DETERMINATION OF A RADIOACTIVE WASTE CLASSIFICATION SYSTEM

FINAL REPORT

FOR

LAWRENCE LIVERMORE LABORATORY
LIVERMORE, CALIFORNIA

PO. 6320703

SEPTEMBER 1977

MASTER

BY

Ford, Bacon & Davis Utah Inc.

375 CHIPETA WAY, SALT LAKE CITY, UTAH 84108 • 801 - 583-3773

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EXB

PREFACE

This document presents the results of a study undertaken by Ford, Bacon & Davis Utah Inc. (FB&DU) for Lawrence Livermore Laboratory (LL) to develop the methodology for quantifying the interfaces in a radioactive waste classification system.

The results of the study are presented in this document in Sequential Thematic Organization of Publications (STOP) format. In this format, the most important ideas of each topic are presented first in **LARGE BOLD TYPE**. More detailed information is then given in a summary paragraph in **MEDIUM TYPE**. Technical details pertinent to the given topic are provided in the small type text. On the page facing the text a table or figure illustrates the topic of each section. Additional detailed information is provided in a series of appendices.

This format is useful for those with limited time. Those wishing a general overview can read the **LARGE BOLD TYPE** and illustrations to obtain a general idea of the contents of the document. Those with more time can additionally read the **MEDIUM TYPE** summaries. People desiring more details then read the textual material in regular type.

An earlier version of this report was reviewed by a Technical Advisory Panel at a meeting in Las Vegas on July 19 and 20, 1977. Names of participants are given in Appendix F. Comments and suggestions received on the working draft have been incorporated where appropriate into this draft of the report.

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GLOSSARY

ALAP	-- As low as practicable.
ALARA	-- As low as reasonably achievable.
Containment	-- Indicates that waste materials are removed from man's access via administrative controls. Some finite release of contained activity over time may occur.
Isolation	-- Indicates complete removal of wastes from the biosphere. Access by man would require substantial effort.
Innocuous Wastes	-- Wastes of such low radioactive content that their direct release to the environment is considered acceptable (essentially nonradioactive).
Low-Level Wastes	-- Wastes requiring containment to assure that credible exposures to man remain within acceptable limits.
High-Level Wastes	-- Wastes requiring long-term isolation because potential hazards are so high that achievable containment in low-level facilities is not sufficient to preclude unacceptable exposures over long periods of time.
MPC	-- Maximum Permissible Concentrations given in 10 CFR 20 indicate levels below which materials may be directly released to the environment.
RHI	-- Relative Hazard Index based on maintaining constant relative risks for various nuclides using MPC values as indicators of relative risk.
Nuclide	-- Specific atom characterized by number of protons and neutrons in its nucleus.
Curie	-- Measure of radioactive intensity - 3.7×10^{10} disintegrations per second.
Half-Life	-- Measure of rate at which particular radioactive species decays. One half of the material initially present decays in one half-life.
Interface	-- Concentration value bounding or between two adjacent categories.
Disposal	-- To get rid of or throw away waste materials by placement in appropriate waste management facility with no intent of recovery or retrieval.

1 INTRODUCTION

1.1 Background and Purpose

REGULATORY DECISIONS REQUIRE QUANTITATIVE DEFINITION OF A PRACTICAL RADIOACTIVE WASTE CLASSIFICATION SYSTEM.

TO BE PRACTICAL, A WASTE MANAGEMENT SYSTEM MUST MEET THE FOUR CRITERIA LISTED IN FIGURE 1.1 THIS REPORT DESCRIBES A SYSTEMATIC METHODOLOGY FOR DEFINING AND QUANTIFYING THE RADIOACTIVITY CONCENTRATION INTERFACES BETWEEN CATEGORIES OF RADIOACTIVE WASTE.

The Nuclear Regulatory Commission (NRC) is developing regulations governing the management and disposal of radioactive waste. In order to provide a broad basis for formulation of such regulations, the NRC requested the Lawrence Livermore Laboratory (LLL) to develop a radioactive waste classification system which would be applicable to all facilities generating, handling, and disposing of radioactive wastes.

The waste classification system should meet the four basic objectives listed in Figure 1.1. The classification system will not replace good operational criteria nor regulations, but will provide guidelines concerning management decisions for those handling the waste. Performance criteria for waste and waste management facilities are separate considerations from this study to determine the waste classification system.

With the concurrence of the NRC, LLL subcontracted with Science Applications Inc. (SAI) of McLean, Virginia to develop the classification scheme. A technical advisory panel was organized to consider the issues in establishing a practical and acceptable waste classification system. Several alternative systems were analyzed, and are summarized in reference 1. A three category system was selected that is based on the ultimate disposition of the waste material.

The objective of this study is to develop the methodology for defining the quantitative interface values between the three waste categories. To demonstrate its applicability, this methodology is applied and specific radioactivity concentrations obtained for the interface between the two higher activity categories. This study is a general scoping study and the recommendations are tentative pending additional analyses and more complete validation of the major assumptions.

CRITERIA FOR WASTE CLASSIFICATION SYSTEM

- * PROTECT PUBLIC HEALTH AND SAFETY
for present and future generations
- * HAVE A SOUND TECHNICAL BASIS
that is defensible and adequate
- * INCORPORATE GOOD ECONOMIC PRACTICES
to meet ALAP and ALARA guidance
- * BE WORKABLE BY WASTE HANDLERS
to provide guidance on waste disposition

FIGURE 1.1 CRITERIA FOR WASTE CLASSIFICATION SYSTEM

1.2 Statement of the Problem

SELECTION OF QUANTITATIVE VALUES SEPARATING THE THREE WASTE CATEGORIES--INNOCUOUS, LOW-LEVEL AND HIGH-LEVEL--DEFINES THE WASTE CLASSIFICATION SYSTEM PROPOSED TO MEET THE CRITERIA.

THE PROPOSED WASTE CLASSIFICATION SYSTEM, SHOWN IN FIGURE 1.2, IS BASED ON THE ULTIMATE DISPOSITION OF THE WASTES. HIGH-LEVEL WASTES ARE THOSE THAT REQUIRE LONG-TERM ISOLATION, LOW-LEVEL WASTES REQUIRE CONTAINMENT TO CONTROL THEIR POTENTIAL RELEASE TO THE ENVIRONMENT, INNOCUOUS WASTES ARE THOSE THAT CAN BE CLASSED AS ESSENTIALLY NON-RADIOACTIVE AND DISCHARGED DIRECTLY AS REGULAR REFUSE. THE LOW-LEVEL WASTE CATEGORY BOUNDARIES ARE DERIVED BASED ON POTENTIAL PUBLIC HAZARDS FROM WASTE DISPOSAL. HAZARDS ARE DETERMINED USING CONSERVATIVE ENGINEERING JUDGEMENTS CONCERNING ANTICIPATED DISPOSAL CONDITIONS.

The three waste classes shown in Figure 1.2 are defined as follows: high-level wastes are those that require long-term isolation in a waste repository; low-level wastes are acceptable for disposal in an active containment facility; and innocuous wastes by definition are classed as non-radioactive and may be discarded as regular refuse. Categorization 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. The ultimate disposition forms the basis for waste classification.

This classification system is intended to provide radioactive waste generators and handlers guidance only on final disposition of the wastes. It is not intended to replace or alter operational and transportation guidelines, practices, and regulations now in use. It will be assumed that all waste management and transportation operations will be in compliance with appropriate regulations, standards, and health and safety practises. Maintaining exposures at levels as low as reasonably achievable is an integral basis for this proposed classification system. The proposed classification system should not be inconsistent with current radioactive waste management philosophies.

FIGURE 1.2 SCHEMATIC OF ASSUMED RADIOACTIVE WASTE CLASSIFICATION SYSTEM

1.3 Approaches for Determining Interfaces

QUANTITATIVE RADIOACTIVITY CONCENTRATION INTERFACES CAN BE DETERMINED BY CONSIDER- ING RISKS TO "MAXIMUM" INDIVIDUALS OR INTE- GRATED POPULATION RISKS FROM WASTE DISPOSAL.

MAXIMUM PROJECTED DOSES TO NON-OCCUPATIONAL INDIVIDUALS FROM WASTE DISPOSAL MUST BE MAINTAINED WITHIN ESTABLISHED GUIDELINES. DOSES INTEGRATED OVER EXPOSED POPULATIONS ARE COUPLED WITH DISPOSAL COSTS TO DETERMINE COST-EFFECTIVENESS. RISKS ARE ESTIMATED BY USE OF MATHEMATICAL MODELS, BASED ON POSTULATED DISPOSAL CONDITIONS. THE MODELS PREDICT MOVEMENT OF RADIOACTIVITY TO ALLOW ASSESSMENT OF POTENTIAL EXPOSURES AND RESULTANT DOSES TO THE PUBLIC. FOR PURPOSES OF THIS STUDY IT IS ASSUMED THAT COMPLIANCE WITH EXISTING REGULATIONS (10CFR20) IS ACCEPTABLE WASTE MANAGEMENT PRACTICE. HOWEVER, THE RESULTS CAN BE SCALED TO OTHER CRITERIA IF DESIRED. THIS REPORT FOCUSES ON DETERMINATION OF THE MAXIMUM CONCENTRATIONS IN LOW-LEVEL WASTES THAT ASSURE RESULTANT RISKS ARE WITHIN THE GUIDELINES.

The central element in determining the quantitative values of the waste-class interfaces is evaluating the potential consequences from wastes placed in an active containment facility. The low-level waste category interfaces thus describe the entire system. This waste category requires social and resource commitments for an extended post-operational period of time requiring such functions as area security, repair of containment barriers and environmental monitoring. Using parameters representative of a shallow land burial facility, a reference containment facility (RCF) was defined to allow calculation of releases of low-level waste material from disposal operations. The existing shallow land burial facilities currently

in operation in the United States were reviewed to determine key parameters such as release fractions, hydrology, climatology, meteorology, and geology. Although the RCF is based on existing facilities, it is assumed that the performance of future facilities will exceed that of current disposal sites.

To properly assess the risks from waste disposal, both airborne and waterborne release pathways to man were analyzed. Inhalation of particulate activity resulting from eventual reclamation and reuse of the disposal site after institutional controls are relaxed, operational releases, hydrologic transport of buried waste materials, and erosion were evaluated and resultant doses determined. Doses can be compared to guidelines (2) for determining acceptable concentrations in the waste.

APPROACHES TO SOLUTION

- * RISK ASSESSMENT
- * DOSE TO "MAXIMUM" INDIVIDUAL
 1. Eventual Site Reclamation
 2. Operational Releases
 3. Hydrologic Transport Releases
 4. Erosion Releases
- * COST-BENEFIT APPROACH
 1. Cost Estimation
 2. Population Doses
 3. Cost-Benefit Comparisons
- * INNOCUOUS-LLW INTERFACE

FIGURE 1.3 APPROACHES FOR DETERMINING INTERFACE VALUES

2. DESCRIPTION OF LOW-LEVEL WASTE DISPOSAL FACILITIES

2.1 Description of Existing Low-Level Waste Disposal Facilities

EXISTING LOW-LEVEL WASTE DISPOSAL FACILITIES ARE REVIEWED TO DETERMINE REASONABLY CONSERVATIVE VALUES FOR INPUT TO THE CALCULATIONAL MODELS.

THE EXISTING ERDA AND COMMERCIAL LOW-LEVEL WASTE BURIAL FACILITIES WERE REVIEWED TO SELECT CONSERVATIVE VALUES FOR USE IN CALCULATING RELEASES FROM DISPOSAL OPERATIONS AND RESULTANT DOSES. THE LOW-LEVEL WASTE CATEGORY IS THE KEY TO DEFINING THE CLASSIFICATION SYSTEM. EXISTING DISPOSAL FACILITIES PROVIDE PERSPECTIVE CONCERNING LIKELY CONSEQUENCES OF FUTURE DISPOSAL EFFORTS.

Current practice in this country is to dispose of solid low-level waste primarily by shallow land burial. The RCF used in this study to analyze the degree of containment is based on shallow land burial. The parameters selected to mathematically describe the RCF were obtained from measured values of those parameters at existing facilities, and from reasonably conservative estimates of the values to be allowed at any future burial sites.

For perspective on current LLW handling and to provide background for judging the reasonability of the RCF, a review of existing low-level waste burial facilities was performed. (3-9) There are presently six commercial low-level radioactive waste burial facilities and five major active sites for burial of defense and research related radioactive wastes at ERDA facilities in the United States. Although the

sources of the wastes may differ from site to site, the general operational characteristics of the facilities and the compositions of the actual wastes generally are 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. More details concerning the existing sites are summarized in Appendix A.

Location	Commercial Capacity (m ³)	Climate	Nearby Rivers	Cover Depth (m)	Observed Radionuclide Migration
HANFORD, WA		Semi-Arid	10 km to Columbia	2.5	Through Uptake by Deep Rooted Plants
RICHLAND, WA *	9 x 10 ⁵	Semi-Arid	10 km to Columbia	2	Not Observed
BEATTY, NV	7 x 10 ⁵	Arid	3 km to Amargosa	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 x 10 ⁵	Humid	Site Boundary	1	Not Observed
MOREHEAD, KY	3 x 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
BARNSWELL, SC	2 x 10 ⁶	Humid	Site Boundary	3	Not Observed
WEST VALLEY, NY	2 x 10 ⁵	Humid	On-Site	3	On-Site Ground Water, Off-Site Surface Water
*ERDA Sites					

TABLE 2.1 SURVEY OF EXISTING LOW-LEVEL WASTE DISPOSAL FACILITIES

2.2 Characterization and Projections of Solid Low-Level Wastes

SINCE THE PHYSICAL CHARACTERISTICS AND VOLUMES OF LOW-LEVEL WASTES DICTATE MANY RISK AND COST FACTORS IN THE ANALYSIS USED IN THIS STUDY, AVAILABLE INFORMATION IS REVIEWED TO PROVIDE INPUT TO CALCULATIONAL MODELS.

LOW-LEVEL WASTES CURRENTLY BURIED ARE DIVERSE IN NATURE AND SOURCE BUT HAVE GENERAL SIMILARITIES. ESTIMATION OF RELEASES OF BURIED WASTE MATERIALS FROM A REFERENCE SITE THEREFORE CAN BE MADE BASED ON ANALYSES OF PRESENT LOW-LEVEL WASTES. PROJECTIONS OF THE ACTIVITIES AND VOLUMES OF WASTES TO BE GENERATED IN THE FUTURE ARE IMPORTANT IN DETERMINING RELATIVE COSTS AND POTENTIAL RISKS.

Because the release of radionuclides from a reference containment facility depends in part on the physical form of the waste, characteristics of currently buried wastes were reviewed. Low-level radioactive wastes are generated in each stage of the nuclear fuel cycle. This study does not include, however, wastes generated during the front end of the cycle; that is, waste from mining, milling, refining and enriching operations. These large volume, low activity wastes are not generally disposed of by burial, and will be subject to future evaluations. The low-level wastes described in this document include those generated in fuel fabrication, reactor and reprocessing operations, clean-up and decommissioning processes, and non-fuel cycle sources such as hospitals and industrial users of radioactive materials.

Because of the diversity of sources and treatments used in handling the wastes, a detailed description of a typical physical waste form is not possible. Most existing disposal facilities do require that certain minimum specifications prohibiting free liquids, pyrophorics, explosives, and hazardous biological or chemical waste forms be met before wastes are buried. Activity and concentration ratios for most low-level wastes are generally similar to those listed in Table 2.2.1.

Projections of waste generation as shown in Table 2.2.2. allow determination of relative changes in volumes, radioactivity concentrations, and costs depending on the acceptance of different categories for burial.

TABLE 2.2.1
CALCULATED CONCENTRATIONS OF RADIOACTIVE MATERIALS IN
SOLID WASTES GENERATED DURING OPERATION OF
LIGHT WATER REACTORS AND SHIPPED TO COMMERCIAL BURIAL GROUNDS(a)

Nuclide	Half-Life (years)	Concentrations		Total Curies/ GW(e) Yr
		Ci/m ³		(630 M ³)
H-3	12.3	7.06		4449
Co-60	5.3	2.47		1557
Cs-137	30.1	1.76		1112
Cs-124	2.06	1.06		667
Cr-51	0.075	1.06		667
Fe-55	2.7	0.71		445
Mn-54	0.85	0.35		222
Sr-90	29	0.0035		2
Pu-238	87.8	0.0035		2
Pu-239	24400	0.0035		2
Pu-241	15	0.035		2
Np-237	2100000	0.0035		2
Am-241	433	0.0035		2
Cm-242	0.45	0.0035		2
Cm-244	17.9	0.0035		2
Ni-59	8000	0.0014		.9
Tc-99	213000	0.0003		.02
I-129	15900000	0.00001		.004
TOTALS		14.50		9132

(a) From Appendix H, NUREG-0216

TABLE 2.2.2
RELATIVE WASTE CONCENTRATIONS AND
VOLUMES PROJECTED PER GW(e) YR(a)

Waste Categories as Generated	Gross Radioactivity Concentration (Ci/m ³)		Volume Generated Per GW(e) Yr (m ³)
	Upper Limit	Average	
Routine Low-Level	100	20	660
Intermediate Level 1	1,000	350	75
Intermediate Level 2	10,000	3,500	20
High-Level	10,000,000	3,500,000	3

(a) Information from ERDA-76-43

2.3 Description of Reference Containment Facility (RCF)

THE CALCULATIONAL MODELS USED TO EVALUATE RELEASES OF WASTE MATERIALS TO DETERMINE THE CONCENTRATION INTERFACES ARE BASED ON A REFERENCE CONTAINMENT FACILITY (RCF) UTILIZING SHALLOW LAND BURIAL FOR DISPOSAL OF RADIOACTIVE WASTES.

THE REFERENCE CONTAINMENT FACILITY (RCF), SHOWN SCHEMATICALLY IN FIGURE 2.3, ALLOWS CALCULATION OF RELEASES TO THE ENVIRONMENT FROM CONTAINMENT OF WASTES AT DIFFERING CONCENTRATION LEVELS. REPRESENTATIVE PHYSICAL, GEOLOGICAL AND RADIOLOGICAL PARAMETERS, LISTED IN TABLES 2.3.1 AND 2.3.2 WERE SELECTED. THE REFERENCE SITE HAS A CAPACITY OF ABOUT $6 \times 10^5 \text{ M}^3$, OR SUFFICIENT VOLUME TO CONTAIN THE RADIOACTIVE WASTES GENERATED BY AMOUNT 1000 GW(e) YR OF POWER PRODUCTION.

The three category waste classification system can be quantified by determining the bounding radioactivity concentration limits for low-level wastes requiring active containment. The main emphasis of the present study is to determine appropriate radioactivity concentration interfaces between low-level wastes and high-level wastes requiring isolation. To accomplish this objective, a reference containment facility (RCF) was determined to obtain a data base from mathematical models and analytical evaluations, which are based on shallow land burial of wastes containing differing concentration levels. The RCF will conform to criteria and regulations typical of those now in effect, and since data exists for burial facilities presently in use, the RCF is based on

shallow land burial. Should waste management practices and philosophies change in the future, the interface levels should be appropriately reevaluated.

Figure 2.3 is a schematic drawing of the RCF, indicating the general hydrogeological assumptions upon which the analytical model of the facility is based. Tables 2.3.1 and 2.3.2 list the pertinent parameters involved in describing the behavior of the buried wastes in the RCF. Using these parameters and the calculational methods described in Section 3, the appropriate waste classification system concentration interface bounding the low-level waste category can be determined. Appendix B contains more details concerning the assumptions used in establishing the RCF.

FIGURE 2.3 SCHEMATIC OF REFERENCE CONTAINMENT FACILITY

TABLE 2.3.1

REFERENCE CONTAINMENT FACILITY PARAMETERS

Site Plan Area (m^2)	2×10^6
Size of Trenches - Length by Width by Depth (m)	$200 \times 10 \times 8$
Number of Pits	100
Distance to Site Boundary (m)	160
Distance between Pit and Aquifer (m)	10
Water Velocity from Pit to Aquifer (m/yr)	10
Annual Precipitation (m/yr)	0.5 - 2
Aquifer Flow Area (m^2)	1000
Distance from Site to Surface Water (km)	1
Aquifer Water Velocity (m/yr)	111
Dispersion Coefficient (m^2/yr)	0.42
Meteorology - Pasqual Stability Conditions	E and F
Average Windspeed (m/s)	1.6
Bulk Soil and Waste Density (g/cm^3)	2
Reference Dust Loading (mg/m^3)	2

TABLE 2.3.2
NUCLIDE SPECIFIC PARAMETERS AND VALUES

	Reference Inventory (Ci)	Sorption Coefficient	Reference Leach Rate (yr^{-1})
Strontium-90	24000	100	1×10^{-2}
Iodine-129	0.73	1	1×10^{-2}
Cesium-137	35000	1000	1×10^{-2}
Neptunium-237	12000	100	6×10^{-2}
Plutonium-239	65000	10000	6×10^{-4}
Americium-241	65000	1000	6×10^{-4}

3.1 Risk Assessment

THE GUIDELINES FOR ACCEPTABLE RISKS AS USED IN THIS STUDY ARE NON-OCCUPATIONAL DOSES TO A FEW "MAXIMUM" INDIVIDUALS SHOULD NOT EXCEED 0.5 REM/YR.

THE MAXIMUM INDIVIDUAL NON-OCCUPATIONAL DOSE LIMIT OF 0.5 REM/YR HAS BEEN USED IN THIS STUDY TO DETERMINE THE CONCENTRATION INTERFACE BETWEEN LOW-LEVEL AND HIGH-LEVEL WASTES. USING THE METHODOLOGY DEVELOPED, OTHER GUIDELINES COULD BE SELECTED AND THE RESULTANT INTERFACES OBTAINED. BASED ON CALCULATED RELEASES OF LOW-LEVEL WASTE MATERIALS FROM THE RCF, CONCENTRATIONS OF RADIOACTIVITY WERE DETERMINED THAT POTENTIALLY COULD RESULT IN MAXIMUM INDIVIDUAL DOSES OF 0.5 REM/YR. POPULATION DOSES FOR A TYPICAL POPULATION DISTRIBUTION WERE ALSO OBTAINED UNDER POSTULATED CONDITIONS. THE RISKS ASSOCIATED WITH DISPOSAL OF LOW-LEVEL WASTES IN THE RCF WERE EXPRESSED IN RESULTANT INDIVIDUAL DOSES (REM) AND POPULATION DOSES (MANREM). FOR COMPARISON, RISKS FROM DISPOSAL OF LOW-LEVEL TYPE WASTES IN A REPOSITORY OR LONG-TERM ISOLATION FACILITY WERE ASSUMED TO BE ZERO. THE APPROPRIATE CONCENTRATION-TO-DOSE CONVERSION FACTORS ARE 870 REM 50 YEAR LIFETIME DOSE PER μCi OF ^{239}Pu INHALED AND 3×10^{-5} REM 50 YEAR LIFETIME DOSE FROM INGESTION PER Ci OF ^{239}Pu ANNUALLY ENTERING THE RIVER SYSTEM.

The analytical procedures for determining the quantitative values of the waste-class interfaces employ five basic steps: establishing potential exposure events, determining source terms and release fractions from the RCF, modeling the transport of the radioactivity through the environment, estimating the doses from the calculated maximum exposure to individuals and populations and relating these doses to the established guidelines. The radioactivity concentration (activity per unit volume) of the waste material is then adjusted to match the doses to the maximum guideline values. The analytical models for the pathway analysis include a hydrogeological, meteorological and direct inhalation models.

Because ^{239}Pu is one of the more environmentally significant components in the waste, risks were estimated based on its release. The conversion factor relating the dose to the ^{239}Pu concentration inhaled is detailed in Appendix C. Corresponding conversion factors relating the dose to the radioactivity concentrations in water were obtained from another source⁽¹⁰⁾ and are detailed in Appendix D. Risks from a few other significant nuclides were also calculated and compared to the ^{239}Pu base.

It is assumed that high-level waste repositories will provide sufficient containment to prevent doses to individuals from low-level wastes if they were sent there for isolation.

Q_i = Release Function for Potential Exposure Event

D_i = Human Uptake to Dose Conversion Factors for Airborne and Waterborne Concentrations

Figure 3.1 Generalized Risk Assessment Methodology

3.2 Dose To "Maximum" Individual

POTENTIAL RELEASES FROM THE RCF ARE EVALUATED TO DETERMINE THE POTENTIAL DOSE TO A "MAXIMUM" INDIVIDUAL RESULTANT FROM DISPOSAL OF LOW-LEVEL WASTES

BOTH AIRBORNE AND WATERBORNE RELEASES FROM THE RCF WERE INVESTIGATED. STANDARD CALCULATIONAL TECHNIQUES ARE USED TO DETERMINE THE CONCENTRATION OF THE RADIOACTIVITY THAT POTENTIALLY COULD REACH MAN THROUGH THESE ROUTES. THE CONCENTRATION-TO-DOSE CONVERSION FACTORS ARE THEN APPLIED TO ASCERTAIN THE POTENTIAL IMPACTS.

The major potential pathways for the movement of radionuclides from the RCF are shown schematically in Figure 3.2

The three scenarios associated with airborne releases are: reclamation and reuse of the site; small releases during routine operations; and the release from an accidental rupture of a container. Doses determined from the first scenario occur from the direct inhalation of radioactive material. Doses from the latter two occur by the transport of the airborne material to the site boundary. The allowable concentrations in the waste at the time of containment, corrected for the decay that occurs before the exposure to man, are proportional to the dose.

Two scenarios are associated with waterborne releases. First is leaching to an underground

aquifer which ultimately discharges to a surface stream. The second involves the erosion of cover and low-level waste material and resulting surface movement to a nearby river.

For those scenarios involving more than a few isolated individuals, i.e., where downstream populations potentially are large, the acceptable dose guidelines of 500 mr/yr were reduced to 5 mr/yr for compliance with the philosophy that exposures to the public be maintained as low as reasonably achievable. In the reclamation and accident scenarios, only a few individuals at most will be exposed, so the 500 mr/yr value was retained.

FIGURE 3.2 SCHEMATIC OF HUMAN EXPOSURE PATHWAYS CONSIDERED

3.2 Dose To "Maximum" Individual

3.2.1 Air Releases

DUSES FROM AIR RELEASES ARE ESTIMATED FROM THE CALCULATED AMOUNT OF RADIOACTIVITY INHALED BY THE PERSON AT RISK.

THE RECLAMATION AND REUSE OF THE DISPOSAL SITE IS POSTULATED TO OCCUR IN THE POST-OPERATIONAL PERIOD AFTER INSTITUTIONAL CONTROLS AT THE RCF HAVE BEEN RELAXED. CONTAMINATED DUST FROM EXCAVATIONS BECOMES AIRBORNE AND IS INHALED BY WORKERS IN THE AREA. SMALL RELEASES OF WASTE MATERIALS COULD OCCUR DURING ROUTINE OPERATIONS FROM HANDLING WASTE CONTAINERS AND FROM ACCIDENTS. RESULTANT AIRBORNE RELEASES MAY EXPOSE NON-OCCUPATIONAL INDIVIDUALS AT THE SITE BOUNDARY.

In the reclamation scenario, it is postulated that a house or other structure may be built on the site, where a few construction workers unknowingly dig into the buried wastes for foundations and footings. Some of the residual activity may become airborne, resulting in an inhalation dose to the individuals involved. Using the acceptable dose guidelines and reasonably conservative exposure assumptions, a suitable interface concentration can be determined.

For routine operations, it is expected that small numbers of waste containers will fail during handling, and that small amounts of contamination may become airborne. This release of small amounts of material was assumed to be continuous and occur during average meteorological conditions. The accidental bursting of single

containers was assumed to occur in the most restrictive meteorological conditions. The dose to a hypothetical individual at the site boundary (160 m from the source) results from breathing the contaminated air. The concentration of activity in the air is proportional to the concentration in the waste. By allowing that hypothetical "fence post" individual to receive no more than the maximum acceptable guideline dose, the allowable concentrations of radioactivity in the waste can be calculated. Appendix C contains details of the airborne pathways calculations.

For airborne releases, the inhalation dose is used because doses from ingestion of activity resulting from airborne releases are insignificant in comparison.

Figure 3.2.1 Airborne Releases

3.2.2 Water Releases

DOSES FROM WATER RELEASES ARE ESTIMATED FROM CALCULATED AMOUNT OF RADIOACTIVITY INGESTED BY A "MAXIMUM" DOWNSTREAM INDIVIDUAL.

GROUND WATER CAN LEACH AND TRANSPORT BURIED RADIO-NUCLIDES TO WATER USE SYSTEMS, RESULTING IN POTENTIAL DOSES TO DOWNSTREAM INDIVIDUALS. SHEET EROSION BY PRECIPITATION AND RUNOFF POSSIBLY COULD MOVE BURIED MATERIALS TO WATER USE SYSTEMS WITH RESULTANT DOSES TO DOWNSTREAM INDIVIDUALS. THESE POTENTIAL ROUTES FOR MOVEMENT FROM THE RCF TO MAN ARE EVALUATED.

Migration of radionuclides through the geosphere from their burial site in water represents an exposure pathway to the environment which is considered:

The hydrology pathway is composed of ground water percolating through the wastes, with the leachate moving down to an underground flowing aquifer which ultimately discharges to a surface river. All subsequent pathways to the population are considered in determining radiation doses to individuals and populations. (10)

Leach rates used in the analysis are representative of values deduced from data obtained at the INEL low-level disposal sites. (11) Sensitivity analyses were performed to determine the effects of uncertainties and variations in the initial values used for the parameters influencing the radioactivity migration.

Nuclide migration is determined from a mass balance equation which considers longitudinal dispersion, convection, sorption and radioactive decay.

Although future containment facilities are not likely to be located in areas of substantial erosion, this occurrence was investigated using a postulated erosion rate of 6 tons of soil per acre per year. It was further postulated that all of this material reaches the river. Appendix D contains details concerning the water transport of radioactivity.

Figure 3.2.2 Water Releases

3. CALCULATIONAL METHODS FOR DETERMINING INTERFACE LEVELS

3.3 Cost-Benefit Approach

INCREMENTAL COSTS FOR LONG-TERM ISOLATION VERSUS RCF DISPOSAL ARE COMPARED TO CHANGES IN RESULTANT DOSES FOR COST-EFFECTIVENESS EVALUATION.

USING PROJECTIONS OF ANTICIPATED WASTE CONCENTRATIONS AND VOLUMES, INCREMENTAL COST DIFFERENTIALS FOR LONG-TERM ISOLATION VERSUS RCF DISPOSAL FOR VARIOUS WASTE CATEGORIES ARE ESTIMATED. CHANGES IN DOSE FROM DISPOSAL OF WASTES CONTAINING HIGHER CONCENTRATIONS OF RADIOACTIVITY CORRESPONDING TO THE VARIOUS PROJECTED WASTE CATEGORIES ARE CALCULATED BASED ON RELEASES FROM THE RCF. THE COST AND DOSE DIFFERENTIALS CAN BE COMPARED TO COST-EFFECTIVENESS GUIDELINES FROM 10CFR50 APPENDIX I.

The differential (marginal) doses calculated for changes in the upper concentration interface can be combined with the marginal costs to provide a marginal cost-to-dose comparison. Appendix I of 10CFR Part 50 gives a value of \$1,000/manrem for guidance as to the cost effectiveness of dose-reducing measures for light water reactor plants, but it may be used for comparison. A value of this type might be used for determining the maximum allowable concentrations in waste to be buried.

The cost factors described in Appendix B average about \$100/m³ (\$3/ft³) for RCF disposal. For repository isolation, no detailed cost estimates are available at this time. However, values from \$1000/m³ (\$30/ft³) have been projected. An average value of \$3500/m³ (\$100/ft³) has been assumed for comparison purposes in this study.

Since all non-innocuous radioactive waste will be either contained or sent to a repository, increasing the interface concentration value increases the volume that can be stored in the RCF and decreases the volume required to be sent to the repository. This results in a net savings of \$3400 for each additional m³ that is contained in the RCF instead of being placed in the repository. By this approach, a differential cost versus volume is obtained for the radioactive waste.

To determine the doses, increased risks resulting from releases of the material contained in the RCF are determined. The doses to "downstream" individuals and populations are calculated. The doses to off-site individuals and populations are assumed to be zero for the repository case which maximizes the differences in doses, minimizing the differences in costs. Thus, burial of wastes represents an increase in risk over sending the wastes to a repository which can be evaluated on a cost-effectiveness basis.

Figure 3.3 Schematic Drawing Illustrating Cost-Benefit Approach

3.4 Innocuous-Low-Level Waste Interface

A PRELIMINARY ESTIMATE OF THE INNOCUOUS-LOW-LEVEL WASTE INTERFACE CAN BE BASED ON RELEASES RESULTING IN CONCENTRATIONS BELOW MPC LEVELS.

THE USE OF MPC LEVEL RELEASES AS AN INITIAL QUANTIFICATION OF THE INNOCUOUS-LOW-LEVEL WASTE INTERFACE IS A SIMPLIFICATION, BUT IS USED FOR DEMONSTRATION PURPOSES. ADDITIONAL EFFORT IS NEEDED TO ESTABLISH THIS INTERFACE.

Further study is needed before definition of this lower interface level is firm. Calculations show that for equal concentrations in the waste, airborne pathways are more restrictive than water pathways to human exposure; therefore, we have used MPC levels in air for estimating an acceptable interface.

If the radioactivity per unit volume in solid wastes is less than the cumulative MPC's in air for the nuclides contributing to the waste activity, it is assumed that the waste can be discharged to the environment without inducing any unacceptable exposure.

This approach presumes that biological concentration factors and consumption probabilities in the human uptake pathways from discharged solid wastes are small enough to be ignored, and that human consumption of MPC levels of activity is acceptable, although unlikely from this source. Much further study is required to adequately define these factors.

Figure 3.4 Innocuous-Low-Level Waste Concentration Interface

4 RESULTS OF THE EVALUATIONS

4.1 Dose to "Maximum" Individual

4.1.1 Reclamation Scenario

THE EVENTUAL RECLAMATION AND REUSE OF THE DISPOSAL SITE AFTER INSTITUTIONAL CONTROLS HAVE BEEN RELAXED DICTATES THE MOST RESTRICTIVE INTERFACE CONCENTRATIONS BETWEEN LOW-LEVEL AND HIGH-LEVEL WASTES CORRESPONDING TO 239Pu CONCENTRATIONS IN THE WASTE OF $1\mu\text{Ci}$ PER CC.

IT IS POSTULATED THAT AFTER A FEW HUNDRED YEARS, INSTITUTIONAL CONTROLS AND RESTRICTIONS ON USE OF THE RCF SITE WILL BE RELAXED. REUSE OF THE SITE WILL INVOLVE EXCAVATIONS FOR FOOTINGS OR FOUNDATIONS. WORKERS IN THE AREA MAY BE EXPOSED TO CONTAMINATED DUST FROM UNCOVERED WASTES. THE POSTULATED INHALATION OF 0.4 GRAM OF DUST LEADS TO AN UPPER CONCENTRATION LIMIT IN THE WASTE OF $1\mu\text{Ci}/\text{CC}$ FOR THE WORKER TO OBTAIN A DOSE OF 0.5 REM/YR.

After an extended period of time, taken to be a few hundred years, when institutional controls and restrictions on use of the RCF site have been relaxed, people could be exposed to radioactively contaminated dust while engaging in earth moving activities at the site. It is postulated that someone may eventually build a house or other structure on the site, and dig unintentionally and unknowingly into the waste for foundations or footings. Some of the residual activity will become entrained in the air and inhaled as dust, resulting in a dose to the individual.

It is assumed for those calculations that the working atmosphere has $2\text{ mg}/\text{m}^3$ of dust. The workers respiration rate is 20 l/min. The worker spends an average of one work month (170 hr) in excavation activities at the site, and does not reside there. He therefore, inhales 0.4 g of dust.

Using a guideline value of 0.5 rem/yr as the maximum dose to the individual's critical organ (the bone for 239Pu), and a dose conversion factor of 870 rem to the bone 50 year dose commitment per $1\mu\text{Ci}$ of 239Pu inhaled, gives a maximum concentration in the dust of 0.06 $\mu\text{Ci}/\text{g}$ or 0.06 $\mu\text{Ci}/\text{cc}$ (assumed density of waste that becomes airborne is $1\text{g}/\text{cc}$).

Assuming a factor of 10 to account for the difference between the upper maximum concentration limit and the average concentration actually in the wastes, and a factor of 3 for dilution of the wastes with soil gives a concentration limit of $1\mu\text{Ci}/\text{cc}$ of 239Pu in the waste.

Using a similar procedure for a few other significant nuclides gives the results shown in Table 4.1.1. Details of these calculations are provided in Appendix C.

Figure 4.1.1 Schematic of Reclamation Scenario

Table 4.1.1
Summary of Interface Concentration Values
for Specific Nuclides from Reclamation Scenario

Nuclide	μCi inhaled to obtain 4 Rem 50 yr Dose to lung (a)	Resultant Interface Concentration (b) ($\mu\text{Ci/cc}$)	μCi inhaled to obtain 20 Rem 50 yr Dose to bone (a)	Resultant Interface Concentration (b) ($\mu\text{Ci/cc}$)
Co-60	3.1	230	400	30,000
Sr-90	222	16,700	7.1	540
Cs-137	100	7,500	560	41,700
Pu-239	0.02	1.5	0.02	1.6
Am-241	0.01	0.9	0.02	1.6

(a) These lifetime doses will result in 500mr/yr peak dose rate in the 50 year interval.

(b) Decay corrections for delay before reclamation are not included in these numbers.

4.1.2 Comparison of Results

THE RESULTS OF CALCULATING INTERFACE CONCENTRATIONS BETWEEN LOW-LEVEL AND HIGH-LEVEL WASTES BASED ON THE OTHER EXPOSURES PATHWAYS ARE COMPARED TO THAT OBTAINED FROM THE RECLAMATION SCENARIO.

AS SEEN IN FIGURE 4.1.2, THE AIRBORNE RELEASES ARE MORE RESTRICTIVE THAN WATERBORNE RELEASES. THE EVENTUAL RECLAMATION SCENARIO IS MORE RESTRICTIVE, SETTING THE CONCENTRATION INTERFACE AT $1 \mu\text{Ci}$ OF 239 PU/CC OF WASTE. THE CONCENTRATION LIMIT IS EXPRESSED PER UNIT VOLUME OF THE WASTE, THE SELECTED UNITS ARE APPROPRIATE FOR COMPARISON WITH MPC VALUES FROM 10CFR20,

Calculating concentrations in the waste that result in maximum individual non-occupational doses gives the relationships between dose and concentration shown in Figure 4.1.2. Doses are higher for the eventual reclamation scenario for wastes at a given concentration than for the other postulated releases.

From the curves in Figure 4.1.2, a concentration interface of $1 \mu\text{Ci/cc}$ of 239 Pu in the waste potentially would result

in 0.5 rem/yr dose to an individual worker for the reclamation scenario. Selection of other values for the guidelines for acceptable risk would give different interface concentrations as shown in the figure.

Choosing to ignore the reclamation scenario would result in the interface being set by the next most restrictive case, accidental releases, at $60 \mu\text{Ci/cc}$.

Details of the calculational results are given in Appendix C for airborne release cases, and Appendix D for waterborne releases.

Figure 4.1.2 Dose To "Maximum" Individual Versus Concentration Interface Values

4.2 Relative Hazard Index Format

THE CONCENTRATION INTERFACES ADJUSTED FOR ANTICIPATED RELEASE CONDITIONS CAN BE NORMALIZED USING THE 10CFR 20 MPC VALUES, TO A RELATIVE HAZARD INDEX (RHI) TO PRESENT THE RESULTS IN A CONCISE FORMAT INDICATIVE OF THE RELATIVE HAZARDS ASSOCIATED WITH DIFFERENT NUCLIDES AT GIVEN CONCENTRATIONS.

BASING THE WASTE CLASSIFICATION SYSTEM ON A RELATIVE HAZARD INDEX (RHI) FORMAT ALLOWS PRESENTATION OF THE INTERFACE CONCENTRATIONS BY USE OF A SINGLE VALUE REPRESENTING THE ACCEPTANCE RISK FOR THAT INTERFACE. THIS FORMAT ALLOWS DETERMINATION OF CONCENTRATION INTERFACES FOR ANY NUCLIDE GIVEN ITS APPROPRIATE MPC AND HALF-LIFE.

To apply the results of this analysis to nuclides other than ^{239}Pu , the MPC for air for the nuclide in question is used as an indicator of relative risk. The interface concentration of another nuclide is determined by normalizing the ^{239}Pu concentration by the ^{239}Pu MPC and multiplying by the MPC for air for that nuclide. Allowance can be made also for differences in half-life based on the radioactive decay that occurs between the time of disposal and the time of the postulated release. Comparison of the results using nuclides other than ^{239}Pu are consistent with the values obtained by this approach.

This use of the MPC values leads to the following definition of a relative hazard index (RHI). $\text{RHI} = \text{DC}/\rho \text{ MPC}_{\text{air}}$, where C is the concentration of the nuclide of interest, MPC_{air} is the maximum

permissible concentration for the given nuclide from 10 CFR Part 20 Appendix B Table II Column 1, ρ is the density of the waste (2g/cc) and D is the reference dust loading factor (2mg/m³). The D/ ρ factor is unitless, and equal to 10⁻⁹.

The RHI for ^{239}Pu , based on an MCP in air for ^{239}Pu of 6×10^{-14} $\mu\text{Ci}/\text{ml}$ is 1.7×10^4 for an initial concentration of ^{239}Pu in the waste of 1 $\mu\text{Ci}/\text{cc}$, with an average waste density of 1g/cc.

The use of this RHI type approach allows description of a constant risk on the basis of a single number, the RHI. The effect of half-life can be accounted for by a single line (or equation) showing the decay allowance. For example, if $(\text{RHI})_0$ is the initial constant risk designator, then $\text{RHI} = (\text{RHI})_0 \exp(-\lambda T)$ is the appropriate value for the time of disposal to give $(\text{RHI})_0$ risk for a nuclide with a decay constant λ and a delay time of T between disposal and loss of control.

TABLE 4.2

SINGLE RADIONUCLIDE LOW-LEVEL/HIGH-LEVEL INTERFACE CONCENTRATION VALUES
(ADJUSTED FOR 100 YEAR DECAY)

Isotope	$T_{1/2}(y)$	10CFR20 MPC _A ($\mu\text{Ci}/\text{ml}$)	Interface Concentration ($\mu\text{Ci}/\text{cc}$)
H-3	12.33	2×10^{-7} (S,I)	2×10^2 b
Cr-51	0.075	8×10^{-8}	8×10^7 b
Mn-54	0.86	1×10^{-9} (S,I)	1×10^6 b
Fe-55	2.7	3×10^{-8} (I)	3×10^7 b
Co-60	5.27	3×10^{-10} (I)	3×10^5 b
Ni-59	3×10^4	2×10^{-8} (S)	3.4×10^5
Ni-63	100	2×10^{-9} (S)	6.7×10^4
Sr-90	29	3×10^{-11} (S)	5.6×10^3
Y-90	0.007	3×10^{-9} (I)	3×10^6 b
Tc-99	213	2×10^{-9}	4.6×10^4
Ru-106	1.010	2×10^{-10} (I)	2×10^5 b
I-129	1.59×10^7	2×10^{-11}	330
Cs-134	2.06	4×10^{-11}	4×10^5 b
Cs-137	30.1	5×10^{-10} (I)	8.5×10^4
Th-232	1.4×10^{10}	1×10^{-12} (S,I)	17
U-235	7.04×10^8	4×10^{-12} (I)	68
U-238	4.5×10^9	3×10^{-12} (S)	50
Np-237	2.14×10^6	1×10^{-13} (S)	1.7
Pu-238	87.8	7×10^{-14} (S)	2.6
Pu-239 ^a	2.439×10^4	6×10^{-14} (S)	1.0
Pu-241	15	3×10^{-12}	3000 b
Am-241	433	2×10^{-13} (S)	4.0
Cm-242	0.45	4×10^{-12}	4000 b
Cm-244	17.9	3×10^{-13}	240
Any other single nuclide not listed above with decay mode other than alpha emission or spontaneous fission		1×10^{-10}	1.7×10^3
Any other single nuclide not listed above which decays by alpha emission or spontaneous fission		2×10^{-14}	0.33

a. Basis from which other numbers were derived.

b. Interface Concentration limited by accident scenario.

4.3 Cost-Benefit Analysis Results

THE INCREMENTAL COSTS FOR ISOLATION INSTEAD OF RCF DISPOSAL DIVIDED BY THE RESULTANT REDUCTION IN POPULATION DOSES DOES NOT GIVE A REASONABLY LIMITING CONCENTRATION INTERFACE.

BASED ON PROJECTIONS OF WASTE CONCENTRATIONS AND VOLUMES, USING RELEASES FROM THE RCF TO DETERMINE POPULATION DOSES, INCREMENTAL COST AND DOSE COMPARISONS GIVE VALUES RANGING FROM $\$10^8$ TO $\$10^5$ PER MANREM FOR AVOIDANCE OF DOSES FROM WASTE DISPOSAL.

Figure 4.3 is a series of graphs showing the results of the cost-benefit analysis performed to evaluate the cost effectiveness of active containment disposal versus long-term repository isolation.

Given projected waste concentrations and volumes and marginal costs for disposal by containment versus isolation at a repository, the cost savings can be determined for including additional categories of waste in the low-level category. By increasing the acceptable interface concentration, additional volumes of waste will be acceptable for RCF disposal. For instance, the first projected category of waste higher in concentration than routine low-level waste would contain 1000 Ci of gross activity per m^3 . The production of this level of waste is estimated to be about $75 m^3/GW(e)$ yr of nuclear power. For a marginal cost of $\$3400/m^3$, $\$255,000/GW(e)$ yr can be saved by containing this category of

waste as opposed to sending it to the repository. Alternatively, it costs $\$3400/m^3$ more to send it to the repository; however, for this increased cost, there will be a decrease in potential risk from this waste volume.

To obtain cost-benefit comparisons in the range of $\$1000/\text{manrem}$ requires average concentrations in the range of $60,000 \text{ Ci}/m^3$ for the higher categories of waste. This concentration is much higher than would result if even high level waste were buried. It is apparent that the cost benefit approach does not yield a limiting upper concentration interface. Although this approach is not as restrictive an indicator of the relative hazards as the maximum individual dose approaches are, it does provide useful insight and perspective on the waste management question. This approach does give an indication of the costs for waste disposal that are expended to protect a potential intruder into the buried wastes.

PROPOSED LOW/HIGH LEVEL WASTE
CONCENTRATION INTERFACE

Figure 4.3 Cost benefit Analysis Results

USING THE MOST RESTRICTIVE CONCENTRATION INTERFACES FROM THESE ANALYSES IN THE RELATIVE HAZARD INDEX FORMAT, THE INTERFACES OF THE WASTE CLASSIFICATION SYSTEM ARE QUANTIFIED.

THE MOST RESTRICTIVE CONCENTRATION INTERFACE OF $1\mu\text{Ci}$ OF ^{239}Pu PER ML OF WASTE, OR A RHI OF 1.7×10^4 , RESULTED FROM THE RECLAMATION SCENARIO. FIGURE 5 IS A GRAPHICAL PRESENTATION OF THE WASTE CLASSIFICATION SYSTEM IN THE RELATIVE HAZARD INDEX (RHI) FORMAT. THIS FIGURE ALSO SHOWS HOW THE RADIOACTIVE HALF-LIFE IS CONSIDERED, THE CUMULATIVE RHI FOR MIXTURES OF NUCLIDES IN THE WASTE MUST NOT EXCEED THE ESTABLISHED RHI LIMIT IF THE MIXTURE IS TO BE CONSIDERED LOW-LEVEL WASTE.

Based on the review of the various approaches from Section 4, the most restrictive interface concentrations between low- and high-level wastes were used to quantify the waste classification system. The upper interface concentrations are appropriately increased to account for decay depending on the delay between cessation of burial operations and reclamation of the land for construction. Assumed delay intervals of 100, 200 and 300 years are shown in the figure. For half-lives greater than a few hundred years, no credit for radioactive decay is taken in determining the longevity of the hazard.

The most restrictive scenario, that of eventual reclamation of the land, gives the basic upper interface. However, the accident scenario becomes most restrictive for very short half-lived materials because the accident is postulated before any significant decay occurs.

Releases of ^{239}Pu have been used to establish the interface values because, for the ratios of nuclides expected in low-level wastes, ^{239}Pu is the most significant contributor to off-site doses. Interface levels for other nuclides have been normalized to that of ^{239}Pu by use of the appropriate MPC values. Use of the MPC's assumes that they were uniformly determined and can be used as relative indicators of risk for the nuclides involved.

For mixtures of nuclides in the waste, the relative hazard indices can be weighted as indicated in 10 CFR Part 20. That is, the cumulative hazard index can be computed by calculating the sum of the concentration-to-MPC ratios of the individual nuclides multiplied by the dust loading correction. Appendix E gives details and examples of the use of this system.

Further study is required to provide a precise and defensible basis for the lower concentration interface.

FIGURE 5

THE METHODS OF 10CFR20 CAN BE APPLIED FOR MIXTURES OF NUCLIDES IN WASTES TO DETERMINE IN WHICH WASTE CATEGORY THE MATERIAL BELONGS.

TYPICAL WASTES ARE COMPOSED OF MIXTURES OF NUCLIDES, THE CONCENTRATIONS OF WHICH ARE NOT ALL KNOWN WITH EQUAL PRECISION. THE WASTE CATEGORY IS OBTAINED BY SUMMING THE RATIOS OF THE NUCLIDE CONCENTRATIONS WITH THE INTERFACE CONCENTRATION VALUES. IF THE SUM OF THESE RATIOS IS ONE OR LESS, THE WASTE MATERIAL IS CLASSIFIED INTO THE LOWER CATEGORY. AS SHOWN IN 10CFR20, APPENDIX B, IF ALL CONCENTRATIONS ARE NOT KNOWN, APPROPRIATE VALUES CAN BE USED TO DETERMINE THE CLASSIFICATION CATEGORY.

If the identity and concentration of each radionuclide in a mixture is known, the waste category is determined by calculating the ratio between the quantity present and the interface concentration value from Table 4.2 for each nuclide. If the sum of the ratios for all nuclides in the mixture is one or less, the waste is in the lower category.

If the identity or concentration of any radionuclide in the mixture is not known, the interface concentration value shall be taken to be 0.33 $\mu\text{Ci/cc}$, unless any of the conditions specified below are met:

a. If the identity of each radionuclide in the mixture is known but the concentration of one or more of the radionuclides in the mixture is not known, the interface concentration value for the mixture is the lowest value for any nuclide in the mixture from Table 4.2; or

b. If the identity of each radionuclide is not known but it is known that certain radionuclides are not present in the mixture, the interface concentration value for the mixture is the lowest value from Table 4.2 for any radionuclide which is not known to be absent from the mixture; or

c. If it is known that alpha-emitters and Sr-90, I-129, Pb-210, Ac-227, Ra-228, Pa-230, Pa-241 and Bk-249 are not present, the interface concentration value is 1700 $\mu\text{Ci/cc}$; or,

d. If it is known that alpha-emitters and Pb-210, Ac-227, Ra-228, and Pu-241 are not present, the interface concentration value is 170 $\mu\text{Ci/cc}$; or,

e. If it is known that alpha-emitters and Ac-227 are not present, the interface concentration value is 17 $\mu\text{Ci/cc}$; or

f. If it is known that Ac-227, Th-230, Pa-231, Pu-238, Pu-239, Pu-240, Pu-242, Pu-244, Cm-248, Cf-249, and Cf-251 are not present, the interface concentration value is 1.7 $\mu\text{Ci/cc}$.

For purposes of this report, a radionuclide may be considered as not present in a mixture if the ratio of the concentration of that nuclide to its interface concentration value from Table 4.2 does not exceed 1/10 and if the sum of all such ratios for all the radionuclides considered as not present in the mixture does not exceed $\frac{1}{2}$.

TABLE 6

EXAMPLES OF USING THE WASTE CLASSIFICATION SYSTEM

(1) Mixture	(2) Radionuclide in Mixture	(3) HLW-LLW Interface Concentration Value from Table 4.2 (uCi/cc)	(4) Hypothetical Concentration in Waste Mixture (uCi/cc)	(5) Ratio Column (4) Column (3)	(6) Category
1	Co-60	3.0x10 ⁵	145	4.8x10 ⁻⁴	
1	Sr-90	5.6x10 ³	325	5.8x10 ⁻²	
1	Cs-137	8.5x10 ⁴	650	7.6x10 ⁻³	
1	Ni-59	3.4x10 ⁵	10	2.9x10 ⁻⁵	
1	Pu-239	1.0	0.01	1.0x10 ⁻²	
1	Mixture Total			7.6x10 ⁻²	Low-Level
2	Unidentified β-γ Waste Mixture	17 (a)	20	1.2	
2	Mixture Total			1.2	High-Level
3	Unidentified β-γ nuclide	1700	500	.29	
3	Unidentified α nuclide	0.17	.01	.06	
3	Mixture Total			.35	Low-Level

(a) Category e in text on mixtures

THE METHODOLOGY DEVELOPED FOR QUANTIFYING THE WASTE CLASSIFICATION SYSTEM INTERFACES DOES PROVIDE AN ACCEPTABLE MEANS FOR DETERMINING RADIOACTIVE WASTE CATEGORIES.

THE THREE CATEGORY WASTE CLASSIFICATION SYSTEM MEETS THE STATED CRITERIA AND OBJECTIVES. THE LOW-LEVEL WASTE/HIGH-LEVEL WASTE INTERFACE HAS BEEN CONSERVATIVELY DETERMINED. THE INNOCUOUS-LOW-LEVEL WASTE CONCENTRATION INTERFACE REQUIRES MORE INVESTIGATION. COST-EFFECTIVENESS DOES NOT ESTABLISH REASONABLE CONCENTRATION INTERFACES. DESCRIBING THE CONCENTRATION INTERFACES ON A PER UNIT VOLUME BASIS FACILITATES COMPARISONS WITH MPC VALUES. BECAUSE THE MOST RESTRICTIVE INTERFACES ARE DERIVED FROM SCENARIOS INVOLVING SMALL VOLUMES OF WASTE, THE ACTUAL CONCENTRATIONS IN THE WASTE ARE MORE LIMITING THAN ARE TOTAL FACILITY INVENTORIES FROM THE PUBLIC HEALTH AND SAFETY STANDPOINT. TOTAL INVENTORY LIMITS MAY BE APPROPRIATE FOR SPECIFIC SITES, BUT ARE NOT LIMITING IN SETTING THE WASTE CLASSIFICATION SYSTEM INTERFACES. A TECHNICAL ADVISORY PANEL (MEMBERS LISTED IN APPENDIX F) HAS REVIEWED THIS REPORT. CONCLUSIONS FROM THAT REVIEW HAVE BEEN INCORPORATED.

A relative hazard index format has been described which concisely defines the classification system interfaces, and provides a conceptually complete system.

Additional efforts may be required to more thoroughly document and justify the low-level waste/high-level waste interface, considering ingrowth of radioactive daughter products and refining calculational parameters.

REFERENCES

1. W. C. King and J. J. Cohen, "Interim Report to the NRC on Radioactive Waste Classification", UCID-17497, published by Lawrence Livermore Laboratory, April 18, 1977.
2. Title 10, Code of Federal Regulations, Part 20.
3. "Alternatives for Managing Wastes from Reactors and Post Fission Operations in the LWR Fuel Cycle", ERDA-76-43, published by United States Energy Research & Development Administration.
4. "Management of Wastes from the LWR Fuel Cycle", CONF 76-0701, proceedings of International Symposium in Denver, Colorado, July 11-16, 1976.
5. "The Shallow Land Burial of Low-Level Radioactivity Contaminated Solid Waste", Panel on Land Burial, Committee on Radioactive Waste Management, Commission on Natural Resources, National Research Council, National Academy of Sciences, 1976.
6. "NRC Task Force Report on Review of the Federal/State Program for Regulation of Commercial Low-Level Radioactive Waste Burial Grounds", NUREG-0217, U. S. Nuclear Regulatory Commission, March 1977.
7. M. F. O'Connell and W. F. Holcomb, "A Summary of Low-Level Radioactive Wastes Buried at Commercial Sites Between 1962-1973, with Projections to the Year 2000", Radiation and Data Reports, December 1974.
8. "Final Environmental Statement on Waste Management Operations at the INEL", ERDA-1536, United States Energy Research & Development Administration, May 1977.
9. "Final Environmental Statement on Waste Management Operations at the Hanford Reservation", ERDA-1538, United States Energy Research & Development Administration, 1977.
10. The Analytic Sciences Corporation, private communication, May 1977.
11. B. L. Schmalz, "Radionuclide Distribution in Soil Mantle of Lithosphere as a Consequence of Waste Disposal at the NRTS", IDO-10049, USAEC, October 1972.

APPENDIX A

REVIEW OF EXISTING LOW-LEVEL
WASTE DISPOSAL FACILITIES

Current practice in this country is to dispose of solid low-level waste (LLW) primarily by shallow land burial, thus, the reference containment facility (RCF) used in this study to analyze the degree of confinement given LLW after disposal is based on shallow land burial. The parameters selected to mathematically describe the RCF were obtained from measured values of those parameters at existing facilities, and from reasonably conservative estimates of the values to be allowed at any future burial sites.

For perspective on current LLW handling and to provide background for judging the reasonability of the RCF, a review of existing low-level waste burial facilities was performed. (1-7) There are presently six commercial low-level radioactive waste burial facilities and five major active sites for burial of defense and research related radioactive wastes at ERDA facilities in the United States. Although the sources of the wastes may differ from site to site, the general operational characteristics of the disposal facilities and the compositions of the actual wastes generally are similar.

Basically, low-level wastes received at the disposal facilities 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. Characteristics of the existing sites are summarized in the following tables.

Table A-1 lists the capacity or volume of the commercial sites, the sizes of trenches or pits employed, cover types and procedures, and quantities of waste buried.

Climatological and hydrogeological information available for each site is tabulated in Table A-2. This includes data concerning precipitation and evapotranspiration, geomorphology, seismicity, surface soils and water, subsurface stratigraphy and aquifers and radionuclide migration.

Table A-3 lists demographic information including downstream populations and distances.

These tables have been included to give perspective to the reference containment facility that has been mathematically modeled for systematically determining the classification system interfaces. The parameters selected for the RCF are given at the bottom of each table. The RCF has been used to predict radionuclide migration from a typical site to analytically develop acceptable concentration levels in wastes requiring containment. However, inclusion of these tables and the RCF parameters should not be construed as either acceptance of current practices or criteria for future confinement facilities.

Because the release of radionuclides from a containment facility depends, in part, on the physical form of the waste, some characteristics of currently buried wastes are described in this appendix. Low-level radioactive wastes are generated in each stage of the nuclear fuel cycle. However, this study does not include wastes generated during the front end of the cycle;

that is, waste from mining, milling, refining and enriching operations. These large volume, low activity wastes are generally not buried, and will be the topic of future evaluations. The low-level wastes described in this document include those generated in fuel fabrication, reactor and reprocessing operations, cleanup and decommissioning processes, and non-fuel cycle sources such as hospitals and industrial users of radioactive materials. Also included in this category of waste are dewatered solids and otherwise solidified low-level liquid wastes.

Decontamination and decommissioning of nuclear facilities produce large items of construction and structural materials that have been contaminated and/or activated during usage. These wastes are currently disposed of by burial with little packaging other than plastic sheeting.

In addition to the wastes 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, who 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 isotopes.

Characterization of the radioactivity in wastes at the commercial sites has been attempted in the past largely by use of broad categories designated as Special Nuclear Materials, Source Materials, and By-product Materials for Fissile, fertile, and fission and activation products, respectively. Table A-4 summarizes the data for existing sites using this method of categorization.

Table A-5 lists nuclide specific data from the Idaho National Engineering Laboratory (INEL) for two recent years. The wastes were generated in activities involving all phases of the fuel cycle after irradiation in test reactors. It is expected that the activity and concentration ratios for these wastes are generally typical of low-level wastes. No other nuclide specific data characterizing low level wastes is readily available. This table has been included for perspective only concerning the nuclide content of generally typical wastes.

TABLE A-2

CLIMATOLOGICAL AND HYDROGEOLOGICAL PARAMETERS AT EXISTING SITES

SITE	CLIMATE (2)	PRECIPITATION Range Annual (3) (in.) (mm)	GEOMORPHOLOGY (4)	CLASSIFICATION (5)	TERR. (6) Thick (inches) (cm)	INTERSTITIAL PERMEABILITY TO AIR (7) (cc/cm)	BED ROCK		DEPTH TO AQUIFER TOP (8) (m)	TYPE	NEAREST SURFACE WATER (9, 10)	WATER FLOW PATHS FROM LOCAL AREAS (11)	OBSERVED GROUNDWATER MIGRATION (12)	DOWNSTREAM RIVER FLOW RATE HIGH, ANNUAL MEAN (13) (14)	SEISMIC HAZARD ZONE (15)
							CLASSIFICATION (2)	STRUCTURE							
CHEROKEE, TN	Semi-arid	200 - 816	Columbia Plateau Semi-forest	Clay, Sand and Gravel	250	Variable	Volcanic Basalt	Massive/Flat-lying	100	Gradient	100m Perennial (Columbia River)	Pores in Sand and Gravel	Migration Through Upstate by Deep Rooted Forest	Columbia River McKays Dam 5350	2
RICHMOND, VA	Semi-arid	200 - 816	Columbia Plateau Semi-forest	Clay, Sand, and Gravel	250	Variable	Volcanic Basalt	Massive/Flat-lying	100	Gradient	100m Perennial (Columbia River)	Unsaturation Flow in Sand and Gravel Pores	Not Observed	Columbia River McKays Dam 5350	2
HEATY, VA	Arid	100 - 1,375	Basin & Range Desert	Alluvial Sand and Gravel	2000	0.02-0.1	Sedimentary and Sedimentary	Folded	80	Gradient	30m Ephemeral (Ganges River)	Unsaturation Flow in Sand and Gravel Pores	Not Observed	None	3
TEXAS, TX	Semi-arid	200 - 600	Volcanic Semi-forest	Alluvial Sand and Gravel	6	Moderate	Volcanic Basalt	Massive/Flat-lying	40-120	Gradient	30m Ephemeral (Rio San Antonio)	Pores in Sand and Gravel	Possibly by On Site Ground Water	Snake River Vicksburg, 1000 250	3
LOS ANGELES, CA	Semi-arid	400 - 570	Mountainous Semi-forest	Weathered Tuff	2	Moderate	Volcanic Tuff	Massive/Flat-lying	200	Gradient	100m Ephemeral Site	Bedrock Fractures and Sand and Gravel Pores	On Site Vadose Water Zone	Rio Grande Albuquerque, New Mexico 28	3
CHICAGO, IL	Humid	800 - 40	Glacial	Glacial Drifts: Silt, Sand, and Gravel	20-30	0.001	Shale, Sandstone and Coal	Flat-lying	5-20 100	Vadose Gradient	Site Boundary Perennial	Pore Spaces in Till	Not Observed	Mississippi River St. Louis, Missouri 5000	1
ALBANY, NY	Humid	1,200 - 300	Ridge & Valley Appalachian	Weathered Shale Clay and Sand	3-5	0.02	Shale	Flat-lying	1-3	Vadose	500m Perennial	Shale Fractures	On and Off Site Ground and Surface Water	Licking River Lexington, Ky. 400 Ohio River Louisville, Ky. 225	1
OKLAHOMA, OK	Humid	1,300 - 460	Ridge & Valley Appalachian	Weathered Shale and Till	10	Very Low	Shale	Folded	5	Vadose	On Site Perennial	Shale Fractures and Pores in Till	On Site Ground Water, Off Site Surface Water	Citrus River, Jay Ridge Tenn. Tennessee River, Chattanooga 1000 Mississippi, Memphis, Tenn. 1355	2
SATURNIA RIVER, SC	Humid	1,100 - 0	Coastal Plain	Sand and Clay	10	Very Low	Clay, Sand, and Sandstone	Flat-lying	10 200	Vadose Gradient	On Site Perennial	Pore Spaces in Sand	On Site Ground Water	Savannah River Climax, Georgia 25	2
BARRETT, SC	Humid	1,100 - 0	Coastal Plain	Sand and Clay	10	0.2	Clay, Sand, and Sandstone	Flat-lying	10 200	Vadose Gradient	On Site Perennial (Lower Trib. Run)	Pore Spaces in Sand	Not Observed	Savannah River Climax, Georgia 25	2
WEST VALLEY, NY	Humid	1,000 - 320	Glacial	Glacial drifts: Clay, Silt, and Sand	20-30	0.5	Shale	Flat-lying	1-20	Vadose Gradient Not Observed	On Site Perennial	Shale Fractures	On Site Ground Water, Off Site Surface Water	St. Lawrence River Lake Ontario Outlet 2000	2
Y-1000 WEST IN RCP	Variable	500 - 2000	Coastal Plain	Sand and Clay	10	0.001	Shale	Flat-lying	10	Gradient	100m Perennial	Pore in Soil Unsaturation Aquifer	Not Observed	5350	-

VENA SITE

TABLE A-3 DEMOGRAPHIC DATA FOR EXISTING SITES

SITE	NEAREST POPULATION (5) CENTERS	(8, 9, 10) DOWNSTREAM POPULATION
HANFORD, WA*	5 km (Richland)	1,000,000
RICHLAND, WA	5 km (Richland)	1,000,000
BEATTY, NV	8 km (Beatty) 180 km (Las Vegas)	6,000
INEL, ID*	26 km (Arco) 62 km (Blackfoot)	1,150,000
LOS ALAMOS, NM*	4 km (Los Alamos) 78 km (Albuquerque)	2,250,000
SHEFFIELD, IL	2.5 km (Sheffield) 17 km (Kewanee) 47 km (Peru)	6,000,000
MOREHEAD, KY	2 km (Morehead) 24 km (Olive Hill) 27 km (Owingsville)	5,600,000
OAK RIDGE, TN*	2.3 km (Oak Ridge) 24 km (Knoxville)	4,000,000
SAVANNAH RIVER, SC*	18 km (Jackson) 24 km (Barnwell) 37 km (Aiken)	350,000
BARNWELL, SC	11 km (Barnwell) 13 km (Williston) 33 km (Aiken)	350,000
WEST VALLEY, NY	12 km (Springville) 27 km (Salamanca) 38 km (Olean)	6,000,000
REPRESENTATIVE OR AVERAGE VALUES	5 km	800,000
*ERDA SITE		

TABLE A-4

LOW-LEVEL WASTE CHARACTERIZATION

AVERAGE CONCENTRATION OF LOW-LEVEL WASTE BURIED AT COMMERCIAL SITES

Ci/m³

YEAR	1963	1964	1965	1966	1967	1968	1969	1970	1971	1972	1973	1974	1975	1976	AVERAGE
Barnwell, SC									3.52	0.27	2.68	18.03	0.96	0.97	4.91
Beatty, NV	1.62	2.28	3.21	3.39	3.39	1.90	2.28	2.98	1.20	1.22	1.40	5.83	4.40	1.16	2.49
Morehead, KY	10.2	38.3	11.1	9.5	5.4	5.6	3.0	3.8	54.6	13.9	11.7	16.9	16.9	15.3	16.0
Richland, WA			0.22	0.42	6.18	15.4	128.0	125.0	41.0	48.6	55.2	8.63	75.5	36.4	34.6
Sheffield, IL					1.52	0.88	1.09	1.92	1.78	0.82	0.28	0.26	0.43	0.57	0.67
West Valley, NY	2.63	1.78	4.56	8.74	10.35	11.46	5.44	7.12	6.67	8.67	22.74	6.47	5.01		8.66
AVERAGE	4.77	12.68	7.01	6.59	5.86	5.94	5.72	6.15	27.4	8.62	8.44	10.66	7.98	5.66	8.99

TABLE A-4 (Con't)
COMMERCIAL LOW-LEVEL WASTE BURIAL HISTORY
BY PRODUCT MATERIAL BURIED

YEAR	1962	1963	1964	1965	1966	1967	1968	1969	1970	1971	1972	1973	1974	1975	1976	TOTAL
COMMERCIAL SITES																
Barnwell, SC Volume (m ³) Curies										1,171 4,118	3,757 997	15,839 42,500	18,244 329,043	18,072 17,420	28,829 27,890	85,912 421,968
Beatty, NV Volume (m ³) Curies	1,860 5,690	3,510 5,690	2,840 6,477	1,990 6,377	3,530 11,974	3,210 10,894	3,580 6,808	2,480 9,761	4,130 12,304	3,580 4,316	4,300 5,228	4,080 5,704	4,100 23,904	4,179 18,389	3,864 4,493	53,042 123,119
Morehead, KY Volume (m ³) Curies		2,210 22,716	3,870 147,322	5,750 63,828	5,560 52,729	7,820 42,280	8,180 45,578	10,400 31,027	12,500 46,968	13,200 720,146	15,600 217,350	10,100 118,274	8,520 143,656	17,098 289,581	13,783 211,356	134,591 2,153,802
Sheffield, IL Volume (m ³) Curies						2,530 3,850	2,710 2,381	2,010 2,192	2,830 5,427	4,430 7,895	5,960 4,857	8,530 2,834	12,400 3,229	14,112 6,104	13,480 7,744	68,992 46,513
Richland, WA Volume (m ³) Curies				670 144	2,400 1,006	870 5,378	670 10,330	440 55,964	420 52,820	580 23,916	680 31,809	1,033 57,037	1,410 12,173	1,500 113,341	2,867 104,306	13,520 468,224
West Valley, NY Volume (m ³) Curies		520 1,372	6,390 11,355	4,720 21,515	4,700 41,056	4,950 51,230	4,500 51,675	4,270 23,264	5,100 36,241	6,360 42,458	7,060 61,208	7,500 170,552	8,580 55,505	2,049 10,273		66,726 577,754
Total Volume (m ³) Curies	1,860 29,778	6,240 29,778	13,100 166,154	13,100 91,874	16,200 106,765	19,400 113,632	19,600 116,772	21,400 122,209	25,000 153,810	29,301 802,849	37,285 321,449	47,046 396,901	53,242 567,510	57,010 455,098	62,823 355,789	422,607 3,800,590

TABLE A-4 (Con't)
 LOW-LEVEL WASTE CHARACTERIZATION
 SPECIAL NUCLEAR MATERIALS* BURIED AT COMMERCIAL SITES

YEAR	1962	1963	1964	1965	1966	1967	1968	1969	1970	1971	1972	1973	1974	1975	1976	TOTAL
Barnwell, SC gm gm/m ³										13,220 11.3	46,718 12.4	99,800 6.30	110,444 6.05	64,425 3.56	92,800 3.22	427,407 4.97
Beatty, NV gm gm/m ³	319 0.17	41,304 11.8	172,030 60.6	334,762 168	5,872 1.66	22,644 7.05	8,602 2.40	5,005 1.17	7,708 1.87	757 0.21	21,177 4.92	15,164 3.72	16,954 4.14	29,276 7.01	2,096 0.54	683,670 12.89
Morehead, KY gm gm/m ³		959 0.43	11,889 3.07	4,261 0.74	7,462 1.34	14,842 1.90	17,771 2.17	31,506 3.03	47,562 3.80	72,770 5.51	71,443 4.58	46,244 4.58	23,832 2.80	25,690 1.50	27,474 1.99	403,705 3.00
Richland, WA gm gm/m ³				3 4.0	1,418 ~3	0.16 0.59	0.27 2.0	32 ~4	200 4.0	15 0.07	832 0.47	6,558 6.35	4,884 3.46	18,978 12.65	24,378 8.50	57,298 4.24
Sheffield, IL gm gm/m ³						1,238 0.49	1,754 0.65	3,843 1.91	5,649 2.0	9,934 2.24	5,898 0.99	6,126 0.72	6,198 0.50	5,285 0.37	1,738 0.13	47,663 0.69
West Valley, NY gm gm/m ³		952 1.82	3,273 0.51	2,433 0.52	4,999 1.06	3,446 0.70	2,045 0.45	7,301 1.71	8,273 1.62	4,816 0.76	7,321 1.04	7,710 1.03	2,984 0.35			56,003 0.84
TOTAL gm gm/m ³	319 0.17	43,215 6.93	187,192 14.29	341,459 26.06	19,751 1.22	42,170 2.17	30,172 1.54	47,687 2.23	69,392 2.78	101,512 3.46	153,389 4.11	181,107 3.85	166,296 3.12	143,654 2.52	149,486 2.36	1,675,801 3.97

* Fissile materials

TABLE A-4 (Con't)
LOW-LEVEL WASTE CHARACTERIZATION
SOURCE MATERIAL* BURIED AT COMMERCIAL SITES

	YEAR	1962	1963	1964	1965	1966	1967	1968	1969	1970	1971	1972	1973	1974	1975	1976	TOTAL
Barnwell, SC																	
Kg											12,546	1,606	45,305	26,961	46,005	16,259	148,682
Kg/m ³											10.7	0.43	2.86	1.48	2.55	0.56	1.73
Beatty, NV																	
Kg		296	472	331	236	91	346	1,040	290	322		9,340	11,500	9,710	1,438	5,000	40,412
Kg/m ³		0.16	0.13	0.12	0.12	0.03	0.11	0.29	0.07	0.08		2.17	2.82	2.37	0.34	1.29	0.76
Morehead, KY																	
Kg			5,210	5,590	568	690	5,680	6,250	2,550	7,220	5,730	8,260	9,340	13,100	82,416	75,944	228,548
Kg/m ³			2.36	1.44	0.10	0.12	0.73	0.76	0.25	0.58	0.43	0.53	0.43	1.54	4.82	5.51	1.70
Richland, WA																	
Kg					0.9	253.0	0.9	2.7	88.4	31.3	606	3,113	2,250	20.3	215	5,011	11,592
Kg/m ³					1.0 -3	0.11	1.0 -3	4.0 -3	0.20	0.07	1.04	4.76	2.18	0.014	0.14	1.75	0.86
Sheffield, IL																	
Kg							3,930	8,703	6,330	2,000	212	3,600	2,410	13,900	35,950	3,854	80,889
Kg/m ³							1.55	3.21	3.15	0.71	0.05	0.68	0.28	1.12	2.25	0.29	1.17
West Valley, NY																	
Kg			7,580	10,100	22,200	38,300	20,300	6,460	80,000	31,700	51,400	72,500	44,200	61,700			446,440
Kg/m ³			14.52	1.58	4.70	4.15	4.10	1.43	18.69	6.22	8.07	10.27	5.89	7.19			6.69
TOTAL																	
Kg		296	13,300	16,000	23,020	39,400	30,224	22,500	89,300	41,300	70,546	98,373	115,195	125,161	166,024	106,068	977,767
Kg/m ³		0.16	2.13	1.22	1.75	2.43	1.65	1.15	4.18	1.65	2.41	2.64	2.45	2.35	2.36	1.69	2.26

* Non-fissile uranium and thorium

TABLE A-5

NUCLIDE INVENTORY AS BURIED AT INEL

<u>Nuclide</u>	<u>% of Curie Activity</u>		
	<u>$T_{1/2}$ (days)</u>	<u>1974</u>	<u>1975</u>
Ce-Pr-144	285	--	4.3%
Ce-144	285	4.1%	3.0%
Co-60	1,920	38.8%	55.3%
Cr-51	28	0.4%	1.7%
Cs-134	10,877	0.3%	0.3%
Cs-137	10,877	7.2%	4.3%
Eu-154	5,840	0.2%	0.3%
Eu-155	661	0.1%	0.2%
Fe-55	486	--	2.7%
Fe-59	45	7.3%	1.5%
MAP	--	0.5%	0.1%
MFP	--	3.8%	2.2%
Mn-54	312	0.4%	2.8%
Mn-56	0.1	--	4.2%
Ni-59	29,000,000	16.2%	--
Pu-238	31,536	1.6%	--
Rb-86	19	0.3%	--
Ru-Rh-106	369	0.3%	0.6%
Ru-106	369	1.7%	1.3%
Sb-125	486	0.5%	0.6%
Sr-Y-90	10,439	5.5%	11.7%
Sr-90	10,439	8.0%	1.6%
Unident. $\beta+\gamma$	--	0.1%	0.1%
Zn-65	245	1.8%	--
Zr-NB-95	65	0.8%	0.9%
Total Curies Buried (Ci)		19,770	13,190
Total Volume Buried (m ³)		3,694	5,685

REFERENCES FOR APPENDIX A

1. "Alternatives for Managing Wastes from Reactors and Post Fission Operations in the LWR Fuel Cycle", ERDA-76-43, published by United States Energy Research & Development Administration.
2. "Management of Wastes from the LWR Fuel Cycle", CONF 76-0701, proceedings of International Symposium in Denver, Colorado, July 11-16, 1976.
3. "The Shallow Land Burial of Low-Level Radioactivity Contaminated Solid Waste", Panel on Land Burial, Committee on Radioactive Waste Management, Commission on Natural Resources, National Research Council, National Academy of Sciences, 1976.
4. "NRC Task Force Report on Review of the Federal/State Program for Regulation of Commercial Low-Level Radioactive Waste Burial Grounds", NUREG-0217, U. S. Nuclear Regulatory Commission, March 1977.
5. M. F. O'Connell and W. F. Holcomb, "A Summary of Low-Level Radioactive Wastes Buried at Commercial Sites Between 1962-1973, with Projections to the Year 2000", Radiation and Data Reports, December 1974.
6. "Final Environmental Statement on Waste Management Operations at the INEL", ERDA-1536, United States Energy Research & Development Administration, May 1977.
7. Dieckhoner, J. E., "Sources, Production Rates and Characteristics of ERDA Low-Level Wastes", paper presented at the Symposium on the Management of Low-Level Radioactive Waste, May 23-27, 1977, Atlanta, Georgia.

APPENDIX B

REFERENCE CONTAINMENT FACILITY DETAILS

Although the RCF is based on shallow land burial, by appropriate selection of parameters, the methodology employed is suitable for calculations based on alternative disposal methods. A list of design assumptions and criteria were formulated for the reference containment containment facility (RCF) used in the analyses. The following items have been assumed to be typical of future waste disposal facilities and appropriate for the RCF.

- a. The facility will utilize shallow land burial in open pits, covering the waste as filled with overburden removed during excavation.
- b. The wastes will arrive at the facility in DOT approved shipping containers suitable for burial and sized to facilitate handling with conventional materials handling equipment.
- c. Wastes will be solidified at their source prior to shipment. No provisions will be made for handling liquid radioactive wastes at the RCF.
- d. A 121 hectar (300 acre) site will be considered, assumed to be in a semiarid locale, located 5 km from a major highway, 2 km from a railroad main line, and 1 Km from a large river. It is assumed that between 1/2 and 2/3 of the total land area will be used for burial. The remaining area will allow for terrain, structures, service roads, and highway and railroad access areas.
- e. The site will be fenced with an intrusion alarm to preclude unauthorized entry of personnel.
- f. Permanent identification markers will be used to locate burial trench boundaries.

- g. The wastes will be covered with soil daily.
- h. Environmental monitoring for the migration of radioactive material will be provided before, during, and after the operational phases of the facility.
- i. After the site is filled, unnecessary structures will be removed, the area graded and seeded for erosion control, and a perpetual maintenance program established.
- j. A forty year operational lifetime for burial operations is assumed. Institutional control and surveillance of the site will be maintained for up to a few hundred years.
- k. About 2/3 of the trench volume is assumed to consist of soil mixed between waste containers as the trenches are filled.
- l. In calculating the concentrations in human exposure pathways, no credit will be taken for the retention of radionuclides in their containers in which the waste is buried.

With 100 typical trenches filled to capacity with wastes, the total site capacity is $569,000 \text{ m}^3$ of waste. This is about $14,000 \text{ m}^3$ per year for a 40 year facility lifetime. At a uniform fill rate, approximately 280 55-gallon drums per day are required to fill the site in 40 years. This capacity is sufficient to handle the wastes projected to be generated by about 1000 Gw(e) yr of nuclear power production.

Table B-1 lists the cost factors used to arrive at the unit costs of \$100/m³ of waste disposed. These costs are consistent with changes now assessed by the commercial site operators across the country.

TABLE B-1 COST FACTORS FOR RCF

COST FACTORS

COSTS

Capital

Site Acquisition & Development	\$ 3,500,000
Structural Construction	1,200,000
Site Preparation	230,000
Operating Equipment	1,140,000

Operating Costs

Labor	\$317,000/yr x 40/yr	14,700,000
Supplies	7,000/yr x 40/yr	280,000
Materials	20,000/yr x 40/yr	800,000
Environmental Impact Statement		500,000
Architect-Engineering		210,000

Site Maintenance & Perpetual Care 770,000

Net Costs \$23,330,000
Financing Charges 28,200,000

Total Costs \$51,530,000

Site Capacity 569,000 m³

Unit Costs \$90/m³ Consider Contingencies to make
\$100/m³ reasonably profitable
costs

APPENDIX C

DETAILS OF CALCULATIONS FOR AIRBORNE RELEASES

In the determination of the allowable concentration limits in the waste to be placed in the low-level burial RCF, various potential airborne releases are considered to identify limiting cases. Two potential occurrences which lead to limiting values of concentration are an accident resulting in the release of a fraction of the contents of a barrel during the burial operation, and the reclamation of land used as a burial site after loss of institutional control of the site after a few hundred years. The first case establishes a limit on concentration considering both short and long half-life elements, while the second case sets a limit based primarily on long half-life elements. In addition to the two limiting cases discussed above, a continuous release was also investigated.

C.1. Single Barrel Accident

It is assumed that low-level wastes will be packaged for disposal in 55 gallon drums or barrels as is the present practice. The single barrel accident is defined as the instantaneous release of a fraction of the contents of a barrel during the burial operation. The major pathway of concern for dose to the off-site population is the airborne transport of the material released from the barrel. For the purposes of this analysis, the following conservative yet realistic assumptions have been made:

1. 0.1% of the barrel contents become airborne
2. The average particle size is such that the deposition velocity is 5 cm/sec.

3. A standard man is on the plume centerline at the site boundary 160 m (0.1 mi) from the source. He is engaged in light activity (20 l/min respiration rate).
4. A Pasquill F stability level is used.
5. The dose to any organ should not exceed 0.5 rem/yr.

The concentration at the plume centerline for an instantaneous point source is given by: ⁽¹⁾

$$X(x,y,z,t) = \frac{Q(2\pi)^{-3/2}}{\sigma_x \sigma_y \sigma_z} \exp \left(- \frac{(x - \bar{u}t)^2}{2\sigma_x^2} \right)$$

where $X(x,y,z,t)$ = concentration, Ci/m³

Q = source strength, Ci

$\sigma_x, \sigma_y, \sigma_z$ = dispersion coefficients, m

$\bar{u}$ = average wind speed, m/s

x = direction of plume axis (wind)

This equation is integrated to get the integrated concentration-time exposure

$$I = \int_0^\infty X \, dt = \frac{Q}{\pi \sigma_y \sigma_z \bar{u}} \text{ in } \frac{\text{Ci-sec}}{\text{m}^3}$$

For an F stability level, $\sigma_y \approx 7$ and $\sigma_z \approx 3.5$, and using $\bar{u} = 1.56 \text{ m/sec (3.5 mph)}$,

$$I = 8.33 \times 10^{-3} Q \frac{\text{Ci released-sec}}{\text{m}^3}$$

In order to correct for deposition of dust particles before they arrive at the site boundary, an effective source is calculated.

$$Q_{\text{eff}} = Q_0 \exp \left[-\sqrt{2/\pi} V_g / \bar{u} \int_0^x \frac{dx'}{\sigma_z(x')} \right]$$

where Q_{eff} = effective source strength, Ci

Q_0 = actual source strength, Ci

V_g = deposition velocity, m/s

$\bar{u}$ = average wind velocity, m/s

$\sigma_z(x)$ = vertical dispersion coefficient, m

x = distance from source to observation point in wind direction

The ratio Q_{eff}/Q_0 is plotted in Figure C-1 and the value for Q_{eff}/Q_0 at 160 m (0.1 mi) from the source for the conditions given above is 0.24 for a 5 cm/sec deposition velocity. Other deposition velocity results are shown in Figure C-1 for perspective but the 5 cm/sec value is used in this evaluation.

The integrated concentration exposure becomes,

$$I' = 1.98 \times 10^{-3} Q_0 \frac{\text{Ci released-sec}}{\text{m}^3}$$

The number of curies inhaled is the product of the integrated concentration exposure and the respiration rate,

FIGURE C-1
EFFECT OF DEPOSITION VELOCITY ON CONCENTRATION
AT SITE BOUNDARY (0.1 MI FROM ACCIDENT)

$$Ci_{\text{inhaled}} = (1.98 \times 10^{-3} \frac{Ci_{\text{released-sec}}}{m^3}) (3.33 \times 10^{-4} \frac{m^3}{\text{sec}})$$

$$Ci_{\text{inhaled}} = 6.59 \times 10^{-7} Q_0 Ci_{\text{released}}$$

The ratio of curies released to curies inhaled is 1.5×10^6 .

The allowable dose to an organ is 0.5 rem/yr, but there are several contributors that may be involved with that dose. Using the dose conversion factors for ^{239}Pu (from the next section) of $1 \mu\text{Ci inhaled} = 30 \text{ rem/yr}$, the allowable inhalation is $0.02 \mu\text{Ci}$. Considering the release fraction of 10^{-3} , and the barrel volume of 0.208 m^3 , the maximum allowable concentration in the barrel is 60 Ci/m^3 or $60 \mu\text{Ci/ml}$.

Elements other than ^{239}Pu have allowable limits that are a function of their hazard potential. In order to estimate the limit of each, the maximum permissible concentration in air, MPC_a , is used. The limit for each element is a function of the ratio of the MPC for that element relative to that for ^{239}Pu . It is assumed in this evaluation that if the allowable concentration is double that of ^{239}Pu , the hazard index is half that of ^{239}Pu .

C.2. Plutonium Inhalation Dose Conversion Factor

In hazard evaluations it is necessary to convert the inhaled activity to dose. To scope the magnitude of the dose conversion factor, assume an inhalation of $1 \mu\text{Ci}$ of ^{239}Pu , with approximately 20% ⁽³⁾ reaching the bone. Then the activity in the bone is $7400 \text{ disintegrations/sec}$ per μCi inhaled. Using

an alpha energy of 5 MeV/dis (8×10^{-6} erg/dis), there is 0.059 erg/sec deposited in bone. The resulting energy deposition rate is 8.5×10^{-6} erg/gm-s for a standard man bone mass of 7Kg. (4) For alpha particles, 1 rem = 10 erg/gm; therefore, the dose to bone is 8.5×10^{-7} rem/sec, or 26.8 rem/yr. In this rough estimate, the energy from decay products and the biological half-life of ^{239}Pu have been neglected in obtaining this number.

Integrating the dose from inhalation of $1\mu\text{Ci } ^{239}\text{Pu}$ over a 50 year period gives a total integrated dose of about 1300 rem total lifetime dose to the bone. A computer calculation (5) of inhalation doses based on the ICRP lung model was used to check these results. Figure C-2 shows the total integrated lifetime dose from the inhalation of $1\mu\text{Ci}$ of ^{239}Pu as a function of dust particle size. The accumulation of dose as a function of time since exposure is shown in Figure C-3 for $5\mu\text{m}$ particles. The value of 30 rem/yr per μCi of ^{239}Pu inhaled is conservative for the $5\mu\text{m}$ particles assumed in our study.

Appendix VI of WASH-1400 (6) also gives dose conversion factors for ^{239}Pu and other nuclides of interest.

Because of differences in residence times in the various organs after inhalation, doses to the organs vary in both intensity and duration, as shown in Figure C.3. For ^{239}Pu , for instance, the dose to the lung is large in the first few years and essentially zero thereafter, while the dose to the bone builds to a maximum 10 to 20 years after the exposure. The total lifetime dose is greater to the bone. However, to keep the annual dose to the lung under 500 mR, less plutonium may be inhaled than to the amount that will result in a 500 mR/yr dose to the

Figure C-2. Total Lifetime Integrated Bone Dose Versus Dust Particle Size

Figure C-3. Total Integrated Dose Versus Time After Exposure

bone. Alternatively, inhalation of 0.020 μCi of ^{239}Pu gives a 4 Rem 50 year dose to the lung with a peak dose rate of 500 mR/yr, while inhalation of 0.023 μCi gives 20 Rem 50 year dose to the bone with a peak dose rate of 500 mR/yr.

The objectives of this study do not include determination of the correct or appropriate guidelines for exposures. The values are used to illustrate the applicability of the methodology for quantifying the waste classification system interfaces. Other guidelines could be used. The results will scale proportionally to the limiting dose accepted. Because the dose to the bones persists longer for plutonium, the bone is generally considered the critical organ for inhalation. Interface limits were determined for both the lung and the bone values given above.

C.3 Continuous Airborne Source

The effect of a continuous airborne source model was considered in determining limiting cases. It is assumed that barrels are ruptured periodically, and that the ensuing release is continuous with time. The airborne release is then transported to the site boundary. The following assumptions have been used:

1. Barrel failure probability is 10^{-4} .
2. Airborne release fraction is 10^{-3} .
3. The concentration of the isotopes of concern is evaluated at the site boundary (160 m).
4. Pasquill E stability level is used as a conservative level for continuous release, with an isotropical wind-rose from source.

TABLE C-1

CONTINUOUS RELEASE AIRBORNE DOSE SUMMARY

Isotope	Inventory per Barrel, Ci	Concentration at 160 m pCi/l Continuous Source ^a	MPC, pCi/l	% MPC ^c
Co ⁶⁰	6.67	8.3×10^{-5}	0.3	3×10^{-2}
Sr ⁹⁰	0.89	1.1×10^{-5}	0.03	4×10^{-2}
Cs ¹³⁷	1.30	1.6×10^{-5}	0.5	3×10^{-3}
I ¹²⁹	2.7×10^{-7}	3.5×10^{-12}	0.02	0
Pu ²³⁹	0.024	3.0×10^{-7}	6×10^{-5}	5×10^{-1}
Np ²³⁷	0.0064	7.9×10^{-8}	1×10^{-4}	8×10^{-2}
Am ²⁴¹	7.4×10^{-4}	9.3×10^{-9}	2×10^{-4}	5×10^{-3}

- A. Probability of barrel failure assumed 10^{-4}
Airborne release fraction 0.1%.
- B. Ratio, in %, of largest concentration at 0.1 mi
to MPC for continuous release.

The continuous source model is derived from the instantaneous source model discussed above for the accident case. An infinite number of instantaneous releases are considered by integrating the instantaneous release equation, which results in the following: (1)

$$x = \frac{Q}{\pi \sigma_y \sigma_z \bar{u}} \exp \left[- \frac{y^2}{2 \sigma_y^2} \right]$$

This equation is integrated numerically to include the effects of wind direction variation to provide average concentration at a given distance from the source. At 160 m from the source, the results of the concentration calculation are as indicated in Table C-1. The concentrations are well below the MPC level for all isotopes of interest, and the continuous release model does not result in a limiting case.

C.4. Reclamation

Another potential occurrence considered is the reclamation of the burial facility site after an extended period of time. For this case, people are considered to be exposed to radioactively contaminated dust while moving earth at the site. The following assumptions are used in the calculations:

1. The working atmosphere to which workers are exposed contains 2 mg/m³ of dust, (7)

2. The workers spend an average of 1 month working at the site, and do not reside there. Consequently, a 170 hour exposure time is used.
3. Respiratory rate is 20 l/min.
4. The maximum allowable organ dose is 0.5 rem/yr.
5. The LLW is assumed to be diluted by a factor of 3 by earth during burial.
6. The maximum interface level concentration is 10 times the average concentration of radioactivity in the wastes.

The air inhaled by a worker at the site is 200 m^3 . Since there are 2 mg/m^3 of dust in the air, the worker inhales 0.4 g of dust, all of it retained in the lungs.

As shown in Section C.1, the inhalation limit for ^{239}Pu is 0.02 μCi . Since 0.4 g were inhaled, the maximum concentration is 0.05 $\mu\text{Ci/g}$ of dust inhaled. Assuming a density of 1.0 g/cc for the airborne particles yields 0.05 $\mu\text{Ci/cc}$. Considering the factor of 3 dilution due to burial, and the factor of 10 between maximum and average concentration leads to a concentration limit of 1 $\mu\text{Ci/cc}$ for ^{239}Pu . Similar calculations using the Rasmussen study^(c) conversion factors for other nuclides gives the results shown in Table C-2.

C.5. Population Doses

Population doses for the accident case were estimated based on the reference population distribution shown in Table C-3, and the same conditions used for the maximum individual dose calculations. Table C-4 summarizes these population dose calculations.

Table C.2. Nuclide Specific Concentration Interfaces

Nuclide	uCi Inhaled	Concentration Implied (uCi/cc)	RHI	RHI(e ^{λt})
4 Rem 50 Year Dose to Lung				
Co-60	3.08	231	770	3.9x10 ⁸
Sr-90	222	16,700	556000	5.6x10 ⁶
Cs-137	100	7500	15000	1.5x10 ⁵
Pu-239	0.02	1.5	25000	2.5x10 ⁴
Am-241	0.01	0.9	4700	5500
20 Rem 50 Year Dose to Bone				
Co-60	400	30,000	100000	5.0x10 ¹⁰
Sr-90	7.1	540	18000	1.8x10 ⁵
Cs-137	560	42,000	83000	8.3x10 ⁵
Pu-239	0.023	1.7	28000	2.8x10 ⁴
Am-241	0.022	1.6	8200	9.9x10 ³

TABLE C-3
REFERENCE POPULATION DISTRIBUTION

Radial Distance From Site (m)	Population
800	30
1,600	30
3,200	200
8,000	200
16,000	2,500
24,000	1,500
32,000	4,000
40,000	4,000
48,000	4,000
64,000	13,000
80,000	25,600
Total within 50-mile Radius	55,060

TABLE C-4

POPULATION DOSE FROM ACCIDENT SCENARIO

Distance from Source	Population along Centerline ^a	σ_y	σ_z	Q_{eff}/Q_0 5cm/sec deposition velocity	Q_{eff}	Dose (manrem/yr)
800	8	31	12	8.2×10^{-2}	2.0×10^{-6}	8×10^{-5}
1,000	8	53	18	4.5×10^{-2}	1.1×10^{-6}	2×10^{-5}
3,200	50	100	29	2.1×10^{-2}	5.0×10^{-7}	2×10^{-5}
8,000	50	230	42	5.6×10^{-3}	1.3×10^{-8}	1×10^{-6}
16,000	600	400	54	1.3×10^{-3}	3.1×10^{-8}	2×10^{-6}
24,000	400	570	60	4.6×10^{-4}	1.1×10^{-8}	2×10^{-7}
32,000	1000	750	68	1.9×10^{-4}	4.6×10^{-9}	2×10^{-7}
40,000	1000	900	70	9.0×10^{-5}	2.2×10^{-9}	7×10^{-8}
48,000	1000	1000	75	4.4×10^{-5}	1.1×10^{-9}	3×10^{-7}
64,000	3200	1300	80	1.3×10^{-5}	3.1×10^{-10}	2×10^{-8}
80,000	6400	1600	85	4.4×10^{-6}	1.1×10^{-10}	1×10^{-8}
TOTALS	13716					1.2×10^{-4}

This gives 6×10^{-3} manrem lifetime dose for the reference ^{239}Pu concentration per accident.

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

REFERENCES FOR APPENDIX C

1. Slade, D.H., Meteorology and Atomic Energy 1968, U.S. A.E.C., TID 24190, July 1968.
2. Culkowski, W.M. and M.R. Patterson, "A Comprehensive Atmospheric Transport and Diffusion Model," ORNL/NSF/EATL-17, April 1976.
3. Coates, R.L. and N.R. Horton, "RSAC - A Radiological Safety Analysis Computer Program," IDO-17151, May, 1966.
4. Morgan, Karl Z., W.S. Snyder, and J.A. Auxier (eds), "Report of ICRP Committee II on Permissible Dose for Internal Radiation (1959), with Bibliograph for Biological, Mathematical, and Physical Data," Health Phys., 3 (June, 1960).
5. "AERIN Computer Code", Unpublished. Lawrence Livermore Laboratory.
6. Rasmussen, et al, " Reactor Safety Study" WASH-1400 (NUREG 75/014) Final Version, 1975.
7. Yanskey, G. R., et al, "Climatology of the NRTS," IDO-12048, Idaho Operations Office, USAEC, Jan. 1966.

APPENDIX D

DETAILS OF THE CALCULATIONS
FOR WATERBORNE RELEASES

D.1. Introduction

Migration of radionuclides through the geosphere from their burial site represents an important exposure pathway to the environment and should be considered in any environmental assessment of nuclear waste management alternatives. There are several processes which influence migration rates and change the rates at which nuclides are released to water systems in the environment. Among these processes are erosion and suspension of waste particles in surface water flows, the leaching of the buried inventory (the process wherein the nuclide is taken up from its original form and becomes suspended or dissolved in water), convection (nuclide movement caused by gross water movement), dispersion (nuclide movement caused by the nuclide concentration gradient), ion exchange between the nuclides in the water and the matrix through which the water flows, and radioactive decay.

In the present analysis, the effect of ingrowth of a particular nuclide from decay of a parent has not been considered. The effects of radioactive decay have been included to show the decrease in amount of radioactivity with time.

This appendix contains a description of the various processes and parameters which have been evaluated in this investigation, a presentation of the results of base cases for several nuclides, and a parametric sensitivity analysis.

D.2. Leaching Model

Leaching is the process by which buried materials are dissolved and enter the groundwater. That is, the rate of addition of a nuclide to water is given by:

$$\frac{dM_w}{dt} = M_{IO} \lambda_l \exp [-(\lambda_l + \lambda_d)t] \quad , \text{ Ci/yr}$$

where M_{IO} is the initial inventory, λ_l and λ_d are the leach and decay constants, respectively.

The inventory remaining in the burial location at any time is described by:

$$M_I(t) = M_{IO} \exp [-(\lambda_l + \lambda_d)t] \quad , \text{ Ci}$$

Values of leach rates for the several nuclides considered in this study are obtained using data from drill cores taken in soil below actual low-level disposal sites at INEL.⁽¹⁾ It is assumed that any concentrations of a given nuclide below the pit or pond result from leaching. Thus, by integrating the concentration over the contaminated depth, correcting for radioactive decay and knowing the burial and sample times, an order of magnitude estimate for the leach rates is obtained.

D.3. Nuclide Migration

Nuclide migration is determined from a second order differential mass balance equation which considers longitudinal dispersion, convection, sorption and radioactive decay. The equation is based on a one-dimensional homo-

genous medium and describes the concentration as a function of space and time. (2)

$$D \frac{\partial^2 C}{\partial x^2} - V \frac{\partial C}{\partial x} - K \frac{\partial C}{\partial t} - \lambda_d C = 0 \quad , \text{ Ci/m}^3 \cdot \text{yr}$$

where C is the nuclide concentration, D is the dispersion coefficient, V is the groundwater velocity, and K is the equilibrium sorption coefficient. The first term in the equation represents longitudinal dispersion. For conservatism and simplicity, lateral dispersion has been ignored. The second term represents convection, the third term sorption processes and the fourth radioactive decay.

Using Laplace transform techniques on the above equation with boundary conditions characterized by a series of unit step functions, the solution is:

$$C(x,t) = \frac{1}{2} \exp \left(\frac{Vx}{2D} \right) \left[\sum_{j=1}^N C_j f(t - \tau_j) u(t - \tau_j) \right] , \text{ Ci/m}^3$$

where N is the number of steps in the boundary condition, τ_j is the time at which the jth step "turns on", u is the unit step function and

$$f(t - \tau_j) = \left\{ \exp(-ab) \operatorname{erfc} \left[\frac{a - 2b(t - \tau_j)}{2(t - \tau_j)^{1/2}} \right] + \exp(ab) \operatorname{erfc} \left[\frac{a + 2b(t - \tau_j)}{2(t - \tau_j)^{1/2}} \right] \right\}$$

with

$$a = \left(\frac{K}{D} \right)^{1/2} \cdot x \quad \text{and}$$

$$b = \left(\frac{V^2}{4DK} + \frac{\lambda_d}{K} \right)^{1/2}$$

and

$$erfc(b) = 1 - \frac{2}{\sqrt{\pi}} \int_0^b e^{-z^2} dz$$

The sorption processes are characterized by the equilibrium sorption coefficient, or time transformation factor, ⁽⁴⁾ K, which is also expressed as a relative nuclide velocity, i.e.

$$K = \frac{V_{\text{water}}}{V_{\text{nuclide}}}$$

Alternatively, K is related to the distribution coefficient (the ratio of the concentration of the nuclide in the soil to the concentration of the nuclide in the water) by the equation

$$K = 1 + \frac{\rho}{\epsilon} K_d$$

where K_d is the distribution coefficient, ρ is the soil density, and ϵ is soil porosity.

The time of arrival of the contamination front is given approximately by:

$$t_a = \frac{KL}{V_{\text{water}}} \quad , \text{ yr}$$

where L is the length of the aquifer.

D.4. Soil and Site Characteristics

In this study, the soil has been characterized by the dispersion coefficient, D, and the equilibrium sorption coefficient, K. The value of D is $0.42 \text{ m}^2/\text{yr}$ ⁽²⁾ and the values of K for the various nuclides are shown in

Table D-1. (3)

Other parameters which characterize soils have been implicitly assumed in the selection of groundwater velocities. (2) These parameters include permeability, porosity and water pressure gradient.

As has been elsewhere noted, (5) there is very little information available about the sorption coefficient for soils. The values of Table D-1 are for a typical western desert soil.

The parameters representative of the RCF are site area, number of pits, distance from pit bottoms to aquifer, aquifer flow area, aquifer velocity, distance from site to nearest surface waters and rain rate. Values of these parameters are shown in Table D-2.

D.5. Base Cases

Table D-3 contains the results of the base case analysis. Strontium-90, Cesium-137 and the Americium-241 initially present do not appear at the aquifer outlet because the initial inventories have essentially completely decayed before they reach the aquifer outlet. The effects of ingrowth of radioactive daughter products may be important (6) but have not been considered. Matching calculations performed by TASC (7) are also given in Table D-3. The agreement is good.

D.6. Erosion

It is reasonably conservative to assume that future confinement 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 would ultimately be released to surface waters or dispersed into the atmosphere. Design features, such as covering the filled burial area with pebbles, through which grass vegetation could be established, would tend to 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, 6 tons of soil per acre per year is a typical sheet erosion rate.⁽⁸⁾ Using this rate, the RCF pit surface area of 50 acres, and a soil density of 2 gm/cm³, it is determined to require 1500 years for a one meter surface cover to be eroded away before the buried wastes begin to erode. Assuming a dilution factor of 3 to account for mixing of the wastes with soil during burial, a factor of 10 to account for the difference between the interface and average concentrations of activity in the

waste, and ignoring dilution and hold up of eroded material between the pits and the river, the concentration in the eroded material that will give guideline doses to maximum individuals and populations can be determined.

If all of the material eroded goes directly into the river, (with a volumetric flow rate of $2550 \text{ m}^3/\text{sec}$), and if sedimentation is neglected, then it would take an initial waste concentration of $10^5 \text{ } \mu\text{Ci/cc}$ of ^{239}Pu to give a maximum individual dose of 0.5 rem/yr , based on the dose conversion factors given in Table 5-3. This calculation presumes that the erosion takes place uniformly over the total surface and that gullying and preferential erosion do not occur. For the erosion case, where potentially large populations may be involved, guideline doses of 5 mR/yr to the maximum individuals may be more appropriate. This would allow concentrations of ^{239}Pu in the wastes of $1000 \text{ } \mu\text{Ci/cc}$.

It is evident that, even with overly conservative assumptions, erosion does not give the most restrictive concentration limitation. Factors such as establishment of a dense vegetative cover over the buried wastes and siting the containment facility where sheet erosion does not occur would reduce even further any possible consequences of erosion.

D.7. Parametric Variations

Figure D-1 contains the effect of variation of the leach rate, $\frac{dM_w}{dt}$, on the concentration at the pit bottom. As the

value of the leach rate decreases significantly below the decay rate, the decay rate becomes more dominant in determining the rate at which the concentration changes. The converse is also true. Figure D-1 also demonstrates that the concentration increases as the leach rate increases.

The influence of the dispersion coefficient, D , on the response at the aquifer outlet is shown in Figure D-2. As the dispersion of the pulse increases, i.e., as the dispersion coefficient increases, the pulse becomes broader and is reduced in magnitude.

It is shown in Figure D-3 that as the hypothetical pulse boundary condition is decreased in duration (but not in magnitude) with a constant dispersion coefficient, the magnitude of the concentration at the aquifer outlet is accordingly decreased. That is, as the pulse is shortened, the output has less time to equilibrate and achieves a smaller fraction of equilibrium concentration.

Figure D-4 shows the effect on individual doses of varying the volume flow rate of the stream into which the wastes flow. This study used a base case of a rather large flow typical of the Columbia River. As long as the relative uses of water along the stream remain constant, the population doses do not depend on the specific volumetric flow rate.

TABLE D-1

NUCLIDE SPECIFIC PARAMETERS AND VALUES

	<u>Initial Inventory(Ci)</u>	<u>Sorption Coefficient</u>	<u>Leach rate(yr⁻¹)</u>
Strontium-90	24000	100	1×10^{-2}
Iodine-129	0.73	1	1×10^{-2}
Cesium-137	35000	1000	1×10^{-2}
Neptunium-237	12000	100	6×10^{-4}
Plutonium-239	65000	10000	6×10^{-4}
Americium-241	65000	1000	6×10^{-4}

TABLE D-2

SITE-SPECIFIC PARAMETERS AND VALUES

Site plan area	2×10^6	m ²
Number of pits	100	ea
Distance between pit and aquifer	10	m
Water velocity from pit to aquifer	10	m/yr
Rain rate	0.5 -2	m/yr
Aquifer flow area	1000	m ²
Distance from site to surface water	1000	m
Aquifer water velocity	111	m/yr
Dispersion Coefficient	0.42	m ² /yr
River volume Flow Rate at nuclide inlet	2550	m ³ /s
River volume Flow Rate at nuclide outlet	3790	m ³ /s

TABLE D-3
PRELIMINARY DOSE CALCULATIONS

FBDU MODEL

Isotope	Burial Facility Inventory Kg	Burial Facility Inventory Ci	Average Burial Facility Activity Ci/m ³	Arrival Time of Contamination Front, Years (Half lives)	Peak Fraction of Original Inventory Released to River, yr ⁻¹	Activity into River per year After Arrival of Contamination Front, Ci/Yr	Average Individual 50 yr Body Dose, Rem	Total Population Dose, Man-Rem 50 yr
239Pu	1060	6.5 x 10 ⁴	0.11	10 ⁵ (4.1)	2.0 x 10 ⁻⁷	1.3 x 10 ⁻²	1.2 x 10 ⁻⁷	8.9 x 10 ⁻²
241Am	1995	6.5 x 10 ⁶	11.0	10 ⁵ (22.)	—	—	—	—
237Np	1.72 x 10 ⁴	1.2 x 10 ⁴	2.1 x 10 ⁻²	10 ³ (4.7 x 10 ⁻⁴)	6.0 x 10 ⁻⁶	7.2 x 10 ⁻²	1.1 x 10 ⁻⁶	8.8 x 10 ⁻¹
129I	4.2	0.73	1.3 x 10 ⁻⁶	10 ¹ (6.3 x 10 ⁻⁷)	9.0 x 10 ⁻³	6.6 x 10 ⁻³	9.9 x 10 ⁻⁸	7.9 x 10 ⁻²
137Cs	40.	3.5 x 10 ⁶	6.2	10 ⁴ (34.)	—	—	—	—
90Sr	17.	2.4 x 10 ⁶	4.2	10 ³ (35.)	—	—	—	—

TASC MODEL

Isotope	Burial Facility Inventory Kg	Burial Facility Inventory Ci	Average Burial Facility Activity Ci/m ³	Arrival Time of Contamination Front Years (Half lives)	Peak Fraction of Original Inventory Released to River, yr ⁻¹	Activity into River per year After Arrival of Contamination Front, Ci/Yr	Average Individual 50 yr Body Dose, Rem	Total Population Dose, Man-Rem 50 yr
239Pu	1060	6.5 x 10 ⁴	0.11	1.3 x 10 ⁵ (5.3)	1.8 x 10 ⁻⁸	1.1 x 10 ⁻³	1.0 x 10 ⁻⁸	8.0 x 10 ⁻³
237Np	1.72 x 10 ⁴	1.2 x 10 ⁴	2.1 x 10 ⁻²	10 ³ (4.7 x 10 ⁻⁴)	6 x 10 ⁻⁶	7.2 x 10 ⁻²	1.1 x 10 ⁻⁶	8.8 x 10 ⁻¹
129I	4.2	0.73	1.3 x 10 ⁻⁶	17 (1.1 x 10 ⁻⁶)	10 ⁻²	7.3 x 10 ⁻³	1.1 x 10 ⁻⁷	8.8 x 10 ⁻²

FIGURE D-1

CONCENTRATION AT PIT BOTTOM FOR ^{239}Pu vs. TIME WITH LEACH RATE AS A PARAMETER

LEGEND

DISTANCE = 1000m.
SORPTION COEFFICIENT = 10000
PULSE DURATION = 10000 YR
WATER SPEED = 111 m/yr

FIGURE D-2

CONCENTRATION AT AQUIFER OUTLET VS. TIME WITH DISPERSION COEFFICIENT
AS A PARAMETER FOR 239 PU

FIGURE D-3
CONCENTRATION AT AQUIFER OUTLET VS TIME FOR ²³⁹Pu WITH
BOUNDARY CONDITION PULSE LENGTH AS A PARAMETER

FIGURE D-4

EFFECTS OF WATER VOLUMETRIC FLOW RATE IN STREAM
ON MAXIMUM INDIVIDUAL DOSES

REFERENCES FOR APPENDIX D

1. Schmalz, B. L., "Radionuclide Distribution in Soil Mantle of the Lithosphere as a Consequence of Waste Disposal at the National Reactor Testing Station", October 1972, Idaho Operations Office, U. S. Atomic Energy Commission, IDO-10049.
2. Lester, D. H., Jansen, G., and Burkholder, H. C., "Migration of Radionuclide Chains Through An Absorbing Medium", December 1974, Battelle Pacific Northwest Laboratories, BNWL-SA-5079.
3. Burkholder, H. C., et al., "Incentives for Partitioning High Level Waste", November 1975, Battelle Pacific Northwest Laboratories, BNWL-1927, UC-70.
4. Inoue, Y. and Morisawa, S., "On The Selection of a Ground Disposal Site for Radioactive Wastes: An Approach To Its Safety Evaluation", Health Physics, Vol 26, pp. 53-63, 1974.
5. Huff, D. D. and Kruger, P., "Simulation of the Hydrologic Transport of Radioactive Aerosols", Chapter 27 of Radionuclides in the Environment, Advances in Chemistry Series 93, published by American Chemical Society, Washington, D. C., 1970.
6. Burkholder, H. C., "Methods and Data for Predicting Nuclide Migration in Geologic Media", Proceedings of the International Symposium on Management of Wastes from the LWR Fuel Cycle, (CONF-76-0701), July 11-16, 1976, Denver, Colorado.
7. The Analytical Sciences Corporation, personal communication (May 1977).
8. Smith, D. D. and Wischmeier, W. H., "Factors Affecting Sheet and Rill Erosion" Trans., American Geophysical Union, December 1957.

APPENDIX E

APPLICATION OF RHI CONCEPT
FOR INDIVIDUAL NUCLIDES AND MIXTURES

APPENDIX E

APPLICATION OF RHI CONCEPT FOR INDIVIDUAL NUCLIDES AND MIXTURES

Let $C_1, C_2, C_3, \dots, C_n$ = concentration of various nuclides in the waste.

For a single nuclide we have by definition

$$RHI = \frac{C K}{MPC_a}$$

where C = concentration in

$\mu\text{Ci/ml}$

$$K = \frac{\text{dust loading}}{\text{soil density}}$$

MPC_a = maximum permissible
concentration in air

For any mixture of nuclides, the total hazard index is given by the sum of the individual RHI's for each nuclide. That is,

$$(RHI)_T = (RHI)_1 + (RHI)_2 + (RHI)_3 + \dots + (RHI)_n$$

The $(RHI)_T$ must not exceed the interface value of 1.7×10^4 as shown in Figure 5.

(An activity weighted average half-life could be used to increase the acceptable total RHI if desired. If the average half-life cannot be reasonably established, then it should be assumed to be long enough to give 1.7×10^4 for the total RHI.)

Rewriting the above equation,

$$(RHI)_T = \frac{C_1 K}{MPC_1} + \frac{C_2 K}{MPC_2} + \frac{C_3 K}{MPC_3} + \dots + \frac{C_n K}{MPC_n} \quad (E-1)$$

The interface concentration, here designated by C_I , is found by solving for C when given K and $(RHI)_T$.

$$C_I = \frac{RHI_T MPC_a}{K} \quad (2) \quad (E-2)$$

Rearranging terms in eq (E-1) yields

$$\frac{C_1 K}{MPC_1 RHI_T} + \frac{C_2 K}{MPC_2 RHI_T} + \frac{C_3 K}{MPC_3 RHI_T} + \dots + \frac{C_n K}{MPC_n RHI_T} = 1 \quad (E-3)$$

Substituting eq. (E-2) into eq (E-3) gives

$$\frac{C_1}{C_{I1}} + \frac{C_2}{C_{I2}} + \frac{C_3}{C_{I3}} + \dots + \frac{C_n}{C_{In}} = 1 \quad (E-4)$$

Therefore, if the sum of the left hand side of eq (E-4) is ≤ 1 , the material may be considered low level waste.

Example: suppose the waste contains the following mixture of nuclides:

⁹⁰ Sr	2.3x10 ²	μC/ml
⁹⁰ Y	2.3x10 ²	μC/ml
¹³⁷ Cs	8.8x10 ³	μC/ml
²³⁹ Pu	4.2x10 ⁻¹	μC/ml

Question: Is the material low-level waste?

Using eq (E-4) with the values of C_I obtained from Table 4-2 yields :

$$\frac{2.3 \times 10^2}{5.6 \times 10^3} + \frac{2.3 \times 10^2}{3 \times 10^6} + \frac{8.8 \times 10^3}{8.5 \times 10^4} + \frac{4.2 \times 10^{-1}}{1.0} = y$$

$y = 0.56$, which is less than 1.

The answer to the question then is yes, it is low-level waste.

APPENDIX F

TECHNICAL ADVISORY PANEL
MEMBERS AND PARTICIPANTS

APPENDIX F

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