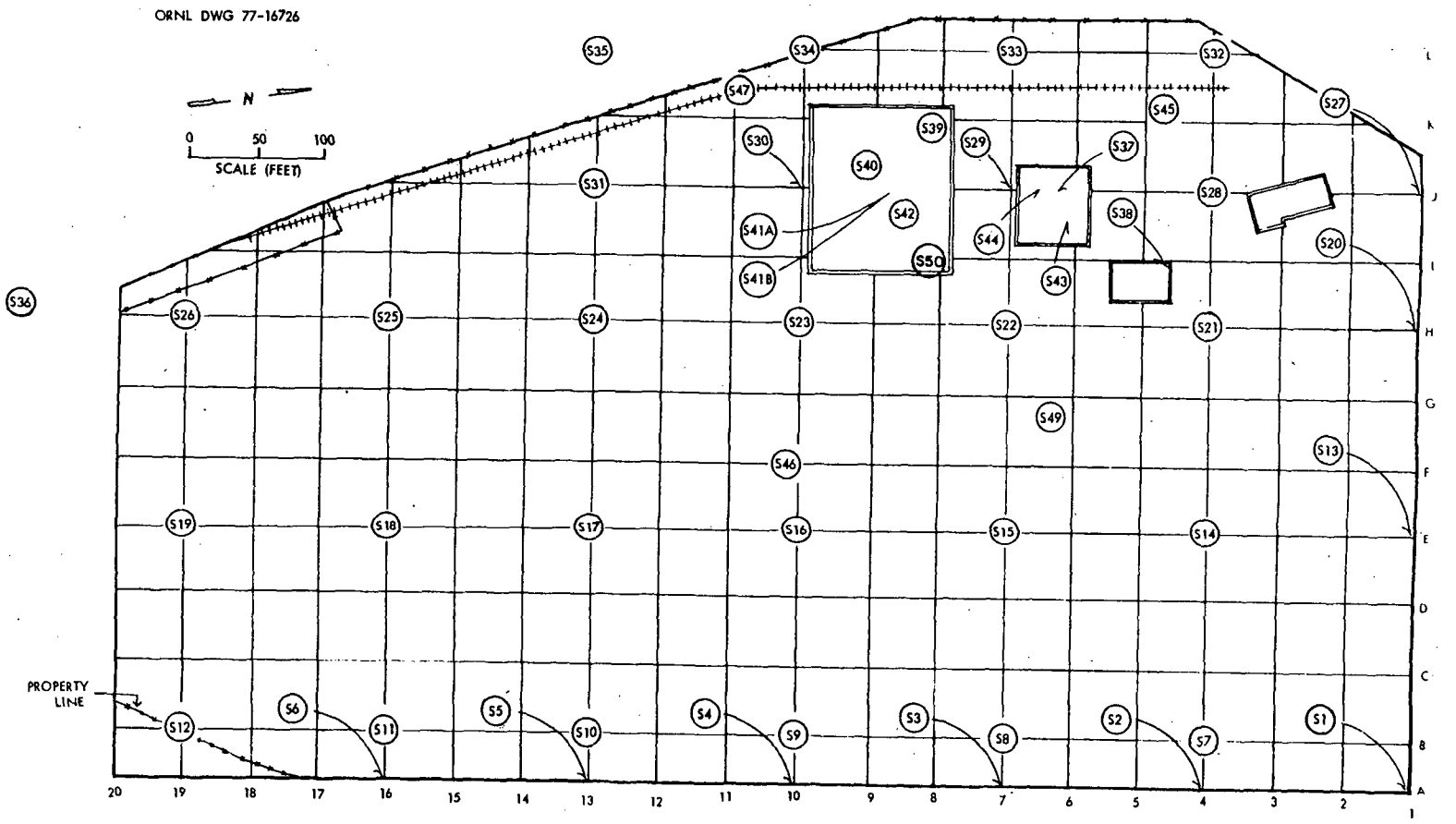


FIGURE 3-15. SURFACE SOIL SAMPLE LOCATIONS

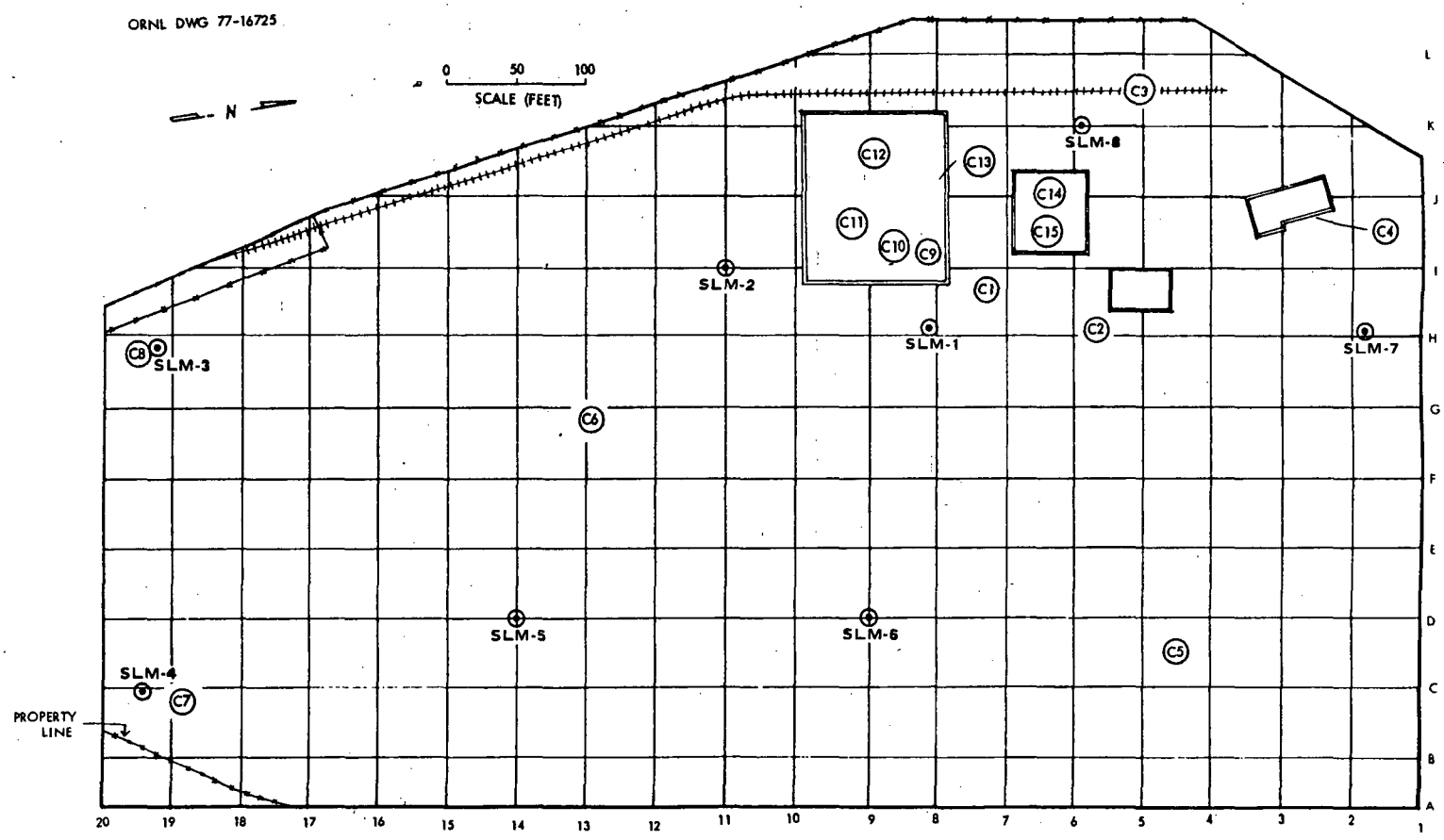
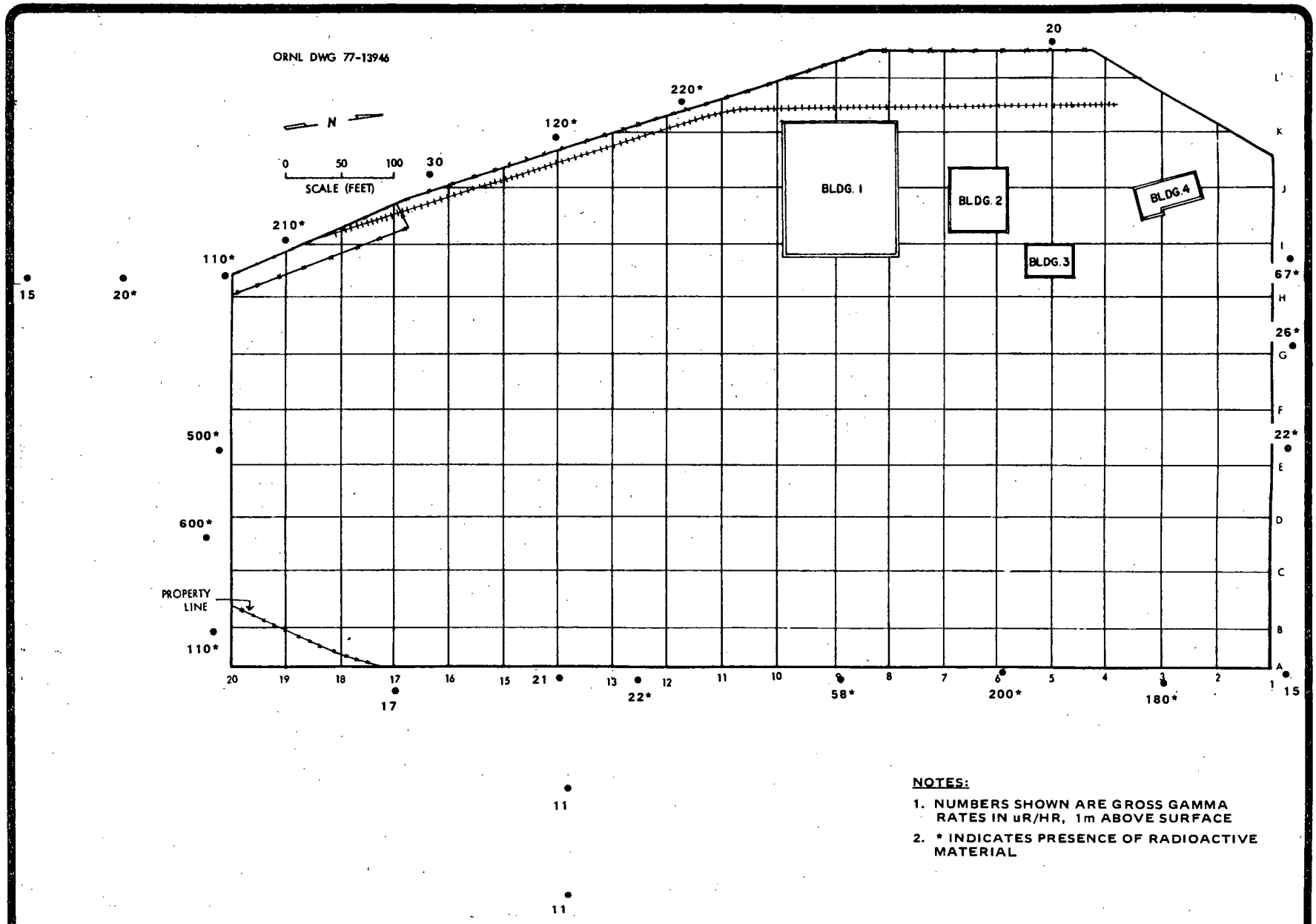


FIGURE 3-16. CORE HOLE LOCATIONS



NOTES:

1. NUMBERS SHOWN ARE GROSS GAMMA RATES IN uR/HR, 1m ABOVE SURFACE
2. * INDICATES PRESENCE OF RADIOACTIVE MATERIAL

FIGURE 3-17. EXTERNAL GAMMA RADIATION OFF SITE

UC 225

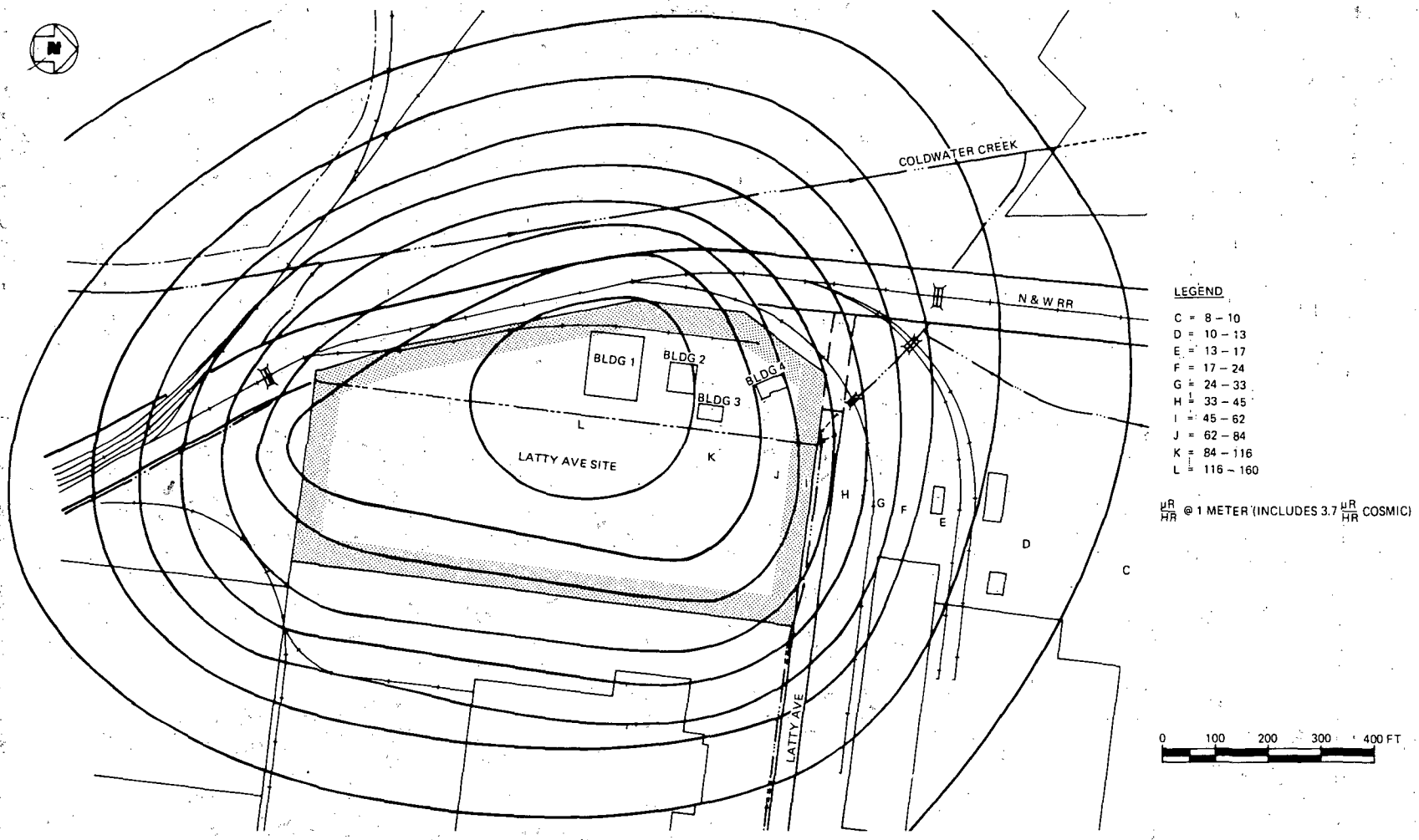


FIGURE 3-18. AERIAL GAMMA RADIATION SURVEY

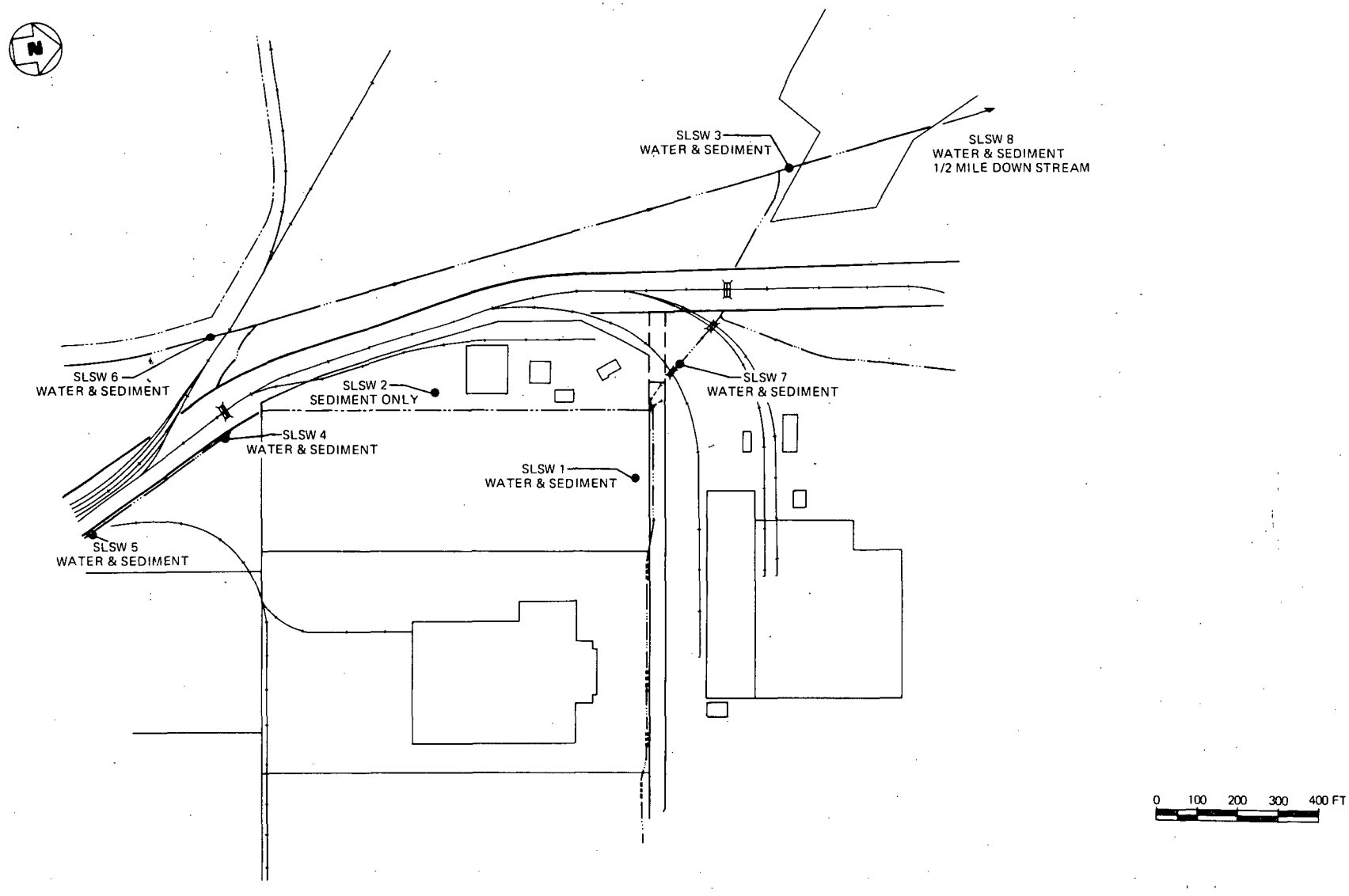


FIGURE 3-19. WATER AND SEDIMENT SAMPLE LOCATIONS

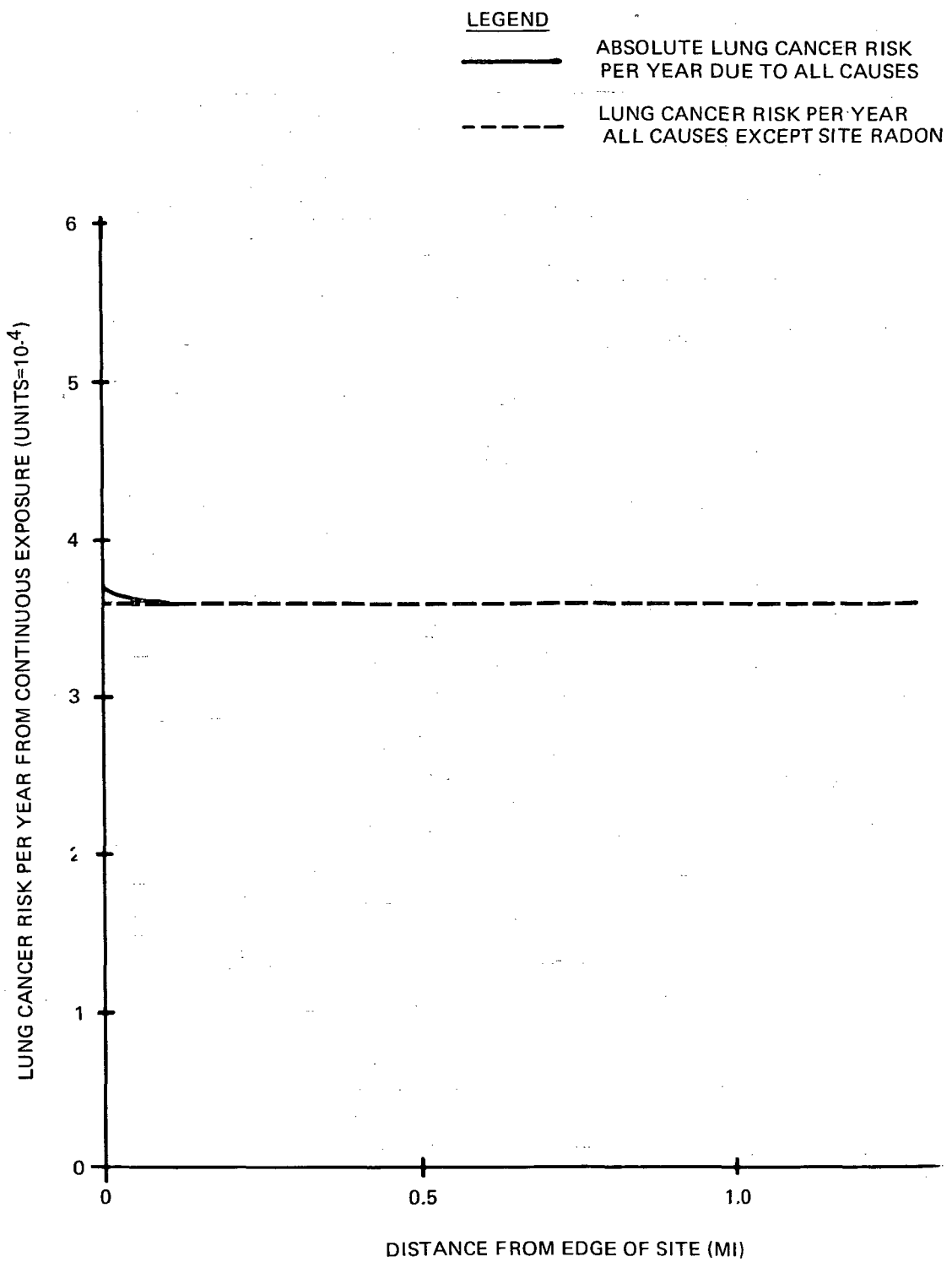


FIGURE 3-20. LUNG CANCER RISK FROM CONTINUOUS EXPOSURE TO RADON DIFFUSION

UC—225

TABLE 3-1

NOTATIONS AND ABBREVIATIONS USED IN CHAPTER 3

Isotope - A particular type of element, differing by nuclear characteristics, identified by the atomic mass number given after the element name, e.g. radium-226.

Isotope Abbreviations:

^{238}U = Uranium-238

^{234}Th = Thorium-234

^{232}Th = Thorium-232

^{234}Pa = Protactinium-234

^{226}Ra = Radium-226

^{222}Rn = Radon-222

^{218}Po = Polonium-218

^{214}Pb = Lead-214

^{214}Bi = Bismuth-214

^{40}K = Potassium-40

Radiations:

alpha particle - helium nucleus; easily stopped with thin layers of material, all energy deposited locally.

beta particle - electron; penetrates about 0.2 g/cm^2 of material.

gamma rays - electromagnetic radiation; similar to X-rays, and highly penetrating.

Half-Life ($T_{1/2}$) - time required for half the radioactive atoms to decay.

TABLE 3-1 (Cont)

Working Level (WL)	- measure of potential alpha energy per liter of air from any combination of short-lived radon daughters (1 WL = 1.3×10^5 MeV of alpha energy).
One Working Level Month (WLM)	- WLM-Exposure to air containing a RDC of 1 WL for a duration of 170 hr.
Roentgen (R)	- that quantity of gamma radiation which yields a charge deposition of 2.58×10^{-4} coul/kg air. This is equal to the energy deposition of 88 ergs/g of dry air or 93 ergs/g of tissue.
μR/hr	- 10^{-6} Roentgen/hr.
Rad	- energy deposition of 100 ergs/g of material
Picocurie (pCi)	- unit of activity (1 pCi = 0.037 radioactive decays/sec or 2.2/min).
MeV	- unit of energy - 1 MeV = 1.6×10^{-6} erg.
Rem	- unit of energy deposition in man. 1 rem = 1 rad x quality factor. The quality factor = 20 for alpha particles.

TABLE 3-2

ANALYSES OF SURFACE SOIL SAMPLES[1]

Location[a]	^{226}Ra (pCi/g)	^{238}U (pCi/g)	^{227}Ac (pCi/g)	^{232}Th (pCi/g)
S1	6.2	7.2	5.8	--[b]
S2	230	310	320	--
S3	35	52	--	--
S4	700	1000	640	--
S5	14	21	7.1	2.1
S6	14	4.6	9.2	1.8
S7	69	100	78	--
S8	88	25	37	--
S9	3.3	5.5	1.0	1.3
S10	1.6	--	--	--
S11	5.6	--	--	--
S12	350	61	130	7.7
S13	830	810	1200	--
S14	6.3	7.8	--	--
S15	--	--	--	--
S16	1.4	0.7	--	1.2
S17	3.1	3.4	--	--
S18	89	--	--	--
S19	14	7.4	7.0	1.3
S20	250	26	--	--
S21	20	--	21	1.6
S22	160	--	--	--
S23	160	240	93	--
S24	220	426	180	--
S25	310	--	200	--
S26	370	--	--	--
S27	4.3	2.3	--	--
S28	130	--	--	--
S29	54	647	35	--
S30	130	--	104	--
S31	170	--	--	--
S32	2.2	4.2	1.0	1.2
S33	8.9	--	--	--
S34	250	330	--	--
S35	1300	--	--	--
S36	2.4	--	--	--
S37	80	84	47	--
S38	82	3.1	120	--
S39	430	860	530	--
S40	320	550	370	--
S41A	320	--	370	5.2
S41B[c]	240	--	240	8.6
S42	190	420	230	--
S43	16	--	10	0.44
S44	28	--	16	0.6
S45	2700	--	1300	--

TABLE 3-2 (Cont)

Location[a]	^{226}Ra (pCi/g)	^{238}U (pCi/g)	^{227}Ac (pCi/g)	^{232}Th (pCi/g)
S46	3.0	--	1.3	1.3
S47	470	530	390	--
S48[d]	120	--	110	4.5
S49	--	210,000	--	--
S50	540	--	700	--

[a]Shown in Figure 3-15.

[b]-- = concentration not determined.

[c]Sample taken at depth of 6 to 9 in.

[d]Sample taken from boots of surveyor who had walked in area shown in Figure 2-2, Chapter 2.

TABLE 3-3

ANALYSES OF SUBSURFACE SOIL SAMPLES[1]

Location[a]	Depth (in.)	^{226}Ra (pCi/g)	^{238}U (pCi/g)	^{227}Ac (pCi/g)	^{232}Th (pCi/g)
C1	0-6	58	--[b]	41	2.6
	6-12	95	--	--	--
	12-18	130	--	97	8.6
	18-24	2.1	--	--	1.2
C2	0-6	270	--	140	6.3
	6-12	54	--	22	1.1
	12-18	1.3	--	--	1.0
	18-24	2.2	--	--	--
C3	0-6	1500	350	--	--
	6-12	25	--	17	1.0
	12-18	12	--	--	--
C4	0-6	8.8	--	9.1	1.5
	6-12	1.7	--	--	--
	12-18	1.9	--	--	--
	18-24	--	--	--	--
C5	0-6	69	--	34	3.1
	6-12	43	--	--	--
	12-18	--	--	--	--
	18-24	49	150	37	3.5
C6	0-6	2.0	--	--	1.3
	6-12	12	--	--	--
	12-18	1.9	--	--	--
	18-24	1.0	--	--	1.3
C7	0-6	1100	--	440	16
	6-12	820	--	--	--
	12-18	11	--	4.1	--
	18-24	3.9	6.8	--	1.3
C8	0-6	29	830	330	11
	6-12	44	--	32	2.7
	12-18	2.4	9.8	1.0	1.3
	18-24	--	--	--	--
C9	0-6	50	70	--	--
C10	0-6	530	--	540	24
	6-12	6.2	670	4.6	1.3
C11	0-6	30	--	22	1.1
	6-12	1.5	1.8	--	1.0
	12-18	1.5	1.7	--	0.84
	18-24	1.4	1.8	0.91	1.1
C12	0-6	72	6.4	85	--
	6-12	3.0	2.6	--	0.95
	12-18	1.6	23	--	1.3
	18-24	16	42	20	1.4
C13	0-6	140	220	65	4.3
	6-12	240	220	220	--
	12-18	200	3.0	210	2.5
	18-24	3.0	190	--	--

TABLE 3-3 (Cont)

Location[a]	Depth (in.)	^{226}Ra (pCi/g)	^{238}U (pCi/g)	^{227}Ac (pCi/g)	^{232}Th (pCi/g)
C14	0-6	46	75	25	--
	6-12	4.2	4.1	--	1.2
	12-18	3.0	1.9	--	--
	18-24	1.9	2.0	--	1.4
C15	0-6	34	38	240	2.0
	6-12	2.7	1.4	--	1.1
	12-18	1.3	2.4	--	1.2
	18-24	1.7	10	--	--

[a]Shown in Figure 3-16.

[b]-- = concentration not determined.

TABLE 3-4

CONCENTRATIONS OF ^{210}Pb, ^{226}Ra, AND ^{230}Th IN WATER AND SEDIMENT SAMPLES

Sample Location	Water			Sediment		
	^{210}Pb (pCi/ml)	^{226}Ra (pCi/ml)	^{230}Th (pCi/ml)	^{210}Pb (pCi/g)	^{226}Ra (pCi/g)	^{230}Th (pCi/g)
Drainage Ditch at southwest corner of property	0.007±0.001	0.002±0.001	0.002±0.001	55.9±8.1	4.19±1.35	91.9±6.3
Coldwater Creek, 1 mile downstream from site	<0.001	<0.001	0.0005±0.0005	--	0.063±0.153	0.252±0.248
Storm sewer at Latty Avenue	0.007±0.003	0.001±0.0005	<0.001	18.0±5.0	<0.014	4.96±0.90
RCG_w (soluble)	0.1	0.03	2.0	--	--	--

TABLE 3-5

ESTIMATED HEALTH IMPACT FROM LATTY AVENUE SITE FOR AN AREA 0-0.75 MILES FROM SITE EDGE

Time Period	Population (Employees)	Population (Residents)	Total Site-Induced RDC Health Effects/yr	Background RDC Health Effects/yr
1970	22,000	660	0.002	0.29
1995 (Static)	22,000	660	0.002	0.29
1995 (Constant Growth[a])	34,000	800	0.003	0.44

25-yr Cumulative Effect	Site-Induced RDC Health Effects	Background RDC Health Effects
Static population	0.05	7.2
Constant growth[a]	0.06	9.9

[a]The number of employees grows by 800/yr for 15 yr then holds constant. The number of residents grows by 14/yr for 10 yr then holds constant.

CHAPTER 3 REFERENCES

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CHAPTER 4

POPULATION AND LAND USE

The Latty Avenue site lies entirely within the city of Hazelwood, St. Louis County, Missouri. Several political entities including planning districts, service districts, municipalities, and other governmental authorities have jurisdiction over the site and its immediate environs. Political jurisdictions and transportation routes are shown in Figure 4-1.

4.1 RECENT HISTORY OF THE AREA

Large estates, established through the early 1800's, in northcentral St. Louis County were first subdivided in the early 1900's. Commercial and industrial establishments were attracted to the area a few years later. Currently the area is continuing to become more industrialized, but at a far slower rate than in the previous decade. The economy and land uses are no longer in transition and consist of a mix of commercial and industrial interests near the site with residential complexes approximately 1 mi east and north from the site.

4.2 POPULATION AND EMPLOYMENT PROJECTIONS

The future demographic and economic conditions of the area can be projected on the basis of past trends and assumptions concerning future economic conditions. The area is primarily industrial and commercial land, and it is not projected that residential populations will increase greatly within 0.75 mi of the site boundary. Residential neighborhoods in the area tend to be of relatively recent construction, with many younger people and very few older people. The residential population within 0.75 mi of the site boundary was estimated in 1970 to be 660 people living in approximately 220 dwelling units.[1] These units presently consist of approximately 60 single-family houses and 25 apartment buildings. In the area beyond the 0.75-mi boundary there are several high-density residential areas of Hazelwood and Berkeley. It is estimated that in 1977 there were 390 single-family houses and 50 apartment buildings in the area between 0.75 mi from the site and 1 mi from the site. This would be approximately 690 dwelling units and a residential population of approximately 2,300.

Employment figures and projections for the area are highly dependent on the individual firms that are included in such a

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[1] See end of chapter for references.

survey. There were approximately 22,000 people employed at firms located within 0.75-mi distance from the site boundary.[1] Approximately 60,000 people are employed in the McDonnell Douglas-airport-Hazelwood area. This complex is the second largest employment center in the entire St. Louis metropolitan area.[2] Of these, 25,000 are employed by McDonnell-Douglas (approximately 1 mi from the site boundary), 5,100 are employed at the Lambert-St. Louis International Airport (2 mi from the site), and 3,100 are employed at the Ford Motor Company assembly plant (0.5 mi from the site).

Based on planning assumptions by municipal, county, and regional governments, the projection is for little potential for major residential population growth within 0.75 mi from the site boundary. This is because little vacant residential land remains to be developed.[3] Therefore, over the next 25 yr it is projected that the residential population will grow by no more than 50 dwelling units (165 residents approximately) and shrink by not more than 10% (20 dwelling units or 70 residents approximately). The East-West Gateway Coordinating Council projects a population of approximately 800 within 0.75 mi of the site in 1995. Residential population projections and employment outlook are shown in Figure 4-2.

Future employment is more difficult to project than population because a single large industrial complex can increase employment in the immediate area by as much as 20%. In the area within 0.75 mi from the boundary of the site the 1970 employee density was 21.53 people/industrial acre. There were 136 acres of vacant industrial property in the area, as well as considerable under-utilized space. If the total industrial acreage (1,130 acres) were utilized at a density of 30 people/acre, the total number of employees within 0.75 mi of the boundary of the site would be approximately 34,000. It would be unlikely that this saturation level would be reached in less than 15 yr. Conversely, if a severe depression affected several industries in the area, a decrease in the employment force up to 10% could last for several years. This high and low range is shown in the projections in Figure 4-2. The East-West Coordinating Council projection of a work force of 23,000 by 1995 is within this range of projected populations.

4.3 LAND USE

As shown in Figure 4-3, the area near the Latty Avenue site is industrial/commercial land. Some of the area near the site lies within the northern section of the town of Berkeley. This industrial area has been slower to develop than other areas of northcentral St. Louis County because of better highway access to other sections of the county. With the completion of State Route 725 as an inner belt route and with the major interchange of Interstate 270 and Lindbergh Boulevard nearby, the area is becoming increasingly more attractive to industry. In 1970, of

the approximately 1,310 acres within 0.75 mi of the site boundary and including the site itself, there were 682 acres in industrial use, 312 acres in commercial use, 174 acres in use as transportation corridors, 136 acres in vacant industrial/commercial land, 93 acres in residential use, and 6 acres in recreational use. The Latty Avenue site is zoned for industrial use and is surrounded by industrial complexes. The probable future land use of the area is therefore for industrial purposes. The land use element of the General Plan discusses future land use of the area:[4]

> Presently vacant or mixed use lots in the area bounded by I-270, proposed inner belt, I-70, and Lindbergh Boulevard will most likely experience the expansion of the facilities of existing major industries such as McDonnell-Douglas.

However, in another section of this report it is indicated that open land which is adjacent to Coldwater Creek and subject to flooding should be kept as open space. The Latty Avenue site would fall under this category.

The presence of radioactive materials on the site has influenced the use of the site, in that industrial development of the property has been slowed because of concerns for human health and safety. However, pressures to use the site as industrial and commercial land will increase with improved highway access to the area and as the remaining vacant industrial land is diminished by development.

4.4 IMPACT OF THE SITE ON LAND VALUES

The 11-acre site, comprising a vacant land parcel and another with 4 structures on it, is located in an area of heavy industrial usage. Virtually all of the land within 0.75 mi of the site's borders is zoned industrial and/or commercial. Approximately 10% of the land is vacant in this same area. Rail spur lines are available to most of the larger industrial parcels in the general area. This, plus the availability and closeness of an excellent freeway and local street system, make the site a desirable piece of land.

Some property adjacent or close to the Latty Avenue site has been assessed as follows (for 1977):

Acreage	Land	Improvements	Totals
18.6	93,460	713,250	806,710
18.5	91,910	678,720	770,630
11.3	53,540	316,480	370,020
4.1	29,750	137,030	166,780
12.0	58,350	Vacant Land	58,350
4.0	28,750	63,420	92,170
27.8	92,740	Vacant Land	92,740

For tax purposes, these segments of land are assessed at 30% by the local governments involved; thus the "book" value of the 11.3 acre parcel not including improvements in the above table would be $178,467.

The western portion of the site, which covers approximately 3.5 acres and has 4 presently unusable structures on it, was sold in July of 1976 for $75,000. This equals a per-square-foot cost of $0.492.

From inquiries to realtors in the area, the current market of price of "clean" vacant land in this vicinity is from $0.75 to $1.00/ft^2. This depends, of course, on the size of the lot, with smaller lots being more expensive, proportionately, per square foot than large lots. Thus, the total 11 acres of the 9200 Latty Avenue site could be worth from $338,000 to $480,000 at current market asking prices.

The presence of the radioactive materials on the site has had no bearing on the demand for, nor the availability or market value of, surrounding properties. The contamination, however, has affected development potential of the site. The city of Hazelwood has placed, at the insistence of the NRC, a moratorium of any development or usage of this site until the matter of the effect of the radioactivity has been determined, and removed as necessary.

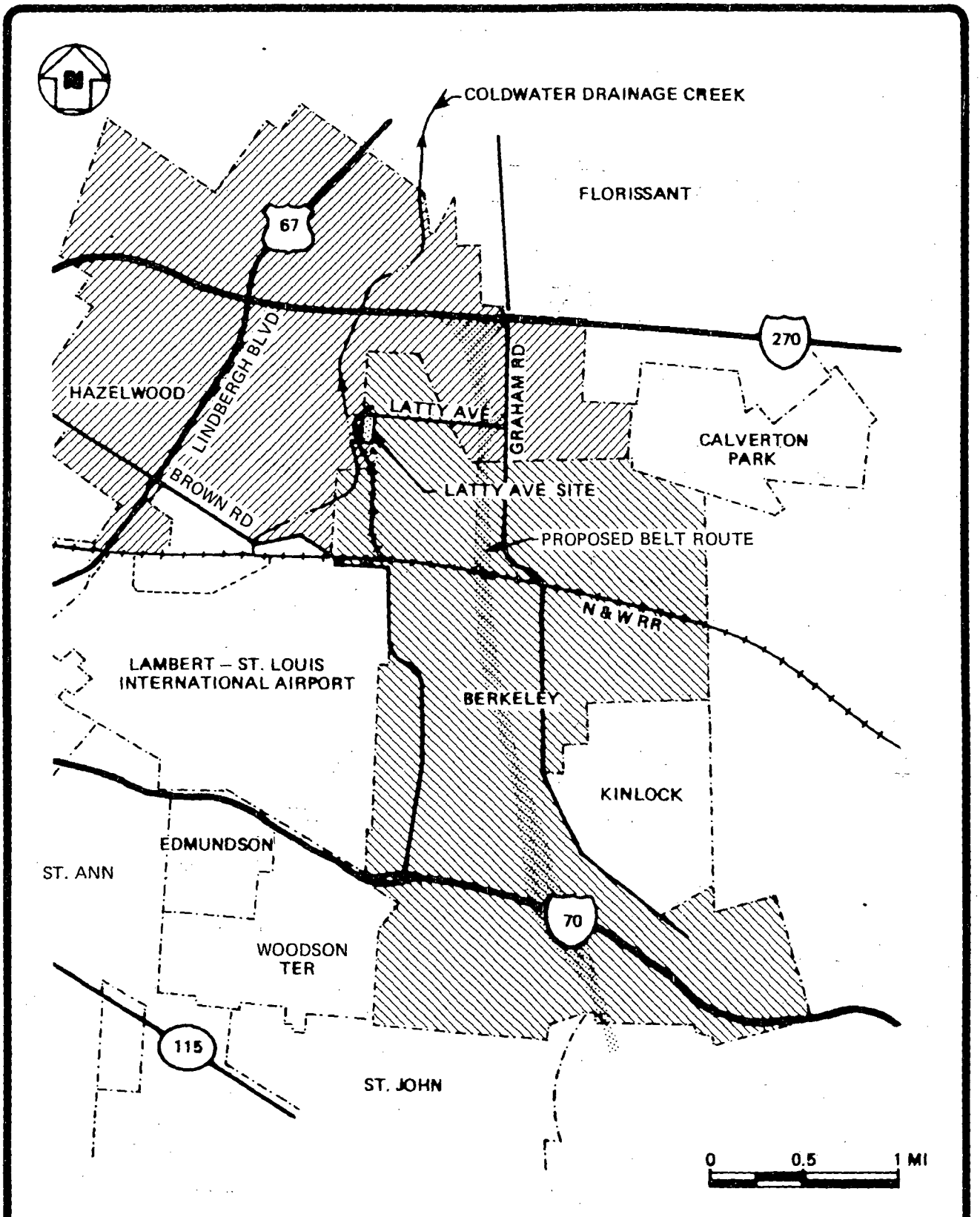


FIGURE 4-1. POLITICAL JURISDICTIONS AND TRANSPORTATION ROUTES

UC 225

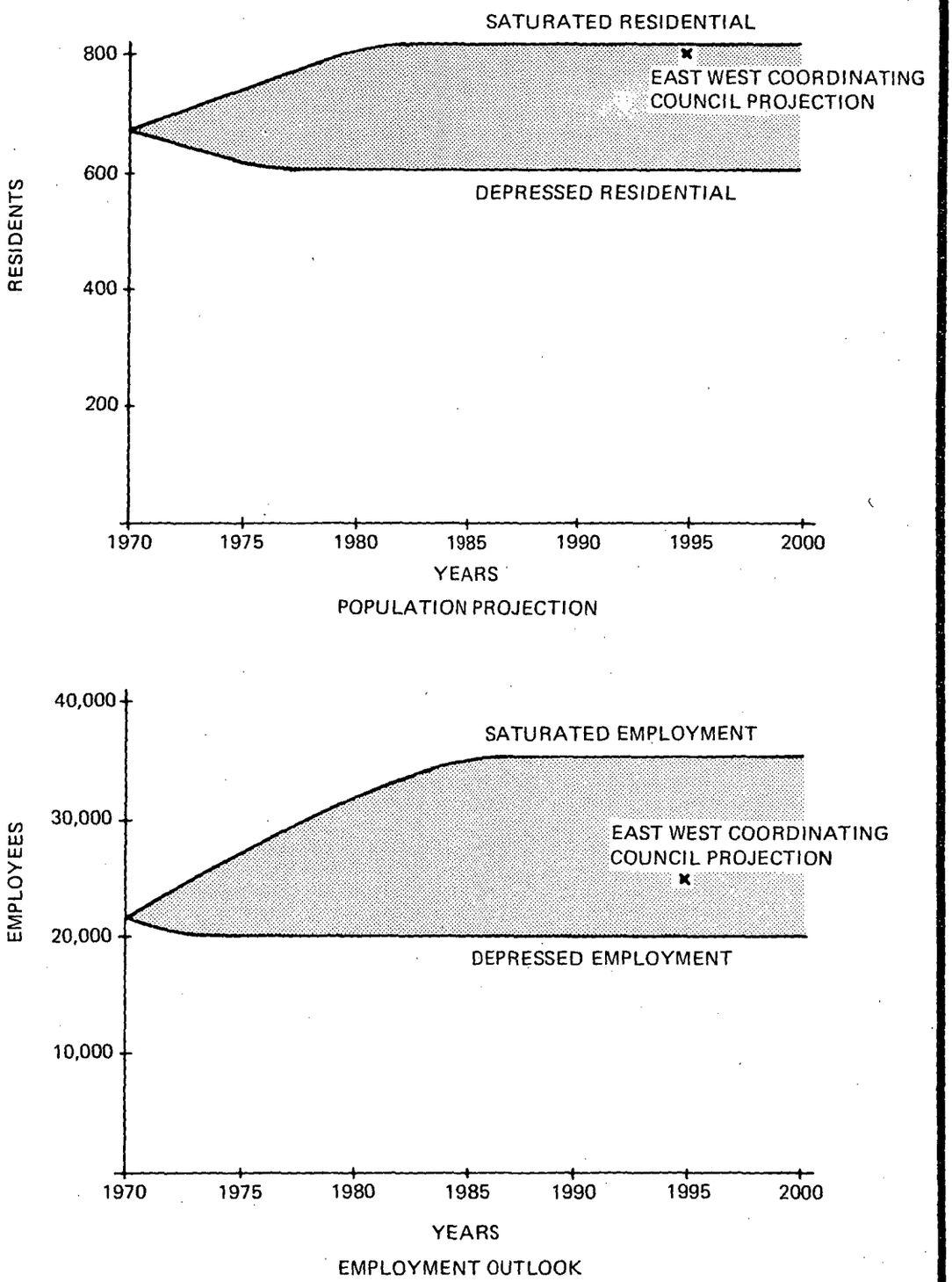
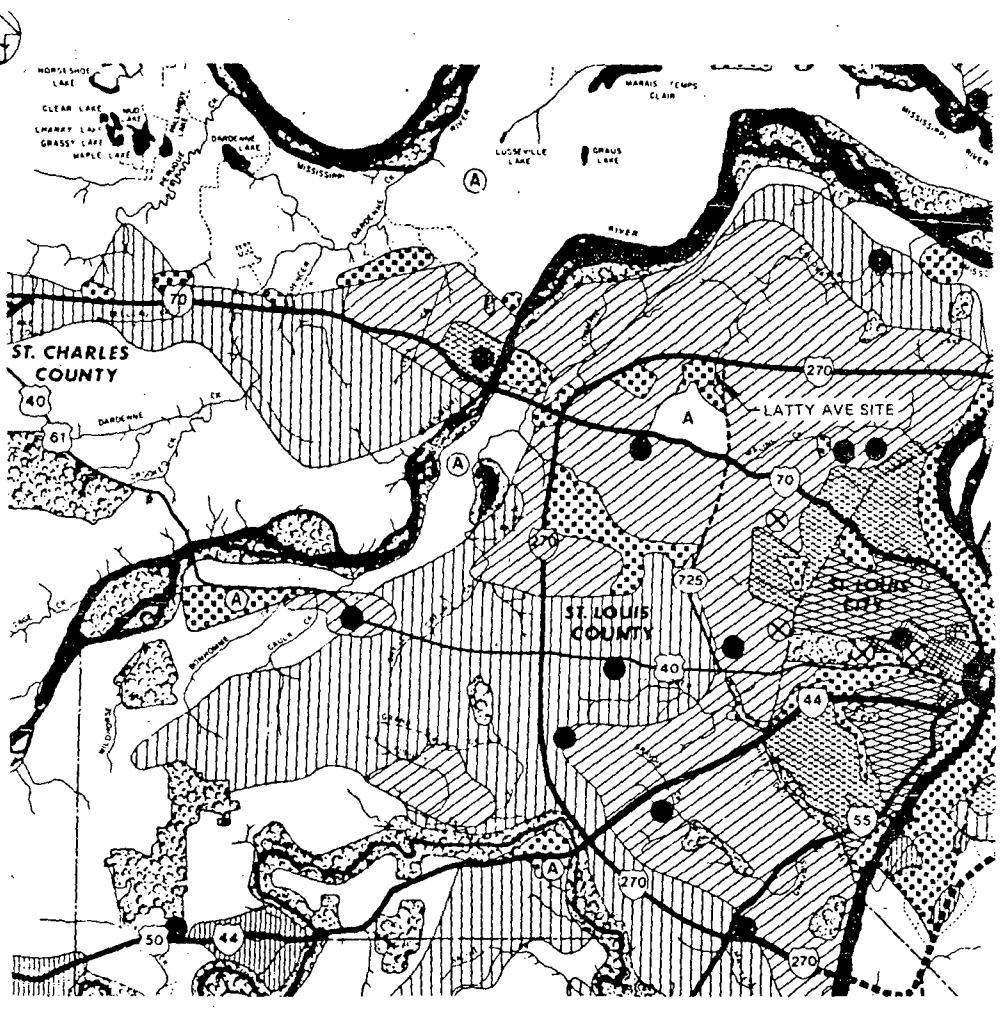


FIGURE 4-2. POPULATION AND EMPLOYMENT OUTLOOK FOR AREA WITHIN 0.75 MI OF SITE



LEGEND

SERVICE AREAS	DENSITIES
URBAN CORE[1]	20 d.u./Ac.
URBAN[1]	10-20 d.u./Ac.
URBAN[1]	5-10 d.u./Ac.
SUBURBAN[1]	3-5 d.u./Ac.
FRINGE[1]	1-3 d.u./Ac.
FREE STANDING[3]	
AGRICULTURAL OTHER RURAL	
REGIONAL COMMERCIAL CORE	
REGIONAL CENTER[4]	
REGIONAL INDUSTRIAL (100 acres and more)	
INSTITUTIONAL[4]	
RECREATIONAL OPEN SPACE[5]	
AIRPORTS[6]	
MAJOR HIGHWAYS	

NOTE:
MAP TAKEN FROM 1977 DRAFT LAND PLAT PREPARED BY EAST-WEST GATEWAY COORDINATING COUNCIL, ST. LOUIS AREA

0 5 10 MI

FIGURE 4-3. GENERALIZED LAND USE MAP

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CHAPTER 5

STABILIZATION COVER FOR LOW-LEVEL RADIOACTIVE MATERIALS

Remedial actions that involve long-term storage require a stabilization cover to prevent the spread of the radioactive materials. Government agencies and private industry have carried out limited research to develop economical and environmentally suitable methods for uranium tailings site stabilization. All present methods, technology, and research data on stabilization that are available were reviewed to determine the best approach to be applied to the Latty Avenue site. In addition, experiments are being conducted to determine the relative effectiveness of various stabilization techniques.

The objective of stabilizing the radioactive materials is to eliminate the pathways to the environment of the radioactive and other toxic particles as described previously in Chapter 3. Ideally, complete stabilization of radioactive materials should permanently eliminate the possibilities of:

(a) Wind and water erosion

(b) Leaching of radioactive materials and other chemicals

(c) Radon exhalation from the materials

(d) Gamma radiation emitted from the materials

5.1 PREVENTION OF WIND AND WATER EROSION

Wind and water erosion can be prevented by chemical stabilization of the surface, complete chemical stabilization, physical (earth) stabilization, vegetative stabilization, or a combination of these methods.

5.1.1 Chemical Stabilization of the Surface

This process involves applying chemicals to the surface of the radioactive materials to form a water- and wind-resistant crust. Chemical stabilizers have been used successfully as a temporary protection on portions of dikes and uranium tailings ponds which have dried and become dusty, and in areas where water shortage or chemical imbalance in the tailings prevents the use of cover vegetation. Chemical surface stabilizers, however, are susceptible to physical breakup and gradual degradation and will not meet the long-term requirements for the St. Louis materials.

Other complications also can arise in achieving satisfactory chemical stabilization in that the materials seldom are homogeneous, and variables such as particle size and moisture content

affect the bonding characteristics of the chemical stabilizers.[1]

Tests were conducted by the Bureau of Mines[1] using certain chemicals (e.g. Compound SP-400 Soil Gard, and DCA-70 elastomeric polymers) on both acidic and alkaline uranium tailings. Subsequently, the chemicals DCA-70 and calcium lignolsulfonate were applied to the surfaces of the inactive uranium tailings ponds and dikes at Tuba City, Arizona, in May 1968, because low moisture conditions and high costs prohibited vegetative or physical stabilization. After 4 yr, approximately 40% of the dike surface showed disruption while the crust in pond areas was affected to a lesser extent. The major disruptions were attributed to initial penetration of the stabilizer by physical means such as vehicles, people, or animals crossing the tailings surface.

In 1969, a portion of the Vitro tailings at Salt Lake City, Utah, was sprayed with tarlike material as a Bureau of Mines experiment to achieve surface stabilization and to reduce wind erosion. The attempt was unsuccessful because the material decomposed and the tailings were exposed within 2 to 3 yr.

Since no chemical sealant has been used successfully to stabilize uranium tailings for more than a few years, this method has not been considered in the alternatives presented in Chapter 7.

5.1.2 Complete Chemical Stabilization

This process, which has been used in other mineral industry operations, involves the addition of chemicals in sufficient quantities to a slurry to produce a chemical reaction which solidifies the slurry. Chemicals may be added in two ways: to a slurry pipeline, and in situ. The in situ method of stabilization is relatively new and extensive research is required in each individual situation to define the optimum chemical addition to produce the desired results.

One of the features claimed for this stabilization method is that all pollutant chemicals are locked in the solidified slurry and chemicals cannot be leached from the solid.

The cost of this stabilization method is expensive for the chemicals alone. A cover material, such as gravel, would be required to protect the solidified slurry from wind and water erosion. It is not known whether vegetation can be established after topsoil and other soil cover have been spread over the solidified slurry. This probably would be a function of the specific chemical makeup of the solidified slurry and would require research to identify the conditions under which vegetation could thrive.

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[1] See end of chapter for references.

5.1.3 Physical Stabilization

Physical stabilization consists of isolating the contained material from wind and water erosion by covering the radioactive materials with some type of resistant material (e.g. rock, soil, smelter slag, broken concrete, asphalt, etc.) Thin covers of concrete or asphaltic materials have been shown to break down over relatively short periods of time; and starting within a few years after application, continuing maintenance is required. A concrete covering sufficiently thick and properly reinforced would be relatively permanent and maintenance-free, but the cost would be prohibitive for large areas.

In some arid regions, where the potential for successful vegetative stabilization is slight, physical stabilization may be the preferred alternative. In such areas, combinations of pit-run sand and gravel, soil, and riprap have been placed over uranium tailings and have been successful in preventing wind and water erosion. An important component of physical stabilization is the proper treatment of the finished surface by such means as contour-grading and terracing. Such treatments can reduce greatly long-term maintenance costs.

Performance objectives have been formulated by the NRC for the siting and stabilization of uranium mill tailings. Many of these objectives are applicable to the problem of long-term storage of the radioactive materials from the Latty Avenue site. Objectives applicable to this program are given below in terms of uranium tailings management:

a. Locate the tailings isolation area remote from people so that population exposures will be reduced to the maximum extent reasonably achievable.

b. Locate the tailings isolation area so that disruption and dispersion by natural forces are eliminated or reduced to the maximum extent reasonably achievable.

c. Design the isolation area so that seepage of toxic materials into the ground water system will be eliminated or reduced to the maximum extent reasonably achievable.

d. Reduce direct gamma radiation from the impoundment area to essentially background.

e. Reduce the radon exhalation rate from the impoundment area to about twice the exhalation rate in the surrounding environs.

f. Eliminate the need for an ongoing monitoring and maintenance program following successful reclamation.

5.1.4 Vegetative Stabilization

This method involves the establishment of vegetative cover on the radioactive materials or on a growing medium placed over the materials.

Many species of plants are self-regenerating and require little or no maintenance after growth becomes established. Vegetation can survive providing that:

(a) Evapotranspiration is not excessive

(b) Landscapes are properly shaped

(c) Nontoxic soil mediums capable of holding moisture are applied

(d) Irrigation and fertilization appropriate to the area are applied

(e) Proper selection of plants conducive to self-regeneration under conditions anticipated over a long time

Growth of vegetation at sites receiving less than 10 in. of annual precipitation and with high evapotranspiration rates requires irrigation and fertilization. At St. Louis, precipitation averages about 35 in. annually.

Vegetation is abundant over most of the site except in areas where it appears that surface runoff of precipitation has prevented establishment of vegetation.

One potential problem in the use of vegetative stabilization is the possibility of pickup of radioactive elements by the plants. The effect of this mechanism has not been considered in the present assessment.

5.2 PREVENTION OF LEACHING

Leaching into underground aquifers is one of the several pathways that chemicals and radioactive materials might take into the environment. The techniques which could be employed to control leaching from radioactive materials include the following:

a. Employ chemical stabilization to prevent leaching into underground aquifers (this is the same stabilization system discussed in paragraph 5.1.2).

b. Physically compact the stored materials to reduce the percolation of water through the materials.

c. Contour the surface, then employ appropriate chemicals

(discussed in paragraph 5.1.1) to seal the surface, thus preventing water from penetrating and destabilizing the materials.

d. For a new site, line the storage area with an impermeable membrane (bentonitic clays and various plastic materials commonly are used for this purpose).

5.3 REDUCTION OF RADON EXHALATION

Little research has been directed toward reduction of radon exhalation from stored radioactive materials. While there are materials that can seal or contain the gas in small quantities, none of these are suitable for permanent coverage of large areas.

From simplified diffusion theory estimates, about 13 ft of dry soil[2,3] are needed to reduce radon flux by 95%, but only a few feet of soil are needed if a high moisture content in the cover material is maintained. Figure 5-1 illustrates curves of the reduction of radon exhalation flux for three soil types versus depth of cover based upon the theory and diffusion coefficients presented in the above references. Research is under way to explore more precisely the problems associated with reducing and eliminating the exhalation of radon from radioactive tailings material. The effects of applying various chemical stabilizers and varying thicknesses of stabilizing earth covers and combinations of materials such as compacted clay and soil are still being investigated. The results may have an important impact in planning radon exhalation control.

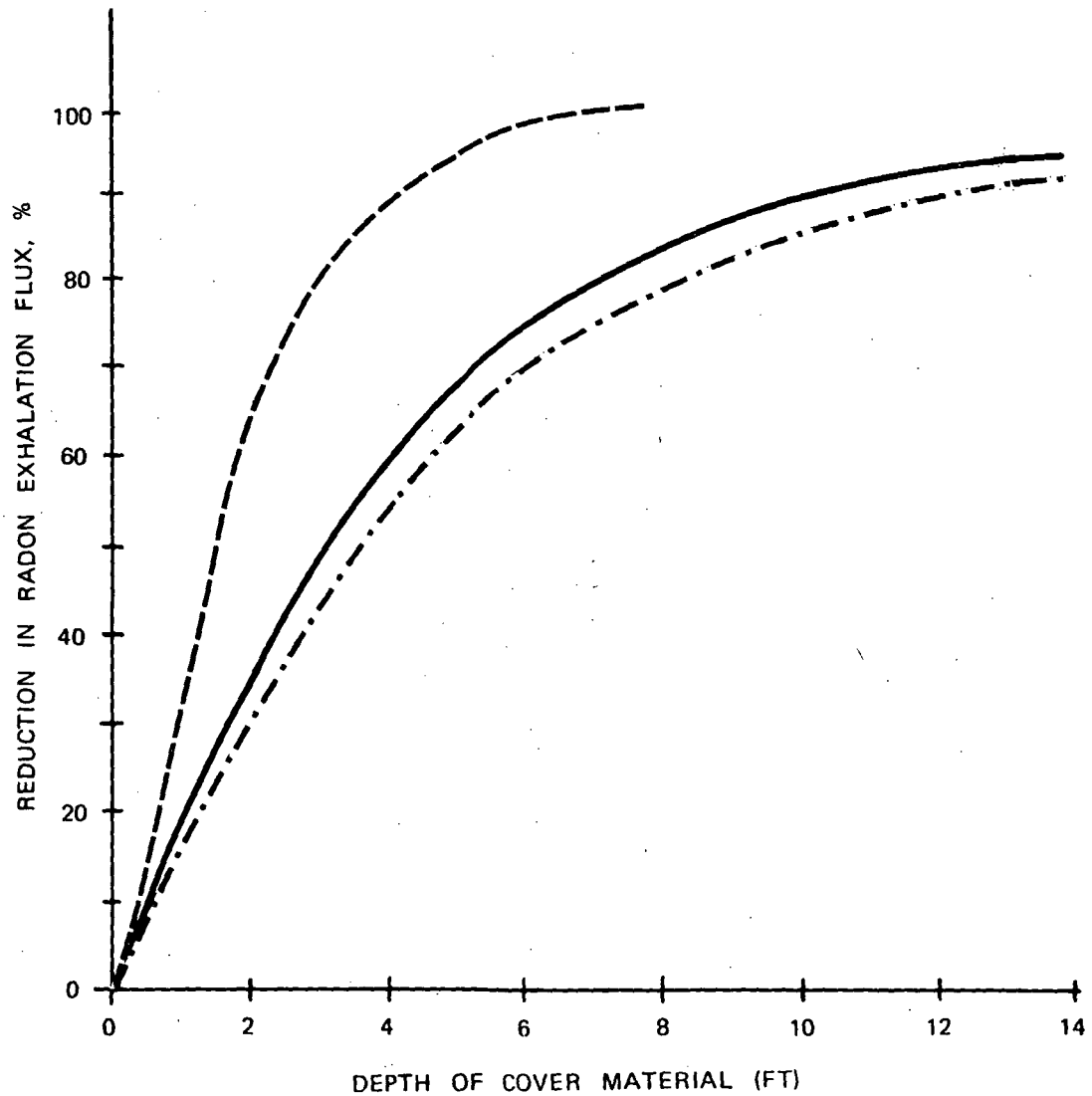
5.4 REDUCTION OF GAMMA RADIATION

A few feet of cover material are sufficient to reduce gamma radiation to acceptable levels.

The reduction of gamma exposure rates resulting from a packed earth covering is given in Figure 5-2.[4,5] Two feet of cover reduces the gamma levels by about two orders of magnitude. Therefore, an average cover of 2 ft should reduce gamma levels to less than 10 μR/hr above background.

5.5 ASSESSMENT OF APPLICABILITY

Available data indicate that none of the methods used thus far to stabilize uranium tailings sites has been a totally satisfactory solution to uranium tailings site radiation problems. Some of the methods examined have exhibited short-term advantages, but no economical long-term solutions have become apparent. Consequently, new methods of stabilization may have to be developed and additional engineering research may be required. However, one of the present remedial action alternatives includes physical stabilization of the radioactive materials with 1 ft of compacted clay and 5 ft of soil. (See Figure 7-2, Chapter 7.) This action will greatly reduce gamma radiation, radon exhalation, and wind and water erosion.



LEGEND

RESULT FOR SOIL (USED IN THIS EVALUATION)

RESULT FOR DRY SAND

RESULT FOR CLAY

FIGURE 5-1. REDUCTION OF RADON EXHALATION FLUX WITH DEPTH OF COVER

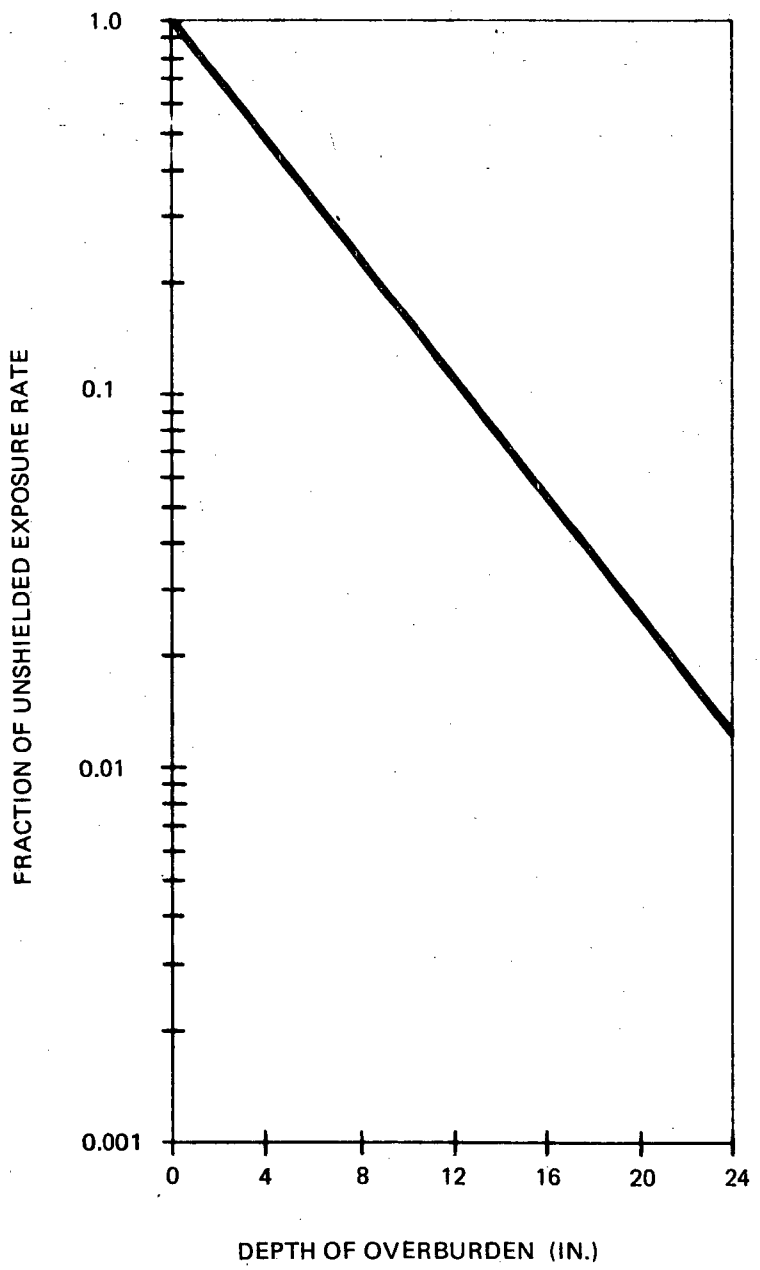


FIGURE 5-2. REDUCTION OF GAMMA EXPOSURE RATE RESULTING FROM PACKED EARTH SHIELDING

130-12

CHAPTER 5 REFERENCES

1. "Methods and Costs for Stabilizing Fine Sized Mineral Wastes"; Bureau of Mines Report of Investigations; RI7896; 1974.

2. A. B. Tanner; "Radon Migration in the Ground: A Review"; The Natural Radiation Environment; J. A. S. Adams and W. M. Lowder, eds; University of Chicago Press; pp. 161-190; 1964.

3. H. W. Kraner, G. L. Schroeder, and R. D. Evans; "Measurements of the Effects of Atmospheric Variables on Radon-222 Flux and Soil-Gas Concentrations"; The Natural Radiation Environment; J. A. S. Adams and W. M. Lowder, eds; University of Chicago Press; 1964.

4. K. J. Schiager; "Analysis of Radiation Exposures on or Near Uranium Mill Tailings Piles"; Radiation Data and Reports; Vol 15; Jul 1974.

5. "Evaluation of Various Methods, Techniques and Materials for Stabilizing Uranium Mill Tailings"; FB&DU Report (in preparation).

CHAPTER 6

LONG-TERM STORAGE (DISPOSAL) SITE SELECTION

In all but the first alternative discussed in Chapter 7, the contaminated material would be removed to a storage location. Long-term (greater than 50 yr) storage sites are referred to in this report as disposal sites.

Locations were sought for disposal of the contaminated materials now on the Latty Avenue site which would meet acceptable criteria for the proper handling of such materials and which would prohibit their entry into the environment.

6.1 AIRPORT FILL SITE (ALTERNATIVE II)

This location is 2.5 road miles south and slightly west of the Latty Avenue site, and consists of approximately 21.7 acres. Two or three acres of this site could be developed into a permanent disposal site for the Latty Avenue contaminated materials. The site is located between Brown Road and the Norfolk & Western Railroad at the extreme north end of the Lambert-St. Louis International Airport. The land is owned by the St. Louis Airport Authority. The site was used as a storage area for wastes generated by the Mallinckrodt Chemical Corporation during their uranium-processing operations from 1946 to 1953. During 1966 and 1967 the piles were sold and removed from the site. As stated in the acquisition permit of November 10, 1969, the St. Louis-Lambert Airport Authority agreed to decontaminate this property. In an agreement with the U.S. Government, it was required that the barium sulfate residue be removed to a federal waste repository at Weldon Spring, Missouri, and that all structures on site except the fence be razed. Building rubble that included a storage shed, truck wash pad, and a concrete storage pit was to be buried. Also, a minimum of 1 ft of clean fill was to be placed over the entire site. This work was performed during the period January through December 1969. The site has been used since 1970, and apparently is still being used, as a dump site for rubble developed during airport remodeling or enlarging operations. The site is enclosed with chainlink fencing, and has gates with access from Brown Road. In 1977, the ORNL found radioactive materials remaining at this site.[1]

Immediately north of the rail line and at the south edge of the site there is a drainage ditch that parallels the site and enters Coldwater Creek at the extreme west edge of the site. (See Figure 2-1, Chapter 2.)

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[1] See end of chapter for references.

There are places on the site where surface water gathers, but these vary because of the continuing placement of debris and fill material onto the area. The soil, geologic, ground water and meteorological conditions at this location would be identical to those of the Latty Avenue site, except for the debris present on the surface. (See Figures 2.4, 2.5 and 2.6, Chapter 2.)

The description of the 2- to 3-acre storage site that would be constructed at this location in order to receive the Latty Avenue material is included in Chapter 7, paragraph 7.2.

6.2 WELDON SPRING CHEMICAL PLANT, PIT NO. 4 DISPOSAL SITE (ALTERNATIVE III)

The Weldon Spring Chemical Plant is located on the north side of Missouri State Highway Route 94, approximately 13 mi southwest of St. Charles and approximately 23 road miles west of the Latty Avenue site in St. Charles County, Missouri.

The facility was an AEC integrated plant complex located on 200 acres of land. AEC operations ceased in 1966 and the U.S. Army acquired most of the site in 1967 for production of defoliants. The AEC (now DOE) retained ownership of approximately 52 acres, principally comprising four pits containing radioactive residues from uranium and thorium processing. It is proposed that the material from the Latty Avenue site be deposited into raffinate pit 4, shown in Figure 6-1.

The pit was constructed in 1964, and has a surface area of about 15 acres.[2] It contains raffinate solids from the processing of thorium recycle materials as well as some uranium residues. The residue fill in the pit is irregular and utilizes slightly more than 10% of the total pit volume. Also, trash and drums containing residue were dumped into this pit during shutdown operations of the Weldon Spring plant. The storage volume of the pit is estimated to be at 444,000 yd^3. Including processing residues, trash and debris, there are approximately 56,000 yd^3 of material now in the pit; this leaves ample room for the estimated 18,300 yd^3 to be removed from the Latty Avenue site.

There is some rainwater in the pit. The water depth varies with the amount of the annual rainfall, from 0 to 8 ft. The top of the dike is at an elevation of 663 ft above sea level. The east dike of pit 4 is shared in common with pit 3. The bottom elevation of pit 4 varies markedly with the existing terrain, with its lowest level some 30 ft below the top dike elevation. The dike has slopes which range from 1.5:1 to 2.5:1.

The subsurface strata in the plant area were investigated[2] prior to plant construction and again in more localized detail before the construction of pit 4. Interpretation of data by geologists familiar with the geology and hydrology of the area led to the conclusion that there is a minimum of 10 ft of very

impermeable clay underlying the pits. This clay, in turn, overlies about 3 ft of somewhat more permeable clay lying above 180 ft of relatively permeable limestone. The average permeability of the upper clay has been measured at about 10^{-8} cm/sec, or about 0.01 ft/yr. It was noted that the pit solids will further "plug" the clay and reduce this permeability. (See Appendix C for a copy of a 1967 geologic and hydrologic report made for the Army of the storage pit area of the site.)

The suggested storage of the contaminated materials from the Latty Avenue site in pit 4 at Weldon Spring is justified by the following[2]:

(a) The risk of contaminating underground aquifers used for drinking water purposes is minimal.

(b) With proper maintenance of the area and embankments, there is only insignificant risk of pollutant discharge through embankment breaches.

(c) Seepage or leakage either under or through the embankments is essentially nonextistent, and is in no way connected with the existence of the low levels of radioactivity found in on-site and off-site drainage.

(d) The risk of physical harm to trespassers is concluded to be minimal because of the limited access to the residue pits, the formidable fence, and the signs warning of the presence of radioactive material.

(e) Radioactivity at the residue pits is calculated to be at levels too low to cause significant exposures to caretakers in the performance of their usual duties.

(f) The haul distance is relatively short when compared with other off-site storage site locations.

There are some locations where embankment slippage has occurred on the west bank of the pit. Minor stabilization repairs, with re-seeding, should correct this situation. The dikes appear to be structurally sound otherwise.

6.3 BARNWELL, SOUTH CAROLINA PRIVATELY OWNED DISPOSAL SITE (ALTERNATIVE V)

This site is located in Barnwell County in the southwest portion of South Carolina, approximately 5 mi northeast of the community of Barnwell. It is a commercial radioactive waste material storage site licensed by the NRC, license No. 46-13536-01. Transportation and waste disposal costs are discussed in Chapter 7.

6.4 REMOVAL OF THE RADIOACTIVE MATERIALS TO A U.S. GOVERNMENT STORAGE FACILITY, SUCH AS AT THE NTS, NEVADA

At such a location the material, shipped in bulk, would be stored under the auspices and conditions as set forth by the DOE. The NTS site is located 65 mi northwest of Las Vegas, Nevada.

6.5 SITES CONSIDERED BUT NOT INCLUDED AS ALTERNATIVES

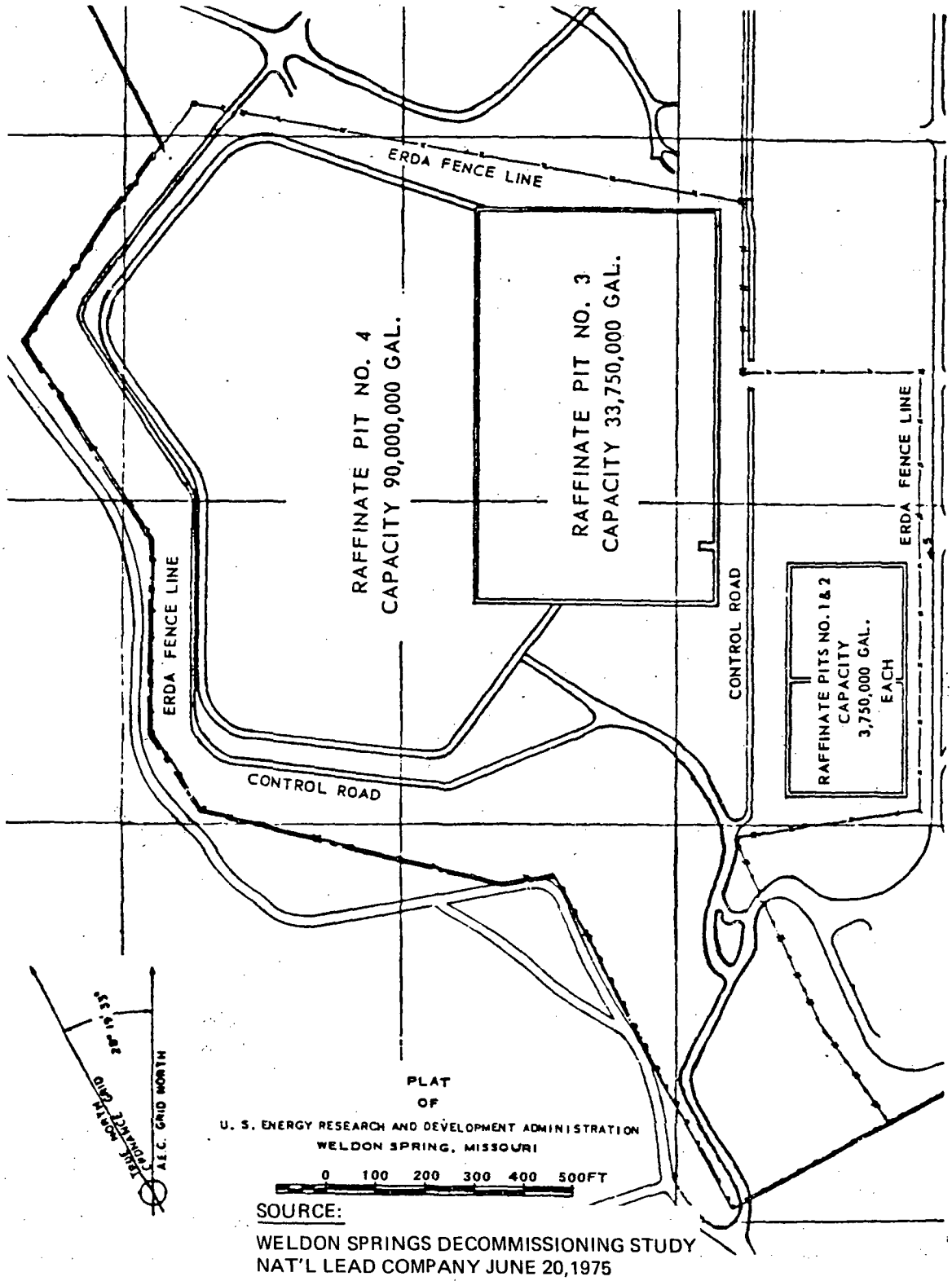
6.5.1 Westlake Landfill, St. Louis

This site, shown on Figure 2-1, Chapter 2, already contains some radioactive material as described in Chapter 2. Because of hydrological conditions and objections expressed by virtually every governmental agency involved, this site was excluded.

6.5.2 The "Quarry" Site at Weldon Spring

The Quarry is an abandoned rock quarry located on a 7-acre site lying between Missouri State Route 94 and Femme Osage Creek, 3 mi southwest of the Weldon Spring site. It was originally used by the Department of the Army for disposition of trinitrotoluol-contaminated rubble during the operation of the Weldon Spring Ordinance Works.

The AEC acquired title and possession of the Quarry in 1958, using it for disposal of drummed thorium residues and uranium-contaminated building rubble. It was not considered as a long-term disposal site for the same reasons that the Westlake landfill was omitted.



SOURCE:
WELDON SPRINGS DECOMMISSIONING STUDY
NAT'L LEAD COMPANY JUNE 20,1975

FIGURE 6-1. | WELDON SPRINGS STORAGE SITE (ALTERNATIVE III)

UC-225

CHAPTER 6 REFERENCES

1. "Radiological Survey of the Former AEC-St. Louis Airport Storage Site, St. Louis, Missouri"; Health and Safety Research Division; ORNL; Oak Ridge, Tennessee; Apr 1977.

2. "ERDA-Weldon Spring, Weldon Spring Decommissioning Study"; Report No NLCO-1121; Feed Materials Production Center; National Lead Company of Ohio; June 20, 1975.

CHAPTER 7

REMEDIAL ACTION ALTERNATIVES

After the extent of radioactive contamination was determined, remedial action alternatives were developed which, if implemented, would leave the site in a "clean" and usable condition. Immediately adjacent areas which are contaminated slightly are also included in the decontamination work. Also, building decontamination versus replacement was considered, with cost and practicality governing the decision as to whether it would be more economical to decontaminate the structures or to remove them from the site and replace them in kind.

In all of the alternatives included, the decontamination efforts are basically the same; the main differences are the method of removal and the place of depositing the removed contaminated earth, building materials and rubble. With but one exception--Alternative I, where a short-term storage site would be developed on the 11-acre Latty Avenue site--all of the proposed alternatives would leave the complete site in a condition which would allow for its unrestricted use for other purposes.

7.1 SHORT-TERM STORAGE SITE DEVELOPED ON SITE (ALTERNATIVE I)

All of the contaminated structures, soil and rubble (see Table 7-1) would be collected and placed in a 1.5-acre storage area which would be constructed in the extreme south end of the site. The storage area would be fenced, and access provided by means of a right-of-way for future removal purposes. No perpetual care fund for continuing maintenance and monitoring would be provided. Geologic and hydrologic studies indicate that this area would be a suitable short-term storage site.

All of the buildings would be decontaminated in order to be of further use. With Buildings 1 and 2, contamination has penetrated the insulation, rusted steel frame, roofing, siding and foundations. To make these buildings clean and usable, the gutters, roofing, siding and insulation would be removed (in that order). These removed parts then would be sand-blasted or steam-cleaned, then stacked while the steel frame and foundation were cleaned. The insulation would be stored for depositing with the site's contaminated soil. The steel frames and purlins would be sand-blasted or wire-brushed. The porous concrete foundation walls would be removed between the column foundations; and the column foundations chipped away enough to remove contaminated concrete, but leaving enough to preserve structural safety. (See Appendix A for a complete analysis of decontamination efforts.) The concrete then would be replaced, the steel frame painted, new insulation applied; then the siding, roofing, and gutters would be replaced in the same locations they were in before removal. The dried-on mastic which would have been removed from between

every 3-ft joint of the siding would be replaced with new mastic, and larger sheet-metal screws employed. In addition, the residue obtained from decontamination would be collected carefully and removed for storage. All of this work would be time consuming and expensive. The estimated total cost of the decontamination work for these two buildings is $156,000, which does not include removal and/or storage costs. Figure 2-4, Chapter 2, a photograph of the interior of the northeast corner of Building 1, shows the typical existing condition of these two structures, the rusted frame, the broken insulation, and how some of the main structural members are bent. The total replacement cost (in kind) of these two structures is estimated to be $135,000, which includes demolition costs but excludes the removal of contaminated materials for storage.

Building 3 would be decontaminated and left on the site since the costs for such work (estimated to be $15,300) would be less expensive than demolishing the structure and replacing it in kind (estimated to cost $17,600).

Building 4 also would be decontaminated and left on the site. Estimated costs for this work is $8,900, while demolition and replacement cost would be about $52,900.

Appendix A includes a procedural and cost analysis for decontaminating Buildings 3 and 4.

For this alternative, the description and sequence of cleanup and on-site storage operations would be as outlined in the following steps:

a. Table 7-1 indicates that there is approximately 18,300 yd^3 of contaminated material on the site that would need to be stored or removed to an alternate area, then buried and stabilized. This would require about 1 acre of land to a depth of 11.5 ft or 1.5 acres to a depth of 7.6 ft. The extent of the contamination cleanup work is shown in Figure 7-1. An area of 1.5 to 2.0 acres at the extreme south end of the site would be selected as a short-term storage site. Configuration would not be important. Care would be taken to include no sewage easement property, and to select no land that would block access to the 7-acre part of the site from railroad spur access should it be needed in the future. Right-of-way access would be selected to allow for maintenance and future removal operations.

b. Buildings 3 and 4 would be covered and sealed with plastic film so that cleanup operations would not further contaminate them.

c. Weeds on the areas to be decontaminated, as well as on the entire site, would be mulched. Existing fences and their posts and foundations would be removed to a storage location adjacent to the storage pit area.

d. Earthmoving equipment would remove all of the contaminated soil from the storage pit area and from another area of sufficient size to facilitate the stockpiling of clean excavation from the pit.

e. During earthmoving operations the area would be sprinkled in order to control dust.

f. The storage pit would be constructed. Depth would be a minimum of 2 ft into "clean" ground. The excavated material would be stockpiled in the clean area and later used as stabilization cover over the pit. A layer of plastic liner would be placed in the pit to keep ground water away from the contaminated soil and rubble and to prevent leaching into the ground. All of the contaminated earth, building structures, weeds, fencing, and rubble would be placed in the pit.

g. As the grading and cleanup operations were proceeding, Buildings 1 and 2 would be removed, cut into pieces and placed in the storage pit along with the contaminated earth.

h. The plastic covering on Buildings 3 and 4 would be removed. The buildings would be decontaminated. (See Appendix A for a description of this effort.) The contaminated residue from this operation, along with the plastic covering, would be placed on the pile in the pit.

i. During all of this decontamination effort, protective clothing and masks would be provided workers, and all vehicles and equipment would be washed down before leaving the site.

j. After monitoring indicates the site to be clean, the pile would be covered with a layer of the same type of plastic liner used in lining the bottom. The pile then would be stabilized with 2 ft of the material excavated from the pit and stockpiled.

k. The stabilization cover would be seeded with grasses native to the area, and the 1.5-acre site would be fenced with a 6-ft-high chainlink fence with 3 strands of barbed wire on top. An access gate would be provided. Radiation warning signs would be placed on the gate, on the fence, and in other appropriate places.

l. Buildings equaling the square and cube footage of those removed (Nos. 1 and 2) would be constructed. They also would be of the same type of construction, have dirt floors, and have no partitions, no utilities, nor electrical services.

m. Clean fill equal to the volume of that removed in decontamination work would be replaced on the site. Asphalt and concrete paving and curbs would be replaced in kind and the same amount of sound fencing that existed before cleanup operations would be installed. The railroad spur also would be replaced.

7.1.1 Security and Maintenance

The short-term storage area of the site would be a "control area", and be fenced and posted. Inspection and physical maintenance of the pile would be required to include such things as assuring the physical integrity of the stabilization cover, controlling the weed growth, and keeping the fencing and signs in good repair. No irrigation is proposed.

7.1.2 Resulting Impacts

As a result of the remedial action of this alternative, the contaminated material would remain temporarily on the site, and the necessary storage area would approximate 1.5 acres or 13.6% of the total site. The balance, or 86.4% of the site, would be returned to "normal" and be usable for any purpose. The location of the storage area is suggested in general, but which of the two landowner parcels of the site would be selected, or whether to use a combinatin of both would be up to others to decide. Approximately 25 working days are estimated as necessary for completion of this work, once approvals and designs are approved, plus another 20 days to complete erection of new Buildings 1 and 2.

The advantages of this alternative are that it could be implemented quickly, and that it is the least costly (considering first costs only). Disadvantages are that the contaminated material would not be relocated to an isolated area, thus potential exposure would not be reduced. Also, even though a plastic film would be used, both as a bottom and top liner, potential for contamination of ground water by leaching is possible. Also, the radon exhalation from the material could not be reduced to background. At some future time it might become necessary to remove the contaminated material to a long-term storage area, so those costs at that time are not included in the cost estimate for this alternative. The complete site would not become available for unrestricted use, and land lease or acquisition costs are not included.

7.1.3 Costs

The estimated total cost, $457,000, for this alternative is the lowest cost of the alternatives considered for cost comparisons. Table 7-2 is a cost comparison summary of all the alternatives. The major components of the estimate are as follows:

Buildings 1 and 2 demolition and replacement	$135,000
Buildings 3 and 4 decontamination	26,370
Decontamination of ground surfaces on site and on adjoining areas	26,370
Site restoration (fencing, back-fill, etc.)	58,000
Storage site preparation, stabilization and fencing	81,430
Subtotal	325,000
Engineering, architectural, and monitoring fees (25%)	82,000
Contingency (15% of subtotal)	50,000
Total Cost	$457,000

Costs of land acquisition or leasing for the storage site and its access right-of-way are not included. No costs are included for any maintenance which may be required to provide for the physical integrity of the short-term storage site.

7.2 DECONTAMINATION AND THE MATERIAL REMOVED TO THE AIRPORT FILL SITE (ALTERNATIVE II)

This alternative calls for all of the contaminated soil, building components, and rubble from the Latty Avenue site to be removed to the Airport fill site. In this location the material would be deposited in a storage pit especially designed to serve as a long-term disposal location. As in Alternative I, Buildings 1 and 2 would be demolished and replaced in kind; and Buildings 3 and 4 would be decontaminated and remain on the site.

The sequence of decontamination site event operations would be as follows:

a. Buildings 3 and 4 would be covered and sealed as in Alternative I.

b. The site's weed growth would be mulched and fences, rubble and rail spurs would be gathered in one location.

c. Buildings 1 and 2 would be demolished and cut into sizes that would allow easy transport to the disposal site.

d. The area would be sprinkled with water to contain dust.

e. A truck washdown facility would be constructed, so that no material could be transported off site on the vehicles.

f. The contaminated material would be gathered by earth-moving equipment and loaded onto trucks for transportation to the storage site 2.5 mi distant. Rubber tarpaulins would keep the material from being wind blown in transit.

g. The disposal site would be constructed at the airport site. This pit would be constructed as shown in Figure 7-2. Basically it would be a pit covering approximately 1.5 acres, lined with a 1-ft layer of impervious clay and covered with the same type of clay plus a minimum of 5 ft of earth cover. The steep side slopes would be protected against erosion by the application of a 1-ft-thick layer of riprap, and the bottom sides of the pit would be lined with a 1-ft-thick layer of gravel which would serve as a french drain in order to direct surface water away from the stored material. No fencing would be required since the site already is well protected with chainlink fencing.

h. After cleanup of the Latty Avenue site, Buildings 3 and 4 would be decontaminated and the residue hauled to the disposal site.

i. During all of this cleanup activity, workers would utilize protective clothing and masks.

j. The replacement buildings would be constructed, as in Alternative I.

k. The Latty Avenue site would be brought up to grade with clean fill and paving, curbs, fencing and rail spur would be replaced, all as in Alternative I.

7.2.1 Security and Maintenance

The existing fence at the airport site would need to be posted with radiation warning signs, and the site would be designated as a "control" area. Radiation and water monitoring wells would need to be installed around the pile. Inspection and physical maintenance would be required to include such things as assuring the physical integrity of the stabilization cover, controlling the weed growth, and maintaining the fencing and signs. A monitoring and sampling program would be undertaken to take water samples and obtain radiometric readings below and above ground. No irrigation is proposed.

7.2.2 Resulting Impacts

As a result of the remedial actions of this alternative, the Latty Avenue site would be fully decontaminated, all structures would be decontaminated or replaced in kind, and the site would be released for full and unrestricted use.

Approximately 35 working days would be required to complete the work, plus another 20 days to complete erection of new Buildings 1 and 2.

The obvious advantages of this alternative are that it could be accomplished quickly, that the Latty Avenue site would be free of contamination, that the contaminated material would be placed in an area already contaminated, and that radon exhalation from the newly deposited contaminated materials would be reduced to twice background. The cost of this alternative is a relative advantage, considering the site would be a long-term storage location. Disadvantages are that the material would not be placed in an isolated area, and that an ongoing monitoring and maintenance program would need to be undertaken. Also, if the airport site in its entirety should have to be decontaminated someday to remove existing contaminants, the costs of this alternative would be wasted because another location for the material would have to be found and funds for relocation again would be spent.

7.2.3 Costs

As shown in Table 7-2, the estimated cost of this alternative is $730,000. The major components of the estimate are as follows:

Buildings 1 and 2 demolition and replacement	$135,000
Buildings 3 and 4 decontamination	24,210
Decontamination of ground surfaces on site and on adjoining areas	62,150
Site restoration	63,800
Storage pit construction	94,500
Loading, hauling and placing of contaminated material into storage pit	70,350
Subtotal	450,000
Engineering, architectural and monitoring fees (25%)	112,500
Contingency (15% of subtotal)	67,500

Perpetual care fund for long-term monitoring and maintenance	$100,000
Total cost	$730,000

No costs are included for land at the airport fill site. The perpetual care fund at 7% annual interest would provide $7,000/yr, which is the estimated cost to perform the necessary maintenance and monitoring work.

7.3 DECONTAMINATION AND THE MATERIAL REMOVED TO WELDON SPRING (DOE) SITE (ALTERNATIVES III - A & B)

7.3.1 Bulk Shipment and Storage (Alternative III-A)

This alternative calls for the complete removal of the contaminated soil, building components, and rubble from the site to the DOE tailing pond area adjacent to the Weldon Spring Chemical Plant. The site would be decontaminated as in the other alternatives. Buildings 1 and 2 would be demolished and hauled to the storage site and subsequently replaced. Buildings 3 and 4 would be decontaminated and remain on site.

The sequence of operations for decontamination, demolition, removal, and reconstruction would be the same as that described in Alternative II, except that the material would be hauled in bulk via trucks and placed into pit No. 4 at the DOE Weldon Spring facility. The material could be placed in whatever configuration and location desired by DOE. Assuming that it were placed to an average depth of ±7.6 ft, it would cover approximately 1.5 acres. No special pit or excavation would be required. If it were necessary to cover or stabilize the contaminated Latty Avenue material at this storage location, it would cost approximately $10,000/ft for the 1.5-acre coverage.

7.3.1.1 Security and Maintenance

No additional security precautions would be necessary. The existing system of security checks, including radiation warning signs, is sufficient. The presence of the Latty Avenue materials in pit 4 would not require any additional maintenance. A monitoring program, with well and surface sample points, already exists at the Weldon Spring site.

7.3.1.2 Resulting Impacts

The proposed remedial action would free the Latty Avenue site from any contamination. The site would be backfilled with clean soil to existing grades. Buildings 1 and 2 would be replaced in kind. Buildings 3 and 4 would be decontaminated. All land and buildings then would be available for unrestricted use.

At Weldon Spring, only a relatively small amount of contaminated material would be added to pit 4, which already contains contaminated material.

It is estimated that the site decontamination and removal of materials would require 35 working days, plus 20 days to finish erection of new Buildings 1 and 2.

The main advantages of this alternative are the complete cleanup of the Latty Avenue site, relocation of all contaminated material along with other contaminated material to an isolated location. Also, no additional monitoring and maintenance programs would need to be established, and the cost would be only $56,000 more than the cost for interim storage at the Latty Avenue site. Disadvantages are that the material would increase radon exhalation from pit 4. Also, should the DOE desire to clean up all of the Weldon Spring site, then the Latty Avenue contaminated material would have to be relocated again.

7.3.1.3 Costs

As shown in Table 7-2, the estimated cost of this alternative is $513,000. The major components of the estimate are as follows:

Buildings 1 and 2 demolition and replacement	$135,000
Decontamination of ground surfaces on site and on adjoining areas	63,150
Loading, hauling and unloading	80,850
Site restoration	63,800
Subtotal	366,000
Engineering, architectural and monitoring fees (25%)	92,000
Contingency (15% of subtotal)	55,000
Perpetual care fund for maintenance	-0-
Total cost	$513,000

Since continued maintenance and monitoring are not required at the Weldon Spring site, none are included in the estimate.

7.3.2 Packaged Shipment and Storage (Alternative III-B)

Under this alternative, the contaminated soil would be packaged, shipped and stored in 55-gal drum containers. The dismantled and cut-up components of the structures to be removed

(Buildings 1 and 2) and the large rubble, such as concrete floor sections, would not be packaged but would be hose-washed before leaving the site. This would enable the materials to be carried via trucks without spilling contaminated particles along the route. The drums also would be shipped to Weldon Spring by truck. A loading hopper would be used at the Latty Avenue site to facilitate the filling of the contaminated soil and small debris.

In all other aspects, this alternative is the same as that of Alternative III-A. The estimated time to complete the work is 70 days, plus 20 additional days to complete the buildings.

The main advantage of this alternative would be the complete cleanup of the Latty Avenue site, and if necessary the easy relocation of the contaminated soil and small rubble which can be carried in containers to another storage site. Gamma radiation and radon exhalation would be controlled as long as the containers remain sound. No monitoring would be required. Disadvantages are the costs of the containers, and the lengthy time and expense of loading (filling) the containers. Also, the contaminated building components and large rubble would need to be packaged should it be required to relocate the materials again. The storage life of the containers would be in the vicinity of 20 to 25 yr, after which time their contents would need to be placed in permanent storage or repackaged again.

7.3.2.1 Costs

As shown in Table 7-2, the estimated cost of this alternative is $1,867,000. The major components of the estimate are as follows:

Buildings 1 and 2 demolition and replacement	$ 135,000
Buildings 3 and 4 decontamination	24,200
Decontamination of ground surfaces, on site and on adjoining areas	62,150
Purchase, loading, hauling and depositing of storage drums	1,213,000
Washing, loading, hauling and depositing of large rubble and building components	20,000
Site restoration	63,800
Subtotal	1,518,150
Engineering, architectural and monitoring fees (8%)	121,450

Contingency (15% of subtotal)	$ 227,400
Perpetual care fund	-0-
Total cost	$1,867,000

7.4 DECONTAMINATION AND THE MATERIAL REMOVED TO A PRIVATELY OWNED DISPOSAL SITE: BARNWELL, SOUTH CAROLINA (ALTERNATIVE IV)

This alternative is similar to Alternative III-A except that the contaminated material would be shipped to a private storage facility in Barnwell, South Carolina owned by ChemNuclear Systems, Inc. Once the material is loaded, it becomes the responsibility of ChemNuclear to transport and store the material safely. Railroad service is available from the Latty Avenue site, but at Barnwell, this service terminates about 5 mi from the storage area. However, even with the costs of double handling of the material at the storage end, it probably is cheaper to haul the material the 620 mi to Barnwell by rail. A total of 60 working days is estimated necessary to complete this work, plus 20 days to complete the buildings.

The advantages of this alternative would be that the Latty Avenue site would be rendered free of contamination, and that the NRC criteria for performance objectives for the siting and stabilization of uranium mill tailings (contaminated soil and rubble, in this instance) would be met (see paragraph 5.1.3, Chapter 5). Disadvantages would be the costs involved that are associated with the complete removal of the problem from the government. Also, to accomplish this cleanup effort would take twice the time required for other alternatives because of the loading and hauling times involved.

7.4.1 Costs

As shown in Table 7-2, the estimated cost of this alternative is $4,131,000. The major components of the estimate are:

Buildings 1 and 2 demolition and replacement	$ 135,000
Buildings 3 and 4 decontamination	24,210
Decontamination of ground surface, on site and adjoining areas	62,150
Loading and hauling of contaminated material	2,178,000
Storage costs (fees for storage and care)	1,038,000

Site restoration	$ 63,800
Subtotal	3,501,150
Engineering, architectural and monitoring fees (3%)	105,050
Contingency (15% of subtotal)	524,800
Total Cost	$4,131,000

7.5 DECONTAMINATION AND THE MATERIAL REMOVED TO A U.S. GOVERNMENT STORAGE FACILITY (ALTERNATIVE V)

This alternative is similar to Alternative III-A except that the contaminated material would be shipped approximately 1,720 mi to the Nevada Test Site (NTS) for storage at a facility operated by the DOE. Shipping would be by rail. Approximately 60 working days should be required for completion of this work, plus an additional 20 days to complete construction of the replacement buildings.

The contaminated materials would be loaded into gondola-type railroad cars by means of a dragline-type of loader. The cars have a capacity of 100 tons each; thus it would require 300 cars. There are no railroad facilities at the disposal site; therefore, the material would be hauled by truck from the closest rail facility to the storage site (an estimated 80 mi). No storage fees or continual maintenance costs at the Nevada storage site are included.

Advantages of this alternative would be the complete cleanup of the Latty Avenue site and the meeting of the NRC criteria for performance objectives. Disadvantages are the number of cars involved, plus the unknown costs for ongoing maintenance and monitoring programs at the NTS. The time for accomplishing the cleanup work is also longer than that for Alternatives I, II and III-B.

7.5.1 Costs

As indicated in Table 7-2, the total estimated cost of this alternative is $3,552,000. The major components of the estimate are:

Buildings 1 and 2 demolition and replacement	$ 135,000
Buildings 3 and 4 decontamination	24,200
Decontamination of ground surfaces on site and adjoining areas	62,150

Loading, hauling and unloading contaminated materials	2,700,000
Site restoration	63,800
Subtotal	2,985,150
Engineering, architectural, and monitoring fees (5%)	119,450
Contingency (15% of subtotal)	447,400
Total Cost	$3,552,000

This cost assumes that there are no costs for storage, monitoring and maintenance of the contaminated materials at this U.S. Government site.

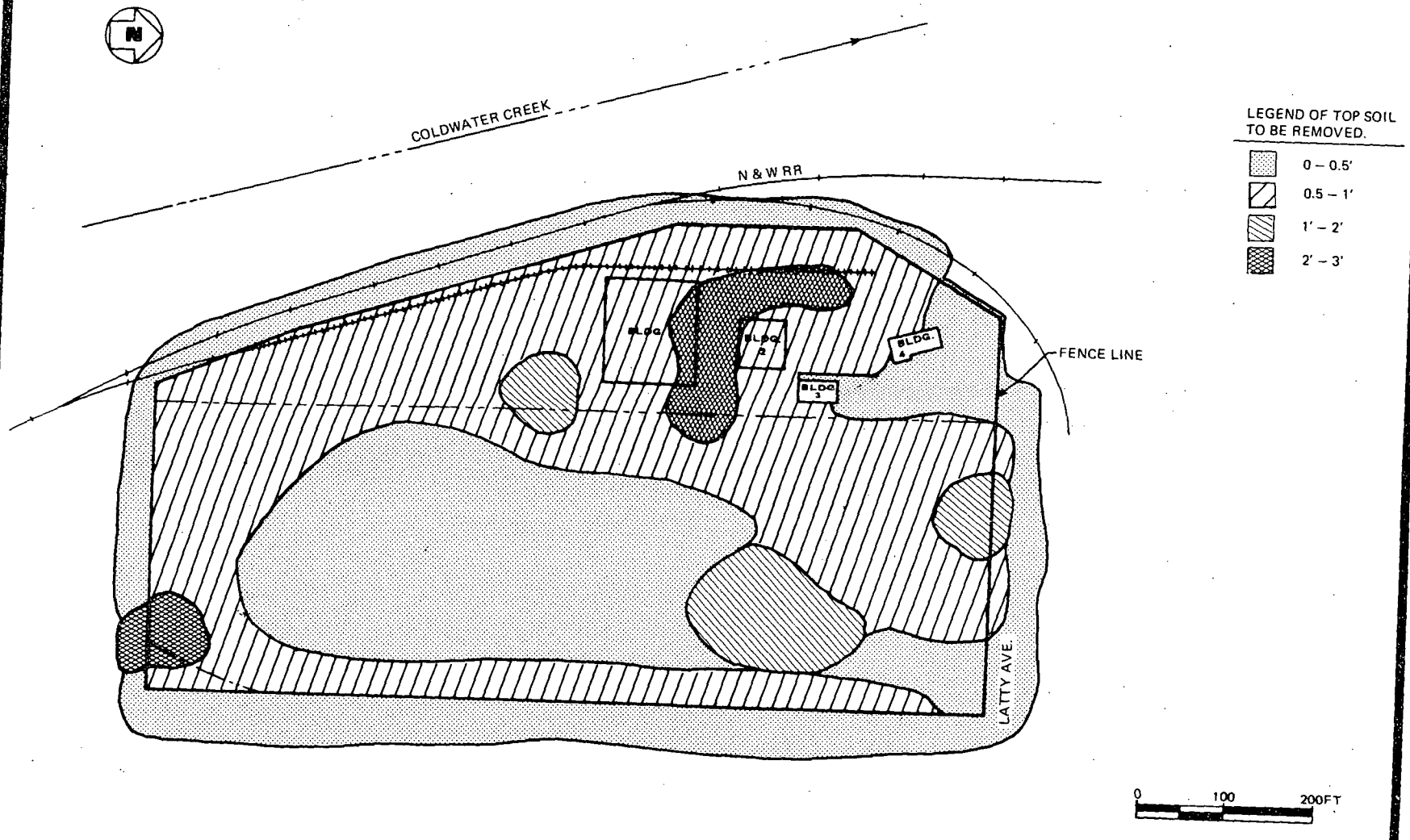


FIGURE 7-1. DECONTAMINATION PLAN

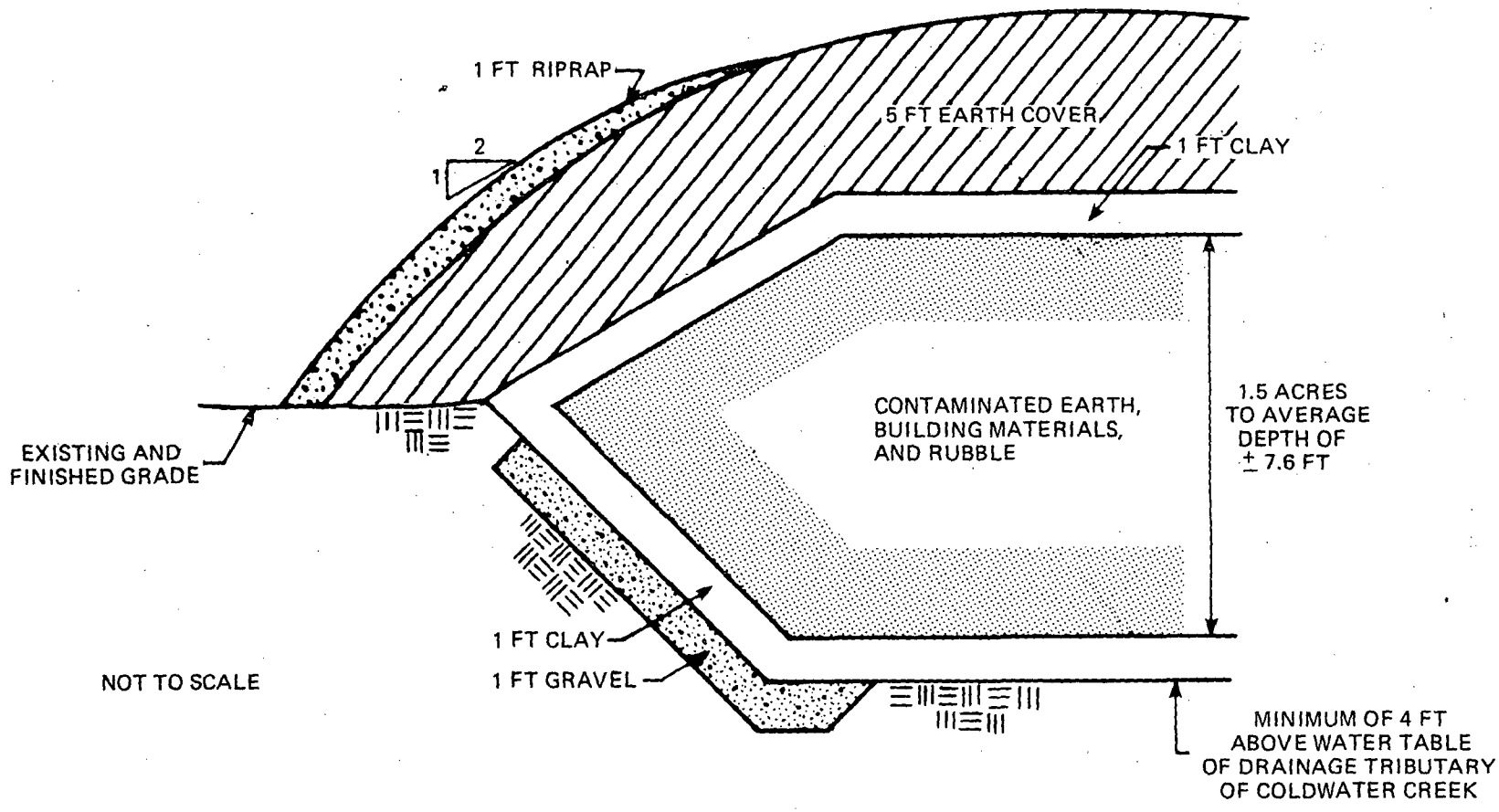


FIGURE 7-2. SCHEMATIC OF AIRPORT DISPOSAL AREA (ALTERNATIVE II)

UC-225

TABLE 7-1

CONTAMINATED MATERIALS ON THE SITE

Material	Volume (yd^3)	Weight (tons)
Soil	17,900	29,000
Structures (Buildings 1 and 2)	150*	94
Rubble: concrete slabs and foundations, fencing, weeds, rails and junk	250*	206
TOTALS	18,300	29,300

*These volumes assume the materials would be torn down, compacted, and/or stacked by heavy construction equipment such as a bulldozer.

TABLE 7-2

COST SUMMARY OF ALTERNATIVES

Alternative No.	Description	Cost
I	Decontamination of site and Bldgs 3 & 4 and replacement of Bldgs 1 & 2; interim storage of contaminated materials on-site	$ 457,000
II	Same as I, except all contaminated materials removed to St. Louis Airport fill site for long-term storage	$ 730,000
III-A	Same as I, except contaminated materials are shipped in bulk for storage at DOE Weldon Spring facility	$ 513,000
III-B	Same as III-A, except contaminated materials are stored in drums at Weldon Spring	$1,867,000
IV	Same as I, except materials are shipped to and stored at private facility in Barnwell, S.C.	$4,131,000
V	Same as I, except materials are shipped by rail to NTS for storage	$3,552,000

APPENDIX D.1

DECONTAMINATION AND COST ANALYSES FOR FOUR BUILDING STRUCTURES AT 9200 LATTY AVENUE, HAZELWOOD, MISSOURI

APPENDIX D.1

DECONTAMINATION AND COST ANALYSES FOR FOUR BUILDING STRUCTURES AT 9200 LATTY AVENUE, HAZELWOOD, MISSOURI

A. CONTAMINATION MEASUREMENTS

A spot-check investigation was conducted to verify perviously determined radiometric data as presented by other investigators. It is evident that the previous investigator's objective was to obtain absolute measurements and not to determine the decontamination feasibility of the structures. The instrumentation surveys made in this decontamination study are intended only to determine the average rather than absolute values as well as establish average decontamination factors using a variety of reagents. Final measurements of release levels would be conducted during the decontamination phase and then would be compared with the current release limits. Emphasis is based on fixed and smearable alpha contamination. The instruments used were:

1. Technical Associates Model PUG IAB and a PAS 9 100 cm^2 alpha scintillation detector. The determined efficiency ranged from 16 to 17% using an NBS traceable Th^{230} check source. The grid protecting the detector area covered about 30% of the detection area making it necessary to multiply the observed readings by a factor of 1.4 to obtain the count for 100 cm^2. A multiplication constant of 8.5 was used to obtain dpm/100 cm^2. Because a smear area of 1 ft^2 was used, the meter reading could be directly interpreted as dpm $\pm$30%. However, more precise measurement methods were usually made.

2. Eberline Instrument Corporation Model E-530 with a thin end-window (2-4 mg/cm^2) GM probe (HP-210). The determinded efficiency was 11 to 13% for gamma energies of 0.662 MEV and 8 to 10% for Th^{230} (NBS traceable standards). This instrument was provided primarily as a backup instrument.

Smears were taken on a surface area totaling 1 ft^2 ($\sim$930 cm^2). The purpose of the large smear area was to provide an average assessment of larger areas than that provided by a 100-cm^2 smear. Also, sensitivity is increased if a larger smear tab is used. The material used to obtain wipes for this investigation was oil-impregnated "Masolinn" cleaning cloth cut in 5 x 5 in. squares. This size allowed the entire smear surface to be placed in the detection area and provided sufficient surface area to prevent smear saturation from dirt that would likely occur if conventional size smear tabs were used. While the "Masolinn" smear matrix and oil presented some burial and absorption difficulties for alpha, counting the oil base, helps to pick up more material on the smear cloth compared with a conventional smear tab. This method usually

involves a conservative evaluation.

While not precisely true, it was assumed that an alpha count rate of $\leq$25 cpm on the Technical Associates instrument was approximately background. Measurements at specific locations were made for periods of 1 to 5 minutes and the average meter reading was accepted as the final measurement. The emphasis was to measure the magnitude of contamination and the effectiveness of decontamination techniques and not to measure release limits of contamination in this survey effort.

B. BUILDING SURVEYS, DECONTAMINATION METHODS AND COST ANALYSIS

1. Buildings 1 and 2

While contamination levels of Buildings 1 and 2 differ considerably, the design and contamination complexity are essentially the same. Current and previous survey data are in reasonable agreement. In some instances, smears were taken as indicated in previous surveys on materials containing several centimeters of dust. This technique is then not unlike smearing a dirt floor and is not truly representative of smearable contamination. It therefore is not representative when compared with most other collected smears.

Throughout these structures the insulation is torn and open; contamination of the insulation is evident. In areas where the insulation is intact the covered metal beneath it is usually below MDA levels.

Direct and smear surveys were made of vertical surfaces in both Buildings 1 and 2 about mid-level (or $\sim$12 ft above the ground). The levels of contamination are less than that on the lower walls measured in the previous surveys by a factor of 2. At ceiling elevation in Building 2 the contamination levels are less by a factor of 3, while Building 1 is lower at most by a factor of 2. (Lower walls range from 5,000 to 10,000 dpm/100 cm^2 direct alpha versus ceiling levels of 2,000 to 3,000 dpm/100 cm^2 direct alpha.) Building 2 ceiling contamination levels on horizontal beams range from 200 to 800 dpm/100 cm^2 direct alpha reading.

There are several factors involved that complicate decontamination of these two structures.

(a) Smears from between accessible wall joints at lower elevations ($\leq$6 ft) indicate alpha activity ranging from 50 to 5,000 dpm/100 cm^2. Fixed alpha contamination on lower sections of Building 1 range from 2,000 to 20,000 dpm/100 cm^2 and for Building 2, 500 to 4,000 dpm/100 cm^2. Indications are that joint-located activity extends in Building 1 at least as high as 10 ft at some locations and 6 ft in Building 2. Decontamination would entail

removing each section and replacing it in its original location.

(b) The insulation is not intact in either building. Decontamination would include replacement with new insulation. This entails the removal of exterior paneling.

(c) In most areas surveyed, the steel framework is rusted, and the decontamination efforts revealed deeper layers of contamination. The use of strong reagents and wire brushing of the beams are necessary on all exposed sides and possibly unexposed areas.

(d) The concrete foundations of the superstructure framework are highly contaminated and the high porosity of the rough-cast concrete does not lend itself to easy means of decontamination. Complete decontamination could weaken the structural supports.

(e) The exterior painted surfaces on the structures are oxidized and direct surveys to heights of 7 ft indicate average levels of 200 to 3,000 dpm/100 cm^2. Removal by solvents or other commercial decontamination agents will be required.

The above complications make decontamination of these structures expensive. The following is a possible decontamination method (after removal of general site contamination) for Buildings 1 and 2, with accompanying costs:

a.	Set up equipment - staging, including conduct of health physics indoctrination.	$ 15,000
b.	Wet structures to remove loose dust and to reduce airborne contamination potential.	400
c.	Remove gutters, roof and side panels.	25,000
d.	Remove and package the bulk of insulation and fixtures.	4,800
e.	Decontaminate and wire-brush the steel framework.	20,000
f.	Remove interior and surrounding contaminated earth and debris. (Included in on-site decontamination.)	
g.	Decontaminate and/or replace foundation.	12,800

h.	Replace insulation.	$ 5,000
i.	Decontaminate and reassemble roof and panels	42,000
j.	Final cleanup and release, and miscellaneous.	5,000
	Subtotal	$130,000
	Contingency	26,000
	Total	$156,000

Disposal of Buildings 1 and 2 and Replacement Cost Analysis

The plan and costs are as follows:

(1)	Indoctrinate workers; fix contamination; remove insulation and panels, including roof; remove steel framework; and remove foundation.	$ 26,000
(2)	Construct new buildings.	109,000
	Total	$135,000

2. Building 3

The recent survey of this "garage" structure was in good agreement with previous survey results. Measurements made on th floor surfaces revealed considerable contamination adhered to grease and dust on the floor. It could be said the contaminatio is semi-fixed. There is evidence that, as in Building 4, contam ination is due primarily to vehicular and personnel transfer as well as environmental transport (wind). Smears of vertical surfaces were at or near minimum detectable activity (MDA) levels. Smears from between metal building joints also indicated equal tc or less than MDA levels. It is evident that contamination has not penetrated these seams. Decontamination efforts were conduct ed on clean concrete, grease stained concrete, frame members (channel irons) and the vertical metal walls. Indications are that commercial decontamination agents such as Turco products wil remove the contamination from the walls, frame, and concrete. Difficult areas can be sandblasted. The survey results are included in Table A-I.

The use of commercial decontamination agents such as Turco products will effectively decontaminate this structure. Concrete cleaning can be accomplished by the above agents and sandblastin in a tent.

The contamination level on the outside concrete pad varies

from 500 to 40,000 dpm/100 cm^2 direct alpha reading. By examining broken segments and soil (turning over broken pieces) it was determined that the soil is not appreciably contaminated. Expansion joints and cracks permit contamination to enter and leach below this concrete pad. Removal of this pad is the best method of decontamination; a small layer of soil should also be removed. The average exterior contamination on the structure is 300 dpm/100 cm^2 direct alpha. The lower 3 ft of the building indicates levels to 1,500 dpm/100 cm^2 direct alpha and up to 500 dpm/100 cm^2 smearable alpha. Above 6 ft, the smearable alpha encountered did not exceed 50 dpm/100 cm^2. Composite smears between outside joints did not reveal contamination equal to or less than MDA. Removal of soil and foundation sandblasting (exterior) may be required.

The following decontamination method is best suited for this structure:

(a) Enclose the structure in plastic sheeting while other phases of site decontamination are occurring (earthmoving).

(b) Remove the exterior concrete and earth up to the present foundation and check for contamination. Sandblast as required after covering clean area with tarps or plastic.

(c) Decontaminate the exterior of the building using commercial agents to remove oxide film layer and sandblast or wire-brush residual "hot spots", using a tent enclosure for sandblasting.

(d) Remove waste material and replastic outside surfaces.

Note: Liquid waste can be captured by the use of absorbent clays placed around the exterior, on top of plastic sheeting.

(e) Mop up loose contamination on the interior floors and walls.

(f) Remove the interior oxide film layer and the liquid capture method, by means of a commercial decontamination agent, as outlined in note under (d) above.

(g) Remove absorbent clays and apply a commercial concrete cleaner or decontamination agent to the concrete floor. Sandblast trouble spots.

(h) Remove generated waste and release structures.

The cost estimate, excluding removal and/or storage costs of the contaminated residence is:

Plastic covering	$ 1,044
Remove concrete and contaminated earth	560
Decontaminate exterior, using agents, sandblasting, etc.	1,600
Decontaminate interior, using agents, sandblasting, etc.	2,400
Sandblast foundation, as required	200
Final cleanup	320
Health physicist technician, including travel and expenses	3,000
Miscellaneous equipment and supplies	2,000
Subtotal	$11,880
Contingency	2,375
Total	$15,300

The estimated cost to demolish the building and replace it with the same square and cubic footage would be:

Demolish	$ 4,000
New structure	12,000
Subtotal	$16,000
Contingency	1,600
Total	$17,600

This cost, also, does not include removal and storage of the contaminated debris.

3. Building 4

The survey results of this structure were in good agreement with past survey data. One location was noted to be higher than previous information (850 dpm/100 cm^2 direct alpha reading). Spot checks were conducted in drains, duct work, under loose floor tile, and other inaccessible surfaces. (Alpha contamination at these locations if present was below the minimum detectable activity for the survey instrument and techniques used.) The walls, ceilings, and other vertical surfaces were also MDA. The principal area of contamination is the floor and evidence indicates the mechanism of

contamination was by personnel transfer and environmental transport (wind). By simply wiping the contaminated floor with an oil-impregnated dust cloth, a decontamination factor of 2 was achieved. Application of other agents (soap, water, dilute citric acid) produced a decontamination factor of 2 to 5. Abrasion of the agents with a stiff wire brush reduced the contamination to MDA levels.

An exterior survey indicated levels of fixed and loose alpha contamination ranging from 200 to 500 dpm/100 cm^2. The asphalt parking lot, walkway and curbing indicated levels of the same magnitude. There is evidence of the lawn containing contaminated soil.

The following decontamination method is best suited for this structure:

(a) Enclose the structure in plastic sheeting while other phases of site decontamination are occurring (earthmoving).

(b) Remove the parking lot, walkway, and top layer of soil from around the building and ensure that the exterior foundation and underground brick is not contaminated. Cover uncontaminated soil with canvas or plastic.

(c) Use a commercial concrete decontamination agent (such as Turco) or sandblast the exterior using a tent to prevent contamination dispersion.

(d) Remove the residual agents and/or sand and apply a light coating of tar to the roof.

(e) Remove the first layer of roofing paper, but first use commercial agents to decontaminate exterior exhaust and vent work.

(f) Concurrently, scrub the interior ceiling and walls, or if less costly remove paneling and replace. Use a commercial decontamination agent such as Turco.

(g) Mop the floors to remove loose contamination and then apply a commercial concrete cleaner to the interior exposed concrete floors and sandblast the remaining "hot spots".

(h) Mop tiles to remove loose contamination, spray tiles with an inexpensive clear lacquer or fast-drying paint (remove tiles and moldings).

(i) Clean windows with a dilute acid solution or commercial window cleaner (use brushes in corners of windows).

The cost estimate for Building 4 is as follows:

Plastic covering	$ 812
Sandblast and clean with commercial agents (interior, exterior) including equipment	1,200
Remove part of roofing and replace, including material	1,200
Labor to clean and assist health physicist technician	1,920
Remove and replace floor tiles, includes tiles	920
Health physicist technician coverage, including travel and expenses	1,500
Subtotal	$ 6,740
Contingency	1,348
Total	$ 8,900

This cost does not include removal and storage costs of contaminated materials. Were this building to be demolished and replaced, the sequence of operations would be:

(a) Fix contamination

(b) Demolish - disassemble

(c) Load debris

(d) Ship to burial site

The cost estimate, excluding removal and storage costs for contaminated material is:

Fix contamination	$ 1,600
Demolish - disassemble structure	5,000
Health physicist technician	1,500
Estimated rebuilding of the structure	40,000
Subtotal	$48,100
Contingency	4,810
Total	$52,910

4. Cost Summary

A cost summary to decontaminate, construct and/or reconstruct the four buildings is presented in Table A-II.

C. SUMMARY

The survey and decontamination investigation indicates that the previous and current data is comparable. Differences in measurements are arrived at by the types of measurements conducted, whether absolute or average value measurements are being conducted. A contingency has been added to the decontamination, disposal and reconstruction efforts. Transportation and disposal costs add considerably to the estimated costs for the on-site work. While low-level contamination provides means for bulk LSA (low specific activity) shipments, the volume of contaminated site material is the governing cost factor. An advantage of commercial disposal site burial is the release of responsibility for the material.

TABLE A-I

DECONTAMINATION EFFORTS

Item	Previous survey (direct)	Current survey (direct/smear)	After Masolinn wipe (direct/smear)	D.F.**	After use of Decon agent (direct/smear)	D.F.	Total D.F.
Clean concrete	0.9 k*	2 k/0.2 k	1.8k/∿0.05k	1.1/4	≤0.2k/MDA	9/≥2.5	10/≥10
Greasy concrete	3.3 k	2.5k/0.18k	1.7k/∿0.08k	∿1.5/∿2.3	0.3k/MDA	∿5.7/≥4	8.3/≥9
Frame channel	--	0.3k/0.1k	≤0.2k/MDA	≥1.5/≥5	MDA/MDA	≥1/--	≥1.5/≥5
Wall	1.2 k	1.7k/0.2k	0.8k/MDA	2.1/≥10	≤0.2k/MDA	≥4/--	8.5/≥10

(All measurements in dpm/100 cm^2 alpha.)

*k = 1,000
**D.F. = Decontamination factor

TABLE A-II

COST SUMMARY*

Building	Decontaminate	Demolish - Reconstruct
1 and 2	$ 156,000	$ 135,000
3	14,500	17,600
4	8,100	52,900
Total	$ 178,600	$ 205,500

*Costs do not include hauling to a long-term storage site.

APPENDIX D.2

REMEDIAL ACTION CRITERIA

B.1 Surgeon General's Guidelines

B.2 Radiological Criteria for Decontamination of Inactive Uranium Mill Sites

B.3 Grand Junction Remedial Criteria (10CFR712)

B.4 Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material

APPENDIX D.2

REMEDIAL ACTION CRITERIA

The remedial action criteria used for the Phase II assessment of the cleanup of mill tailings are presented in the following documents:

B.1 SURGEON GENERAL'S GUIDELINES

DEPARTMENT OF HEALTH, EDUCATION AND WELFARE,
PUBLIC HEALTH SERVICE,
Washington, D. C., July 1970.

DR. R. L. CLEERE,
Executive Director, Colorado State Department of Health, 4210 E. 11th Avenue, Denver, Colorado

DEAR DR. CLEERE: I am pleased to respond to your letter of January 29 in which you asked Dr. M. W. Carter, Director of our Southwestern Radiological Health Laboratory, for Public Health Service and/or U. S. Atomic Energy Commission assistance in providing exposure guidelines applicable to homes with high concentrations of radon progeny.

The enclosed graded recommendations for action have been developed within the framework of existing Federal Radiation Council guidance for occupational exposure to airborne concentrations of radon and its daughters (progeny). Also, graded action levels applicable to external gamma radiation are included.

You will note in the accompanying Explanatory Notes that these recommendations apply specifically to dwellings constructed with or on uranium mill tailings. Further qualifications in the Explanatory Notes should be consulted before these recommendations are applied.

The specific information which your Department is developing on the variability of radon daughter concentrations in dwellings and on optimum control measures will be essential towards making those decisions necessary in applying the recommendations.

These recommendations have been directed to the Atomic Energy Commission for comment. Because of the urgency attached to your receiving the recommendations as soon as possible, they have been forwarded to you in advance of receiving AEC views and comments. We will advise you of the AEC response when received.

Sincerely yours,

PAUL J. PETERSON,
Acting Surgeon General

Enclosure:

RECOMMENDATIONS OF ACTION FOR RADIATION EXPOSURE LEVELS IN DWELLINGS CONSTRUCTED ON OR WITH URANIUM MILL TAILINGS

External gamma radiation:

Level:	Recommendations
Greater than 0.1 mR/hr . . .	Remedial action indicated.
From 0.05 to 0.1 mR/hr . . .	Remedial action may be suggested.
Less than 0.05 mR/hr . .	No action indicated.

Level:	Recommendations
Greater than 0.05 WL	Remedial action indicated.
From 0.01 to 0.05 WL	Remedial action may be suggested.
Less than 0.01 WL	No action indicated.

EXPLANATORY NOTES

1. These recommendations are written specifically for dwellings constructed on or with uranium mill tailings. This situation may involve continuous exposure of members of the public to radon daughter product activities and whole-body gamma irradiation levels in excess of the background radiation levels found within dwellings in the area not constructed with or on uranium mill tailings.

2. Although the initial concern was the presence of radon daughter product activities within these dwellings, preliminary surveys have indicated that in some instances, the gamma radiation levels were of prime importance. Thus, recommendations are made concerning both types of radiation. The recommendations applicable to a particular dwelling will be determined by whichever type of radiation has the high level.

3. Three levels for action are recommended for both external gamma and radon daughter product exposures. This graded system of actions is proposed to allow latitude in the middle ranges for the judgment of the on-site investigators.

4. The external gamma and radon daughter product levels proposed constitute exposures which are in addition to the natural background levels found within dwellings in the area not constructed on or with uranium mill tailings. In the Grand Junction, Colorado, area these levels are approximately 0.01 mR/hr (approximately 90 mrem/yr) and 0.004 Working Levels (WL) (approximately 0.2 CWLM/yr) respectively (1).

5. The expected health effects of concern will be different for the two types of radiation; i.e., leukemia for whole-body gamma radiation exposure and lung cancer for exposure to inhaled radon daughter products. This expectation is based, in part, on findings derived from population studies such as the Japanese atomic bomb

survivors and uranium miners. These specific health effects are considered to be mutually exclusive. The basis for this assumption is that the expected radiation contribution to whole-body exposure from inhaled radon and daughter products would be considerably less than the direct exposure from external gamma radiation at the levels encountered in the dwellings. Conversely, the external gamma radiation contribution to the lung dose is considered to comprise a negligible additional risk of lung cancer.

6. (a) A Working Level (WL) is the term used to describe radon daughter product activities in air. This term is defined as any combination of short-lived radon daughter products in 1 liter of air that will result in the ultimate emission of 1.3×10^5 MeV of potential alpha energy (2). The numerical value of the WL is derived from the alpha energy released by the total decay through Ra C' of the short-lived radon daughter products, Ra A, Ra B and Ra C, at radioactive equilibrium with 100 pCi of ^{222}Rn per liter of air (3).

6. (b) A Working Level Month (WLM) is the term used to express the occupational exposure incurred in one working month of 170 hours by a uranium miner laboring in an atmosphere containing radon daughter products; i.e., one working month in a mine atmosphere containing 1 WL of radon daughter products equals 1 WLM.

6. (c) Cumulative Working Level Months (CWLM) is the term used to express the total accumulated occupational exposure to radon daughter products in air; i.e., an air concentration of radon daughter products of 1 WL would, in one working month, equal 1 WLM, and in 1 year or 12 months would equal 12 CWLM.

6. (d) Since occupational exposures are based upon 170 hours per month and continuous exposure involves approximately 170 hours per week, then an occupational exposure to an air concentration of 1 WL is equivalent to continuous exposure to 0.025 WL.

7. These recommendations are based on the assumption of a linear, non-threshold dose-effect relationship. The lack of definitive information precludes allowances for possible differences in radio-sensitivity due to age, sex, or other biological characteristics.

8. No action is indicated when the external gamma exposure rate is less than 0.05 mR/hr and the radon daughter product activity is less than 0.01 WL since under conditions of continuous exposure these levels would result in maximum annual exposures of approximately 400 mrem and 0.5 CWLM, respectively. The maximum annual value of 400 mrem is less than the dose limits recommended for an individual body exposure to external gamma irradiation.

The ICRP (5) recommends that the annual dose limit for members of the public shall be 1/10 of the corresponding annual occupational maximum permissible dose. The maximum annual value of 0.5 CWLM of radon daughter product exposure is approximately 1/10 of the 4 CWLM annual occupational exposure limit recommended by the FRC (6) for implementation on 1 January 1971, and less than 1/20 of the

annual occupational exposure limit of 12 CWLM recommended for uranium miners in the present FRC regulations (4).

9. Remedial action may be suggested in the case of external gamma exposure rates of 0.05-0.10 mR/hr or radon daughter product activities of 0.01-0.05 WL since under conditions of continuous exposure these levels would result in maximum annual exposures of approximately 400-900 mrem and 0.5-2.5 CWLM. The upper limit of these ranges exceeds the strictly applied recommendations of the FRC and ICRP for exposures of an individual member of the public. However, this extension seems justified in situations in which unforeseen exposures have occured, since as stated by ICRP (5) "in general it will be appropriate to institute countermeasures only when their social cost and risk will be less than those resulting from the exposure." It is further stated by the ICRP (5) that very low levels of risk are implied in the dose limits for members of the public and that it is likely to be of minor consequence to their health if the dose limits are marginally or even substantially exceeded.

10. Remedial action is indicated at gamma exposures greater than 0.1 mR/hr or at radon daughter product activities greater than 0.05 WL. Under conditions of continuous exposure, these levels would result in minimum annual exposures of 900 mrem and 2.5 CWLM. All values above these would indicate the necessity for remedial action, since at these levels the maximum annual exposures recommended by the FPC and ICRP for an individual member of the public is exceeded.

11. With respect to the external gamma irradiation, from the estimates published by ICRP (7), it can be interpolated that the annual risk of leukemia under conditions of continuous exposure to 500 mrem per year is an increased incidence of about 10 cases per year per million persons exposed. The natural annual incidence of leukemia for all ages is given by ICRP (8) as 10-100 cases per million persons. With respect to radon daughter product exposures, it has been estimated by Archer and Lundin (9) that an exposure of 120 CWLM to a group of white adult males in the United States appears to approximately double the normal lung cancer incidence which for this population is about 2-3 cases per year per 10,000 persons. At an annual exposure of 2.5 CWLM, 48 years would be required to reach 120 CWLM.

12. It is considered that implementation of these recommendations for the various exposure ranges would make it highly unlikely that any serious health effects would result from exposure to radon daughter products or external gamma irradiation in this particular situation.

13. It is suggested that remedial action be taken only after an adequate number of measurements taken under a diversity of temporal and climatic conditions have clearly established that the average exposure is in excess of 0.1 mR/hr or 0.05 WL exist and in instituting corrective measures. However, it is considered that the additional health risks from continued exposure over this time period are of lesser consequence than the economic and social discomfitures of precipitous action.

Approved.

/s/ PAUL J. PETERSON,
for Jesse L. Steinfeld, M.D.,
Surgeon General, Public Health Service

July 27, 1970

REFERENCES

1. Personal communication, Mr. Robert D. Siek, Colorado State Department of Health.

2. U.S. Public Health Service Publication No. 494, Control of Radon and Daughters in Uranium Mines and Calculations on Biologic Effects, 1957.

3. Federal Radiation Council Report No. 8 Revised, Guidance for the Control of Radiation Hazards in Uranium Mining, 1967.

4. Federal Radiation Council Report No. 1. Background Material for the Development of Radiation Protection Standards, 1960.

5. Recommendations of the International Commission on Radiological Protection, ICRP Publication 9, 1966.

6. Federal Register, Vol. 34, No. 10, pp 576-577, 1969.

7. The Evaluation of Risks from Radiation, ICRP Publication 8, 1966.

8. Radiosensitivity and Spatial Distribution of Dose, ICRP Publication 14, 1969.

9. V.E. Archer and F. E. Lundin, Jr., Radiogenic Lung Cancer in Man: Exposure-Effect Relationship, Environmental Research 1, pp 370-383, 1967.

B.2 RADIOLOGICAL CRITERIA FOR DECONTAMINATION OF INACTIVE URANIUM MILL SITES*

1. General

Radiological criteria for an engineering assessment of possible remedial actions applicable to uranium mill tailings piles and for the decontamination of inactive uranium mill sites are provided herein. These criteria are applicable to the sites, to their surrounding areas which have been contaminated by radioactive materials from the sites, and to buildings in which the materials have been used.

Critical radiation exposure pathways from inactive uranium mill sites to members of the general population are:

(a) Radon escaping from the tailings pile carried by the wind into habitable structures where the holdup time is long enough, resulting in buildup of radon daughters to levels greater than the ambient air.

(b) Tailings material used for construction of habitable structures can result in a buildup of radon daughters and increased gamma levels.

(c) Gamma rays from tailings material cause whole body radiation exposure. This includes not only the "gamma shine" from the tailings pile that exposes people living nearby, but also the radiation exposure from tailings material that has been eroded off the pile onto surrounding land. The mill sites always show elevated gamma exposure levels because of contamination by ore, tailings solids, and process solutions.

(d) ^{226}Ra, Th, and other radionuclides from tailings piles can be leached into ground water and thereafter into public and irrigation water supplies.

(e) Windblown particulate material (Ra and Th) from the tailings pile can be inhaled causing a radiation dose to the lung.

Remedial actions may be required on inactive uranium mill tailings piles to reduce or prevent excess radiation exposure from radon progeny, gamma radiation, ^{226}Ra, and radioactive particulate material. If tailing material has been used as a building material, remedial actions may be required to reduce radon concentrations and/or gamma activity levels. Remedial actions performed on tailings piles

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*Provided by U S Environmental Protection Agency, as attachment to letter dated Dec 1974.

and decontamination of mill sites and surrounding contaminated areas should result in residual exposures that are as low as practicable. There is no single permissible exposure level applicable to all such cases. An evaluation should be made on a case-by-case basis of the risk involved, balanced against (1) the cost of reducing the residual contamination, and (2) the economic effect on alternatives such as restricting the use of the land. The result of such an analysis can be used by all concerned to define the "as low as practicable" residual level of contamination that will be acceptable and determine whether restrictions will be required on the use of any contaminated land.

2. Tailings Pile or Pond

The operation of uranium mills results in the generation of waste material which is disposed of in tailings piles and ponds. Environmental contamination has occurred at those sites where measures were not taken to control the movement of the radioactive material. In order to restore the environmental quality and provide for protection of the public, such sites should be decontaminated and result in residual gamma radiation levels which are as low as practicable. For most situations this would require decontamination of the area by (1) removal of radioactive material to a location where the material would be isolated from the biosphere, or (2) providing sufficient cover such that the resultant gamma radiation levels are as low as practicable, preferably at background. However, under certain topographical conditions and economic considerations wherein complete removal is not practicable, the residual levels should not exceed 40 μR/hr above background. This value is arbitrarily chosen for the purpose of providing an engineering estimate on cleanup of contaminated areas. It is considered to be sufficiently low that the expected exposures occurring after any remedial action at this level would not constitute a public concern. However, this should not be considered as the final criterion.* The gamma radiation level is the net, corrected measurement at 3 ft above the ground.

For each site a determination should be made of the radium concentration in the soil. Cleanup should reduce the soil concentration to less than two times the radium background specific for the area.

If the radioactive material remains in place and stabilized, the area should be designated as a controlled area. Due to the difficulty of controlling radon diffusion and the existing state-of-the-art of stabilization, the land should be restricted as to human occupancy and be properly fenced to limit access.

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*When all phase II information is complete and the health impact of remedial actions identified an overall determination of as low as practicable protection levels can be assessed appropriately. Therefore, the above numbers are subject to change.

The ^{226}Ra activity contribution from the site in ground or surface water should meet applicable state or federal standards.

3. Open Land Areas

This area refers to all land beyond the fence of the sites where tailings are located. As with the tailings areas, decontamination of the uranium mill site and other areas contaminated by wind- or water-eroded tailings should result in residual gamma levels which are as low as practicable. Cleanup of the area would require returning of the windblown tailings material to the site and establishing a controlled area, or moving all the material to a location that will isolate the material from the biosphere.

If the residual gamma levels are less than 10μR/hr above background, the land may be released for unrestricted use. If residual levels are equal to or greater than 10μR/hr above background at a given site a determination should be made of the radium concentration in the soil. Cleanup should reduce the soil concentration to no more than two times the radium background specific for the area. Under certain topographical conditions wherein complete removal of tailings is not possible or practicable, the residual levels should be as low as practicable but should not exceed 40μR/hr above background and access should be controlled. This value is arbitrarily chosen for the purpose of providing an engineering estimate on cleanup of contaminated areas. The gamma radiation level is the net, corrected measurement at 3 ft above the ground.

4. Structures

It is possible that there will be several industrial and residential structures where tailings have been utilized for construction purposes. When it has been determined that tailings were used in the construction, the lower limits of the guidelines established by the Surgeon General for structures in Grand Junction, Colorado, will be used.

PART 712—GRAND JUNCTION REMEDIAL ACTION CRITERIA

Sec.
712.1 Purpose.
712.2 Scope.
712.3 Definitions.
712.4 Interpretations.
712.5 Communications.
712.6 General radiation exposure level criteria for remedial action.
712.7 Criteria for determination of possible need for remedial action.
712.8 Determination of possible need for remedial action where criteria have not been met.
712.9 Factors to be considered in determination of order of priority for remedial action.
712.10 Selection of appropriate remedial action.

AUTHORITY: Sec. 203, 86 Stat. 226.

§ 712.1 Purpose.

(a) The regulations in this part establish the criteria for determination by ERDA of the need for, priority of and selection of appropriate remedial action to limit the exposure of individuals in the area of Grand Junction, Colo., to radiation emanating from uranium mill tailing which have been used as a construction-related material.

(b) The regulations in this part are issued pursuant to Pub. L. 92-314 (86 Stat. 222) of June 16, 1972.

§ 712.2 Scope.

The regulations in this part apply to all structures in the area of Grand Junction, Colo., under or adjacent to which uranium mill tailings have been used as a construction-related material between January 1, 1951, and June 16, 1972, inclusive.

§ 712.3 Definitions.

As used in this part:

(a) "Administrator" means the Administrator of Energy Research and Development or his duly authorized representative.

(b) "Area of Grand Junction, Colo." means Mesa County, Colo.

(c) "Background" means radiation arising from cosmic rays and radioactive material other than uranium mill tailings.

(d) "ERDA" means the U.S. Energy Research and Development Administration or any duly authorized representative thereof.

(e) "Construction-related material" means any material used in the construction of a structure.

(f) "External gamma radiation level" means the average gamma radiation exposure rate for the habitable area of a structure as measured near floor level.

(g) "Indoor radon daughter concentration level" means that concentration of radon daughters determined by: (1) Averaging the results of 6 air samples each of at least 100 hours duration, and taken at a minimum of 4-week intervals throughout the year in a habitable area of a structure, or (2) utilizing some other procedure, approved by the Commission.

(h) "Milliroentgen (mR) means a unit equal to one-thousandth (1/1000) of a roentgen which roentgen is defined as an exposure dose of X or gamma radiation such that the associated corpuscular emission per 0.001293 gram of air produces, in air, ions carrying one electrostatic unit of quantity of electricity of either sign.

(i) "Radiation" means the electromagnetic energy (gamma) and the particulate radiation (alpha and beta) which emanate from the radioactive decay of radium and its daughter products. 56778

(j) "Radon daughters" means the consecutive decay products of radon-222. Generally, these include Radium A (polonium-218), Radium B (lead-261), Radium C (bismuth-214), and Radium C' (polonium-214).

(k) "Remedial action" means any action taken with a reasonable expectation of reducing the radiation exposure resulting from uranium mill tailings which have been used as construction-related material in and around structures in the area of Grand Junction, Colo.

(l) "Surgeon General's guidelines" means radiation guidelines related to uranium mill tailings prepared and released by the Office of the U.S. Surgeon General, Department of Health, Education and Welfare on July 27, 1970.

(m) "Uranium mill tailings" means tailings from a uranium milling operation involved in the Federal uranium procurement program.

(n) "Working Level" (WL) means any combination of short-lived radon daughter products in 1 liter of air that will result in the ultimate emission of 1.3×10^5 MeV of potential alpha energy.

§ 712.4 Interpretations.

Except as specifically authorized by the Administrator in writing, no interpretation of the meaning of the regulations in this part by an officer or employee of ERDA other than a written interpretation by the General Counsel will be recognized to be binding upon ERDA.

(b) Where ERDA approved data on indoor radon daughter concentration levels are not available:

(1) For dwellings and schoolrooms:

(i) An external gamma radiation level of 0.05 mR/hr. or greater above background.

(ii) An indoor radon daughter concentration level of 0.01 WL or greater above background (presumed).

(A) It may be presumed that if the external gamma radiation level is equal to or exceeds 0.02 mR/hr. above background, the indoor radon daughter concentration level equals or exceeds 0.01 WL above background.

(B) It should be presumed that if the external gamma radiation level is less than 0.001 mR/hr. above background, the indoor radon daughter concentration level is less than 0.01 WL above background, and no possible need for remedial action exists.

(C) If the external gamma radiation level is equal to or greater than 0.001 mR/hr. above background but is less than 0.02 mR/hr. above background, measurements will be required to ascertain the indoor radon daughter concentration level.

(2) For other structures: (i) An external gamma radiation level of 0.15 mR/hr. above background averaged on a room-by-room basis.

(ii) No presumptions shall be made on the external gamma radiation level/indoor radon daughter concentration level relationship. Decisions will be made in individual cases based upon the results of actual measurements.

§ 712.3 Determination of possible need for remedial action where criteria have not been met.

The possible need for remedial action may be determined where the criteria in § 712.7 have not been met if various other factors are present. Such factors include, but are not necessarily limited to, size of the affected area, distribution of radiation levels in the affected area, amount of tailings, age of individuals occupying affected area, occupancy time, and use of the affected area.

§ 712.5 Communications.

Except where otherwise specified in this part, all communications concerning the regulations in this part should be addressed to the Director, Division of Safety, Standards, and Compliance, U.S. Energy Research and Development Administration, Washington, D.C. 20545.

§ 712.6 General radiation exposure level criteria for remedial action.

The basis for undertaking remedial action shall be the applicable guidelines published by the Surgeon General of the United States. These guidelines recommend the following graded action levels for remedial action in terms of external gamma radiation level (EGR) and indoor radon daughter concentration level (RDC) above background found within dwellings constructed on or with uranium mill tailings:

EGR	RDC	Recommendation
Greater than 0.1 mR/hr.	Greater than 0.05 WL.	Remedial action indicated.
From 0.05 to 0.1 mR/hr.	From 0.01 to 0.05 WL.	Remedial action may be suggested.
Less than 0.05 mR/hr.	Less than 0.01 WL.	No remedial action indicated.

§ 712.7 Criteria for determination of possible need for remedial action.

Once it is determined that a possible need for remedial action exists, the record owner of a structure shall be notified of that structure's eligibility for an engineering assessment to confirm the need for remedial action and to ascertain the most appropriate remedial measure, if any. A determination of possible need will be made if as a result of the presence of uranium mill tailings under or adjacent to the structure, one of the following criteria is met:

(a) Where ERDA approved data on indoor radon daughter concentration levels are available:

(1) For dwellings and schoolrooms: An indoor radon daughter concentration level of 0.01 WL or greater above background.

(2) For other structures: An indoor radon daughter concentration level of 0.03 WL or greater above background.

§ 712.9 Factors to be considered in determination of order of priority for remedial action.

In determining the order of priority for execution of remedial action, consideration shall be given, but not necessarily limited to, the following factors:

(a) Classification of structure. Dwellings and schools shall be considered first.

(b) Availability of data. Those structures for which data on indoor radon daughter concentration levels and/or external gamma radiation levels are available when the program starts and which meet the criteria in § 712.7 will be considered first.

(c) Order of application. Insofar as feasible remedial action will be taken in the order in which the application is received.

(d) Magnitude of radiation level. In general, those structures with the highest radiation levels will be given primary consideration.

(e) Geographical location of structures. A group of structures located in the same immediate geographical vicinity may be given priority consideration particularly where they involve similar remedial efforts.

(f) Availability of structures. An attempt will be made to schedule remedial action during those periods when remedial action can be taken with minimum interference.

(g) Climatic conditions. Climatic conditions or other seasonal considerations may affect the scheduling or certain remedial measures.

§ 712.10 Selection of appropriate remedial action.

(a) Tailings will be removed from those structures where the appropriately averaged external gamma radiation level is equal to or greater than 0.05 mR/hr. above background in the case of dwellings and schools and 0.15 mH/hr. above background in the case of other structures.

(b) Where the criterion in paragraph (a) of this section is not met, other remedial action techniques, including but not limited to sealants, ventilation, and shielding may be considered in addition to that of tailings removal. ERDA shall select the remedial action technique or combination of techniques, which it determines to be the most appropriate under the circumstances.

B.4 GUIDELINES FOR DECONTAMINATION OF FACILITIES AND EQUIPMENT PRIOR TO RELEASE FOR UNRESTRICTED USE OR TERMINATION OF LICENSES FOR BYPRODUCT, SOURCE, OR SPECIAL NUCLEAR MATERIAL*

The instructions in this guide in conjunction with Table I specify the radioactivity and radiation exposure rate limits which should be used in accomplishing the decontamination and survey of surfaces or premises and equipment prior to abandonment or release for unrestricted use. The limits in Table I do not apply to premises, equipment, or scrap containing induced radioactivity for which the radiological considerations pertinent to their use may be different. The release of such facilities or items from regulatory control will be considered on a case-by-case basis.

1. The licensee shall make a reasonable effort to eliminate residual contamination.

2. Radioactivity on equipment or surfaces shall not be covered by paint, plating, or other covering material unless contamination levels, as determined by a survey and documented, are below the limits specified in Table I prior to applying the covering. A reasonable effort must be made to minimize the contamination prior to use of any covering.

3. The radioactivity on the interior surfaces of pipes, drain lines, or ductwork shall be determined by making measurements at all traps, and other appropriate access points, provided that contamination at these locations is likely to be representative of contamination on the interior of the pipes, drain lines, or ductwork. Surfaces of premises, equipment, or scrap which are likely to be contaminated but are of such size, construction, or location as to make the surface inaccessible for purposes of measurement shall be presumed to be contaminated in excess of the limits.

4. Upon request, the Commission may authorize a licensee to relinquish possession or control of premises, equipment, or scrap having surfaces contaminated with materials in excess of the limits specified. This may include, but would not be limited to, special circumstances such as razing of buildings, transfer of premises to another organization continuing work with radioactive materials, or conversion of facilities to a long-term storage or standby status. Such requests must:

   a. Provide detailed, specific information describing the

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*From U.S. Nuclear Regulatory Commission, Division of Fuel Cycle and Material Safety, Washington, D.C. 20555, Nov 1976.

premises, equipment or scrap, radioactive contaminants, and the nature, extent, and degree of residual surface contamination.

b. Provide a detailed health and safety analysis which reflects that the residual amounts of materials on surface areas, together with other considerations such as prospective use of the premises, equipment or scrap, are unlikely to result in an unreasonable risk to the health and safety of the public.

5. Prior to release of premises for unrestricted use, the licensee shall make a comprehensive radiation survey which establishes that contamination is within the limits specified in Table I. A copy of the survey report shall be filed with the Division of Fuel Cycle and Material Safety, USNRC, Washington, D.C. 20555, and also the Director of the Regional Office of the Office of Inspection and Enforcement, USNRC, having jurisdiction. The report should be filed at least 30 days prior to the planned date of abandonment. The survey report shall:

a. Identify the premises.

b. Show that reasonable effort has been made to eliminate residual contamination.

c. Describe the scope of the survey and general procedures followed.

d. State the findings of the survey in units specified in the instruction.

Following review of the report, the NRC will consider visiting the facilities to confirm the survey.

TABLE I

ACCEPTABLE SURFACE CONTAMINATION LEVELS

NUCLIDES[a]	AVERAGE[b c f]	MAXIMUM[b d f]	REMOVABLE[b e f]
U-nat, U-235, U-238, and associated decay products	5,000 dpm α/100 cm^2	15,000 dpm α/100 cm^2	1,000 dpm α/100 cm^2
Transuranics, Ra-226, Ra-228, Th-230, Th-228, Pa-231, Ac-227, I-125 I-129	100 dpm/100 cm^2	300 dpm/100 cm^2	20 dpm/100 cm^2
Th-nat, Th-232, Sr-90, Ra-223, Ra-224, U-232, I-126, I-131, I-133	1,000 dpm/100 cm^2	3,000 dpm/100 cm^2	200 dpm/100 cm^2
Beta-gamma emitters (nuclides with decay modes other than alpha emission or spontaneous fission) except SR-90 and others noted above	5,000 dpm $\beta\gamma$/100 cm^2	15,000 dpm $\beta\gamma$/100 cm^2	1,000 dpm $\beta\gamma$/100 cm^2

[a]Where surface contamination by both alpha- and beta-gamma-emitting nuclides exists, the limits established for alpha- and beta-gamma-emitting nuclides should apply independently.

[b]As used in this table, dpm (disintegrations per minute) means the rate of emission by radioactive material as determined by correcting the counts per minute observed by an appropriate detector for background, efficiency, and geometric factors associated with the instrumentation.

[c]Measurements of average contaminant should not be averaged over more than 1 square meter. For objects of less surface area, the average should be derived for each such object.

[d]The maximum contamination level applies to an area of not more than 100 cm^2.

[e]The amount of removable radioactive material per 100 cm^2 of surface area should be determined by wiping that area with dry filter or soft absorbent paper, applying moderate pressure, and assessing the amount of radioactive material on the wipe with an appropriate instrument of known efficiency. When removable contamination on objects of less surface area is determined, the pertinent levels should be reduced proportionally and the entire surface should be wiped.

[f]The average and maximum radiation levels associated with surface contamination resulting from beta-gamma emitters should not exceed 0.2 mrad/hr at 1 cm and 1.0 mrad/hr at 1 cm, respectively, measured through not more than 7 milligrams per square centimeter of total absorber.

APPENDIX D.3

GEOLOGIC AND HYDROLOGIC CONDITIONS OF RAFFINATE PITS AT WELDON SPRING SITE

APPENDIX D.3

GEOLOGIC AND HYDROLOGIC CONDITIONS OF RAFFINATE PITS AT WELDON SPRING SITE

The following information was submitted October 20, 1967 as an attachment to a letter from Willis G. Fish, Chief, Engineering Division, Department of the Army, Kansas City District, Corps of Engineers, to the Commanding Officer at Edgewood Arsenal in Maryland:

MRKED-FG 19 October 1967

GEOLOGIC APPRAISAL OF SETTLEMENT PITS, WELDON SPRING ORDNANCE WORKS

1. On 11 October 1967 the undersigned visited the Weldon Spring Ordnance Works. A meeting was held with Messrs. A. Hilsmeier, Edgewood Arsenal; F. Belcher, Atomic Energy Commission; and F. Pagel, National Lead Company of Ohio. Mr. Hilsmeier requested that we evaluate the leakage possibility from the settlement pits on the reservation. We inspected the pits and were given copies of reports, drawings and boring logs pertaining to the construction and geology of the pits.

2. The study was made by researching the available literature and by a field reconnaissance of both the site and the surrounding countryside. The reports and drawings referred to in preparing this report are listed at the end of the report. The pits are located on the drainage divide of the Missouri and Mississippi Rivers. South of the divide the land surface drops off rapidly to the Missouri River and is characterized by a rugged topography laced with deep, short tributary streams draining to the Missouri, 1-1/2 to 2 miles from the divide. North of the divide the topography is moderate to gentle. Drainage is to Dardenne Creek which flows northeasterly to the Mississippi. The elevation of the crest of the divide is approximately 670 m.s.l. with the Missouri River floodplain at elevation 460 and elevation 470 for the Dardenne Creek floodplain.

3. The pit area is mantled with a relatively thick overburden cover overlying an irregular bedrock surface. The overburden consists of wind-blown deposits (loess) overlying residual soils developed from the weathering of the underlying limestones. The thickness of the overburden is reported to range from 17' to 58' thick. One report mentions the overburden as being glacial drift, however, it is doubtful that there are any extensive drift deposits on the reservation. Isolated remnants of drift may occur below the loess since the area is near the southern limit of glaciation.

4. The uppermost bedrock unit in the area is the Burlington-Keokuk Formation of Mississippian age. This formation is reported to be 150-200 feet thick under the pit area. The unit is a cherty, crinoidal, massive-bedded limestone. The upper 50 to 60 feet is reported to contain solution enlarged joints and small caverns. Deep weathering of this formation has resulted in an irregular, pinnacled bedrock surface. Drilling water losses have been frequently reported in the Burlington-Keokuk. The Burlington-Keokuk is immediately underlain by the Fern Glen and Chouteau Formations also of Mississippian age. They are also limestones and have an aggregate thickness of approximately 100 feet. The Sulfur Springs Formation probably of Devonian age underlies the Chouteau. The Sulfur Springs is primarily a sandstone, the Bushberg, and is less than 10 to as much as 30 feet thick. The Sulfur Springs is underlain by Ordovician and Cambrian limestones and dolomites including the St. Peter Sandstone which is the most dependable aquifer in the immediate vicinity.

5. The regional bedrock dip in the Weldon Spring area is less than one degree to the northeast. Fishel and Williams calculated the dip to average 60 feet per mile, less than one degree, to the northeast. The observed major joint trends of N30°E and N70°E with a trend of N10°E to N45°E observed where the alignment of Dardenne Creek tributaries appeared to be joint controlled. Solution enlarged joints are reported to be numerous in the Burlington-Keokuk and have also been observed by the writer in the Kimmswick Limestone.

6. Groundwater is obtained from the alluvium of the Missouri River valley and lesser quantities from the Dardenne Creek alluvium. Small quantities are probably available on the uplands at the overburden-bedrock contact. Most of the goundwater on the upland areas comes from bedrock aquifers. There are reported to be three horizons which yield potable water. The uppermost is the solutioned upper portion of the Burlington-Keokuk which yields small to moderate quantities of hard water. The next horizon is the Bushberg Sandstone of the Sulfur Springs Formation, at a depth of approximately 300 feet below the floor of the pits. This water is reported to be high in nitrates and chlorides. The lower-most aquifers is the St. Peter Sandstone. The St. Peter yields water of higher quantity and quality than the other bedrock aquifers and is reported to be under artesian pressure. The depth is approximately 700 feet below the pits. The groundwater map in the Fishel and Williams report indicates the groundwater flow in the vicinity of the reservation to be toward the northeast and generally follows the dip of the bedrock. The groundwater flow at the base of the overburden would probably follow the slope of the bedrock with losses into the underlying solutioned bedrock.

7. Loessial soils such as occur at Weldon Spring often possess moderate vertical permeability in the virgin state, however, when modified by the weathering process or reworked, it is relatively impervious. The residual soils underlying the loess is generally impervious except for the lower few feet which is reported to be very gravelly. The report prepared by the St. Louis District of the Corps of Engineers states that the residual clays are traversed with shrinkage cracks and that they give evidence of leaching and prolonged weathering. If the excavations had been carried down into the residuum, a moderate amount of seepage could be expected through it. The soils are generally non-swelling, therefore, saturation could not be depended on to cause them to close.

8. Visual inspection of the pits, verbal information from personnel at the installation, and reports and test data on the soils indicate the pits are essentially impervious. Apparently the virgin loess structure has broken down and the lower portions have been enriched with clay minerals to an unknown degree by leaching from the upper horizons. Most of the loess exposed in cuts surrounding the reservatoin appeared to be modified and did not display the characteristics of the virgin material. We could see no indications in the pit excavations that the loess had been penetrated and the underlying residual soils encountered. The reports indicate a minimum of 10 feet of relatively impervious soil remains in the bottom of the pits. No surface leaks, swampy or boggy areas, or dead vegetation was observed that might indicate seepage from the pits. Records were not maintained as to the quantity of inflow and outflow of liquid stored in the four retention ponds or pits. Estimates of the quantity leakage from pits No. 1 and 2 indicate that, since the inflow has ceased the source of liquids is limited to water drained from consolidation of the slurry and to rainfall on the area of the pond and the small side slopes of the retention dike less probable loss from surface evaporation, the permeability of the underlying clays is less than the 3' per year quoted in the Task Force Report, June 1, 1967.

9. Although no seepage is apparent, it would be wise to construct observation wells adjacent to the pond in order to be able to test the groundwater supplies and take remedial measures should adverse seepage be indicated. Because of the solutioned condition of the Burlington-Keokuk Formation and the well-jointed nature of the underlying limestone, it is felt that a large percentage of any seepage water would move downward into the bedrock. It is recommended that observation wells be constructed adjacent to the pits, especially the two largest ones, and the water from them sampled and analyzed on a regular basis. The wells would be drilled and cased to varying depths and more wells added if traces of the waste materials were found in the initial wells, or if

drilling and testing of any wells indicate that they were not per etrating a favorable sequence of materials, joints, etc. The locations and depths of the initial wells are shown on the attach ed map.

10. Any contaminants entering the groundwater would be diluted by fresh water flowing in the formation. However, because of the concentration of some contaminants, especially the high percentage of nitrates, the natural dilution by fresh water may not be great enough to reduce concentrations. Existing deep wells drilled into the Kimmswick Limestone or St. Peter Sandstone shoul be sampled periodically. Regular inspections should be made of the pit embankments for leakage and animal burrows and of the surrounding areas, especially natural drains, for seepage or the appearance of new springs.

11. The slopes of the dike or embankment containing pit No. 4 are steeper (avg. 1 on 2) than are ordinarily established for permanent retention of fluids. Some local sloughing of both inward and outward slopes has reportedly occurred and necessitated remedial action. The stability of the embankment of pit No. 4 should therefore be analyzed to determine if it is adequate or if modification is required to assure absolute safety of the embankment. The safety factor against failure should be much higher than for ordinary fresh water ponds of equal size.

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Geologist

/s/LOUIS G. ELSER
Civil Engineer

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DRAWINGS REFERRED TO

Underground Explorations, dwg. no. B-6000-22
Blaw-Knox Company, Chemical Plants Division
Rough Grading Plan, Sheet No. 1, dwg. no. 8500-3
Rough Grading Plan, Sheet No. 2, dwg. no. 8500-11
Subsurface Exploration Plan, dwg. no. 8500-9
Plan of Facilities, Block 502, dwg. no. 8502-1

Mallinckrodt Chemical Works, Uranium Division
Additional Raffinate Pit
Contours, dwg. no. 94-385-001-2
Raffinate Pit Profiles, Sheet No. 1, dwg. no. 94-385-001-3
Raffinate Pit Profiles, Sheet No. 2, dwg. no. 94-385-001-4
Structural Outlet to Process Sewer, dwg. no. 94-385-001-6
12M Cu. Ft. Raffinate Pit
Cross Section thru Embankment, Sheet No. 1, dwg. no. 44-385-005-008
Cross Section thru Embankment, Sheet No. 2, dwg. no. 44-385-005-009

Raffinate Pit Improvement
Topography and Plan, dwg. no. 54-385-008-001
Cross Sections, dwg. no. 54-385-008-002

Henry M. Reitz, Consulting Engineers
12M Cu. Ft. Raffinate Pit
Proposed Test Hole Locations, Fig. 1
Boring Logs, Fig. 2-1
Boring Logs, Fig. 2-2
Boring Logs, Fib. 2-3
Boring Logs, Fig. 2-4
Moisture Density Curves, Fig. 3-1
Moisture Density Curves, Fig. 3-2
Location Plan of Site Improvements, dwg. no. 44-385-005-001
Topography & Plan of Raffinate Pit, dwg. no. 44-385-005-002
Raffinate Pit, Cross Sections 1, dwg. no. 44-385-005-003
Raffinate Pit, Cross Sections 2, dwg. no. 44-385-005-004
Miscellaneous Details, dwg. no. 44-385-005-005

GLOSSARY

Abbreviations/Terms	Definitions
absorbed dose	Radiation energy absorbed per unit mass.
ALARA	As Low As Reasonably Achievable.
alpha particle (α)	A positively charged particle emitted from certain radioactive material. It consists of two protons and two neutrons, hence is identical with the nucleus of the helium atom. It is the least penetrating of the common radiation (α, β, γ), hence is not dangerous unless alpha-emitting substances have entered the body.
aquifer	A water-bearing formation below the surface of the earth; the source of wells. A confined aquifer is overlain by relatively impermeable rock. An unconfined aquifer is one associated with the water table.
background radiation	Naturally occurring low-level radiation to which all life is exposed. Background radiation levels vary from place to place on the earth. Estimates of background levels normally do not include alpha radiation from radon daughters.
beta particle (β)	A particle emmited from some atoms undergoing radioactive decay. A negatively charged beta particle is identical to an electron. A positively charged beta particle is called a positron. Beta radiation can cause skin burns and beta-emitters are harmful if they enter the body.

Ci	Curie (the unit of radioactivity of any nuclide, defined as precisely equal to 3.7×10^{10} disintegrations/second).
D and D	Decontamination and decommissioning.
daughter product	The nuclide remaining after a radioactive decay. A daughter atom may itself be radioactive, producing further daughter products.
dose equivalent	A term used to express the amount of effective radiation when modifying factors have been considered (the numerical product of absorbed dose and quality factor.
erg	The basic unit of work or energy in the centimeter-gram-second system (1 erg is equal to 7.4×10^{8} ft-lb).
exposure	Related to electrical charge produced in air by ionizing radiation per unit mass of air.
exhalation	Emission of radon from earth (usually thought of as coming from a uranium tailings pile, but actually from any location).
FB&DU	Ford, Bacon & Davis Utah Inc.
gamma background	Natural gamma ray activity everywhere present, originating from two sources: (1) cosmic radiation bombarding the earth's atmosphere continually, and (2) terrestrial radiation. Whole body absorbed dose equivalent in the U.S. due to natural gamma background ranges from about 60 to about 125 mrem/yr.

gamma ray (γ)	High energy electromagnetic radiation emitted from the nucleus of a radioactive atom, with specific energies for the atoms of different elements and having high penetrating power.
ground water	Subsurface water in the zone of full saturation which supplies wells and springs.
health effect	Adverse physiological response from tailings (in this report, one health effect is defined as one case of cancer from exposure to radioactivity).
HLW	High Level Waste.
insult	Negative impact on the environment or the health of individuals.
INEL	Idaho National Engineering Laboratory.
isotope	One of two or more atoms with the same atomic numbers (the same chemical element) but with different physical properties.
LLW	Low Level Waste.
LWR	Light Water Reactor.
MAC	Maximum Allowable Concentration.
MACH	Maximum Allowable Concentration for Hulls.
MACM	Maximum Allowable Concentration for specific material.
μR/hr	Microroentgen per hour.
mrem/hr	Millirem (milliroentgen equivalent man).

MeV	Million electron volts.
MPC	Maximum Permissible Concentration (the highest concentration in air or water of a particular radionuclide permissible for occupational or general exposure without taking steps to reduce exposure).
MSCR	Maximum Surface Count Rate
noble gas	One of the gases, such as helium neon, radon, etc., with completely filled electron shells which is therefore chemically inert.
NRC	Nuclear Regulatory Commission.
nuclide	A general term applicable to all atomic forms of the elements; nuclides comprise all the isotopic forms of all the elements. Nuclides are distinguished by their atomic number, atomic mass, and energy state.
ORNL	Oak Ridge National Laboratory.
pCi/ℓ	Picocurie per liter.
QF	Quality factor (an assigned factor which denotes the modification of the effectiveness of a given absorbed dose by the linear energy transfer).
R	Roentgen (a unit of exposure to ionizing radiation. It is that amount of gamma or X-rays required to produce ions carrying 1 electrostatic unit of electrical charge, either positive or negative, in 1 cubic centimeter of dry air under standard conditions, numerically equal to 2.58 x 10^{-4} coulombs/kg).

rad	The basic unit of absorbed dose of ionizing radiation. A dose of 1 rad means the absorption of 100 ergs of radiation energy per gram of absorbing material.
radioactivity	The spontaneous decay of disintegration of an unstable atomic nucleus, usually accompanied by the emission of ionizing radiation.
radioactive decay chain	A succession of nuclides each of which transforms by radioactive disintegration into the next until a stable nuclide results. The first member is called the parent, the intermediate members are called daughters, and the final stable member is called the end product.
radium	A radioactive element, chemically similar to barium, formed as a daughter product of uranium (^{238}U). The most common isotope of radium, ^{226}Ra, has a half-life of 1,620 yr. Radium is present in all uranium-bearing ores. Trace quantities of both uranium and radium are found in all areas, contributing to the gamma background.
radon	A radioactive, chemically inert gas, having a half-life of 3.8 days (^{222}Rn); formed as a daughter product of radium (^{226}Ra).
radon background	Low levels of radon gas found in an area, due to the presence of radium in the soil.
radon concentration	The amount of radon per unit volume. In this assessment, the average value for a 24-hr period of atmospheric radon

	concentrations, determined by collecting data for each 30 min period of a 24-hr day and averaging these values.
radon daughter	One of several short-lived radioactive daughter products of radon (several of the daughters emit alpha particles).
RCF	Reference Containment Facility.
RCG	Recommended Concentration Guide.
RDC	Radon daughter concentration (the concentration in air of short-lived radon daughter, expressed usually in pCi/l; also measured in terms of working level (WL).
radon flux	The quantity of radon emitted from a surface in a unit time per unit area (typical units are in pCi/cm^2-sec).
raffinate	The liquid part remaining after a product has been extracted in a solvent extraction process.
recharge	The processes by which water is absorbed and added to the zone of saturation of an aquifer, either directly into the formation or indirectly by way of another formation.
rem	(Acronym of roentgen equivalent man) The unit of dose of any ionizing radiation which produces the same biological effect as a unit of absorbed dose of ordinary X-rays, numerically equal to the absorbed dose in rads multiplied by the appropriate quality factor for the type of radiation. The rem is the basic recorded unit of accumulated dose to personnel.

residual value	The value of minerals in tailings material.
riprap	A irregular wall of broken rock, placed as a retaining wall, as a protection for dikes, etc.
RSLF	Reference Sanitary Landfill Facility.
RWDCS	Radioactive Waste Disposal Classification System
sands	Relatively coarse-grained materials produces along with the slimes as waste products of ore processing in uranium mills (see tailings). These sands normally contain less radioactive material than the slimes.
slimes	Extremely fine-grained materials, mixed with small amounts of water, produced along with the sands as waste products of ore processing in uranium mills (see tailings). Most of the radioactive material remaining in tailings is found in the slimes.
tailings	The remaining portion of a metal-bearing ore after the metal, such as uranium, has been extracted. Tailings also may contain other minerals or metals not extracted in the process (e.g. radium).
TRU	Transuranic.
WL	Working level. A unit of radon daughter exposure, equal to any combination of short-lived radon daughters in 1 liter of air that will result in the ultimate emission of 1.3×10^5 MeV of potential alpha energy. This level is equivalent to the energy produced in the decay of the

	daughter products RaA, RaB, RaC, and RaC' that are present under equilibrium conditions in a liter of air containing 100 pCi or Rn-222. It does not include decay of RaD (22 yr half-life) and subsequent daughter products.
WLM	Working level month. One WLM is equal to the exposure received from 170 WL-hours.

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