KEY WORDS:
At-grade Disposal
Performance Assessment
Waste Management

United States Department of Energy

Savannah River Site

SPECIAL ANALYSIS: NAVAL REACTOR WASTE DISPOSAL PAD (U)

WSRC-RP-2001-00948

Revision 2

December 19, 2002

Prepared by:
Andrew D. Yu[a]
Laura M. McDowell-Boyer[a]
James R. Cook[b]
Karen E. Young[c]

[a] Alara Environmental Analysis, Inc.
[b] Westinghouse Savannah River Company
[c] Exploration Resources, Inc.

Westinghouse Savannah River Company, LLC
Savannah River Site
Aiken, SC 29808

Prepared for the U.S. Department of Energy under Contract No. DE-AC09-96SR18500

KEY WORDS:
At-grade Disposal
Performance Assessment
Waste Management

United States Department of Energy

Savannah River Site

SPECIAL ANALYSIS: NAVAL REACTOR WASTE DISPOSAL PAD (U)

WSRC-RP-2001-00948

Revision 2

December 19, 2002

Prepared by:
Andrew D. Yu[a]
Laura M. McDowell-Boyer[a]
James R. Cook[b]
Karen E. Young[c]

[a] Alara Environmental Analysis, Inc.
[b] Westinghouse Savannah River Company
[c] Exploration Resources, Inc.

Westinghouse Savannah River Company, LLC
Savannah River Site
Aiken, SC 29808

Prepared for the U.S. Department of Energy under Contract No. DE-AC09-96SR18500

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

EXECUTIVE SUMMARY

This report presents the results of a special study of the Naval Reactor Waste Disposal Pad located within the boundary of the E-Area Low-Level Waste Facility at the Savannah River Site. This study is a follow-up to the assessment of performance for this facility which was conducted for Revision 1 of the E-Area Low-Level Waste Facility Radiological Performance Assessment (McDowell-Boyer et al. 2000). What distinguishes the present study from earlier analyses is that the Waste Disposal Pad is considered to be built at-grade in this study (i.e., the as-built condition), which places the base of the facility at least 18.3 m above the water table. The previous analyses assumed the base of the pad at approximately 7.6 m above the water table to simplify the analysis. Another distinguishing feature of the present study is that release of radionuclides from the waste is tracked according to whether it is originally associated with corrosion products (termed "crud") in the waste, or activated metal; whereas, the previous study neglected the smaller contribution from the "crud" waste.

Because this study is essentially a reanalysis of work that was done for a performance assessment, it was prepared in consideration of the performance assessment requirements of Chapter IV of the United States Department of Energy Order 435.1. The Order specifies that a performance assessment should provide reasonable assurance that a low-level waste disposal facility will comply with the performance objectives of the Order. It further requires, for purposes of establishing limits on radionuclides that may be disposed near-surface, assessments of impacts to water resources and to hypothetical inadvertent intruders.

The Naval Reactor Waste Disposal Pad is an area covering approximately 11,000 m^2, located in the northeast corner of the E-Area Low-Level Waste Facility. Waste to be disposed in the Waste Disposal Pad consists of a variety of solid activated metal Naval nuclear reactor components. While there is no standard waste disposal container for the Naval Reactor waste, nominal wall thicknesses of the water-tight corrosion-resistant steel containers will be 35 cm, with nominal weld thicknesses on the order of 4 cm. The long-term performance of the engineered features of the Naval Reactor Waste Disposal Pad within E Area, is key to estimating the transport of radionuclides into the environment, and thus evaluating the performance in terms of performance objectives and requirements of Order 435.1. Therefore, degradation of these features, including moisture barriers, waste containers, and waste forms, was addressed in this analysis.

To evaluate the long-term performance of the Naval Reactor Waste Disposal Pad, site-specific conceptual models were developed to consider: 1) exposure pathways and scenarios of potential importance; 2) potential releases from the facility to the environment; 3) effects of degradation of engineered features, and 4) transport in the environment to a designated point of compliance. For evaluation of doses to off-site members of the public and water resource impacts, the point of compliance is assumed to be the point of highest concentration in groundwater or air more than 100 m from the disposed waste. For evaluation of doses to inadvertent intruders, the point of compliance is at the point of highest concentration of radionuclides after a 100-year institutional control period following closure of the facility.

The results of this study were used to calculate allowable radionuclide concentrations and inventories in the Naval Reactor Waste Disposal Pad. Allowable inventory limits were calculated by comparison of estimated groundwater concentration or off-site and intruder doses with limits on groundwater concentration or doses as set forth in the performance objectives and requirements. The calculated inventory limits were then compared with the projected radionuclide inventory for the Naval Reactor Waste Disposal Pad, to evaluate whether reasonable assurance is provided that the performance

objectives and requirements will be met. The calculated inventory limits will be used to revise waste acceptance criteria for the E-Area Low-Level Waste Facility.

The calculated inventory limits for the Naval Reactor Waste Disposal Pad will allow disposal of the projected inventory for this facility, according to the analyses conducted. For most radionuclides, calculated inventory limits are more than two orders of magnitude greater than the projected inventories. The favorable results of this study can be attributed in large part to the considerable time calculated for corrosion of the waste disposal casks, such that intrusion into the material inside the casks is not deemed credible within the 10,000-year time frame of this assessment. Protection of water resources can be attributed to the slow release of radionuclides from the metal matrix of the waste. Thus, any factors that change the estimated corrosion times of either the disposal casks or steel waste form may change the results of this analysis. A discussion of critical assumptions in this analysis is provided at the conclusion of this report.

TABLE OF CONTENTS

TABLE OF CONTENTS (cont.)

LIST OF FIGURES

LIST OF TABLES

LIST OF TABLES (cont.)

LIST OF APPENDICES

LIST OF ACRONYMS AND ABBREVIATIONS

ACRONYMS

ACRI	Analytic and Computational Research, Inc.
ARAR	applicable or relevant and appropriate requirement
CB/TS	core barrels/thermal shields
CERCLA	Comprehensive Environmental Response, Compensation, and Liability Act
CFR	Code of Federal Regulations
DCF	dose conversion factor
EDE	effective dose equivalent
ICRP	International Commission on Radiological Protection
KAPL	Knolls Atomic Power Laboratory
LLWF	low-level waste facility
MCL	maximum contaminant level
NBS	National Bureau of Standards
NCRP	National Council on Radiation Protection and Measurements
NPL	National Priorities List
NRC	Nuclear Regulatory Commission
PA	Performance Assessment
SAR	safety analysis report
SDCF	scenario dose conversion factor
SRS	Savannah River Site
SRTC	Savannah River Technology Center
USDOE	United States Department of Energy
USEPA	United States Environmental Protection Agency
WAC	waste acceptance criteria
WITS	Waste Information Tracking System
WSRC	Westinghouse Savannah River Company

ABBREVIATIONS

Bq	Becquerel
Ci	curie
cm	centimeter
d	day
ft	foot
g	gram
gal	gallon
hr	hour
in	inch
Kd	sorption coefficient
kg	kilogram
km	kilometer
L	liter
m	meter
μCi	microcurie
μg	microgram
mL	milliliter
mrem	millirem

LIST OF ACRONYMS AND ABBREVIATIONS (cont.)

ABBREVIATIONS (cont.)

nCi	nanocurie
pCi	picocurie
rem	roentgen equivalent man
s	second
Sv	Sievert
yr	yea

0.0 REVISION HISTORY

Revision 2 of this document was prepared to address inconsistencies in Tables 4.1-2 and 6.1-1 and to incorporate clarification of disposal requirements based on questions raised subsequent to approval of revision 1.. The inconsistencies were the result of differences in rounding methodology for some changes, and one change made following the design check to correct the MCL for U-235. Requirements were clarified for cask and wasteform material and thickness, radiation rate and radionuclide content.

1.0 INTRODUCTION

The present study is a reanalysis of an assessment of performance that was done for the Naval Reactor Waste Disposal Pad located within the boundary of the E-Area Low-level Waste Facility (LLWF) at the Savannah River Site (SRS), due to changes in the representation of the vertical placement of the pad. The assessment being reanalyzed was part of the first and most recent revision of the radiological Performance Assessment (PA) for the entire E-Area LLWF, which is reported in Westinghouse Savannah River Company (WSRC) WSRC-RP-94-218 (McDowell-Boyer et al. 2000). The original PA for E Area was issued in April 1994 and received conditional United States Department of Energy (USDOE) approval in September 1994. The report herein is supplemental to the Revision 1 PA.

In order that the context of the present report is understood, information pertinent to the PA in general, and more specifically to the Naval Reactor Waste Disposal Pad, is reviewed here. A description of the performance criteria to which results of the present analysis are compared is presented in Section 1.3. Much of this information is taken from Section 2 of the PA revision (McDowell-Boyer et al. 2000), and the reader is referred to that document for a more complete discussion of the information presented here. Throughout this report, there are references to the PA revision by section to facilitate locating pertinent information in the reference document.

1.1 Approach to Performance Assessment

The E-Area LLWF PA was developed using the USDOE's requirements and guidance for the PA specified in USDOE Order 435.1. Because the present work supplements the PA, these requirements and guidance have also been used here.

In order to focus on radionuclides and pathways to exposure of greatest significance in the assessment of dose and inventory limits, screening methodologies were employed in the same manner as in the PA. Screening methods and results are discussed in this report, and the reader is referred back to the PA for more in-depth discussion of the methods.

The level of technical detail presented in this report is sufficient to allow a reviewer to reproduce the results of the PA calculations. Input data to models are provided, and sample input files for numerical simulation are provided in the appendices. Whenever possible, supporting information such as meteorological data and environmental information, is provided in general terms to meet the needs of this document with references provided for access to additional details.

1.2 General Background on Naval Reactor Waste

Naval reactor waste will be disposed in the northeastern-most section of the E-Area LLWF (Figure 1.2-1). The E-Area LLWF is the site for low-level radioactive waste disposal at the SRS. The E-Area LLWF has been designed to manage all low-level waste resulting from SRS operations for the next 20 years. The facility for disposing of Naval Reactor waste is termed the Naval Reactor Waste Disposal Pad.

The waste destined for this facility will be disposed in heavily shielded casks, which are placed on an at-grade pad. Up to 100 casks will be accepted for disposal in the pad, which covers approximately 11,000-m^2 area. The waste within the casks consists of large pieces of activated metal Naval Reactor components, including core barrels, adapter flanges, closure heads, and other similar equipment.

This facility will continue to operate until the capacity is reached, either by number of casks or allowable radionuclide inventory, or until the E-Area LLWF is closed. It is possible that shipments of Naval Reactor waste will cease before capacity is reached. A closure plan has been prepared for the E-Area LLWF (Cook et al. 2000).

1.3 Performance Criteria

The specific performance objectives for solid waste disposal in E Area are contained in USDOE Order 435.1 (USDOE 1999):

Performance Objectives. Low-level waste disposal facilities shall be sited, designed, operated, maintained, and closed so that a reasonable expectation exists that the following performance objectives will be met for waste disposed after September 26, 1988:

(a) Dose to representative members of the public shall not exceed 25 mrem (0.25 Sv) per year total effective dose equivalent (EDE) from all exposure pathways, excluding the dose from radon and its progeny in air.

(b) Dose to representative members of the public via the air pathway shall not exceed 10 mrem (0.10 Sv) per year total EDE, excluding the dose from radon and its progeny.

(c) Release of radon shall be less than an average flux of 20 pCi/m^2/s (0.74 Bq/m^2/s) at the surface of the disposal facility. Alternatively, a limit of 0.5 pCi/L (0.0185 Bq/L) of air may be applied at the boundary of the facility.

In addition to the performance objectives, the Order requires, for purposes of establishing limits on the concentrations of radionuclides that may be disposed near-surface, an assessment of impacts to water resources and to hypothetical persons assumed to inadvertently intrude for a temporary period into the low-level waste disposal facility.

Table 1.3-1 lays out the performance measures and the associated points of compliance. In this analysis, and in the E-Area LLWF PA, the point of assessment for protection of the public and for the assessment of impacts to water resources is taken to be that location more than 100 m from any disposed waste at which the projected dose or projected concentrations of contaminants in groundwater are the highest. Although this point of assessment is conservative, given the SRS land-use plan that foresees no unrestricted public access, it is consistent with USDOE 435.1.

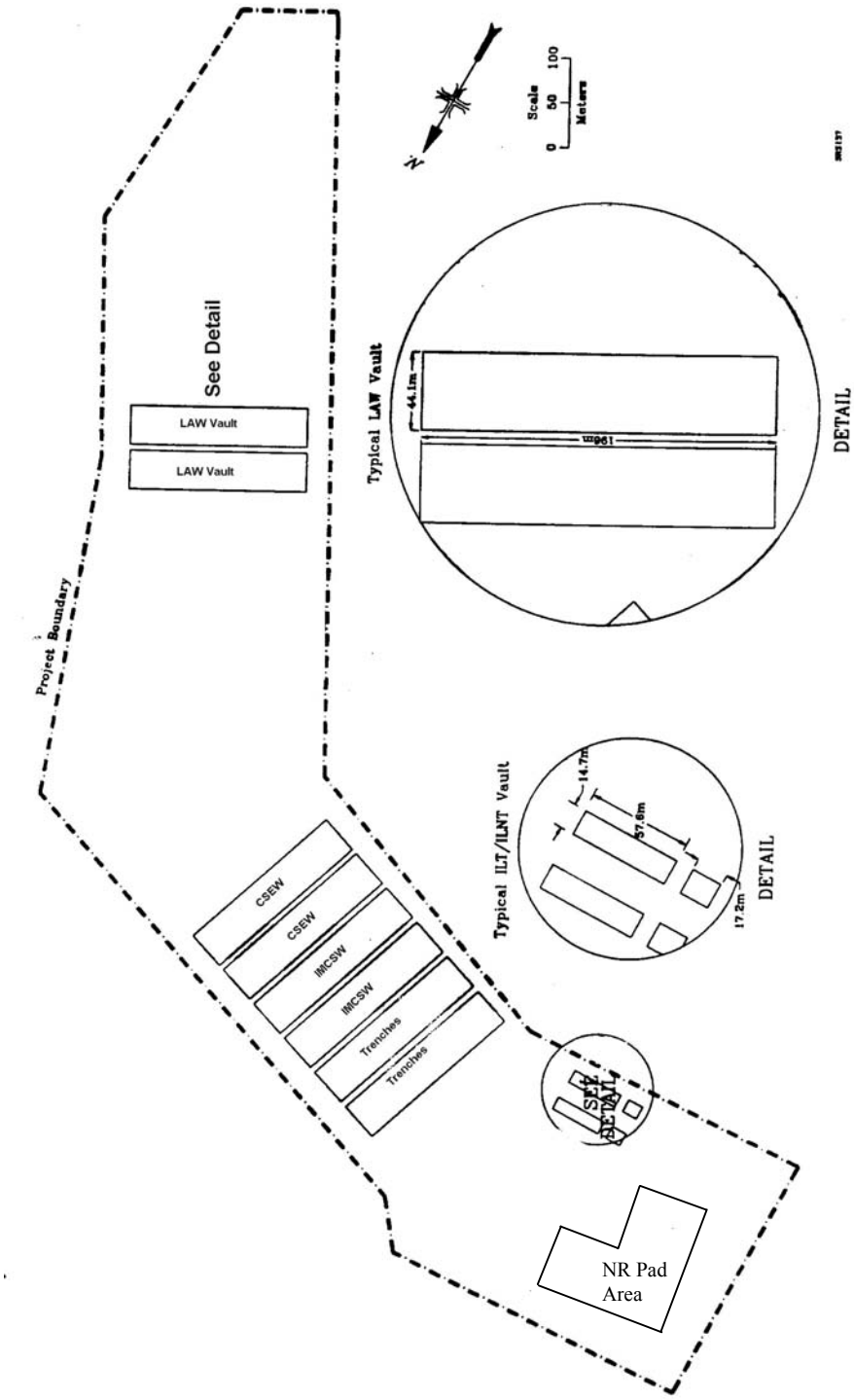

Figure 1.2-1 Location of Naval Reactor Components Pad

Table 1.3-1. Performance Objectives, Assessment Requirements, and Points of Compliance

Performance Objective	Measure	Point of Compliance
All pathways	≤ 25 mrem in a year, not including doses from radon and progeny	Point of highest projected dose or concentration beyond a 100-m buffer zone surrounding the disposed waste
Air pathway	≤ 10 mrem in a year, not including doses from radon and progeny	Point of highest projected dose or concentration beyond a 100-m buffer zone surrounding the disposed waste
Radon	either (1)an average flux of ≤ 20 pCi/m^2/s, or	Disposal facility surface
	(2)an air concentration of ≤ 0.5 pCi/L	Point of highest projected dose or concentration beyond a 100-m buffer zone surrounding the disposed waste
Assessment Requirement	**Measure**	**Point of Compliance**
Hypothetical inadvertent intruder	100 mrem in a year from chronic exposure	Disposal facility
	500 mrem from a single event	Disposal facility
Impact on water resources	The SRS interpretation is that concentrations of radioactive contaminants should not exceed standards for public drinking water supplies established by the USEPA (40 CFR Part 141).	Point of highest projected dose or concentration beyond a 100-m buffer zone surrounding the disposed waste

USDOE Order 435.1 states that "The performance assessment shall include calculations for a 1,000-year period after closure of potential doses to representative future members of the public and potential releases from the facility to provide a reasonable expectation that the performance objectives identified in this Chapter are not exceeded as a result of operation and closure of the facility."

However, a more conservative approach than that required by USDOE Order 435.1 has been taken in this analysis with respect to the time period for compliance with the performance objectives. Consistent with the approach taken in the E-Area LLWF PA, the performance objectives for protection of off-site members of the public, and the performance measures for the analysis of impacts on inadvertent intruders and water resources are applied for 10,000 years after disposal. The 10,000-year period was selected to be consistent with PA guidance from the Nuclear Regulatory Commission (NRC) (NRC 1997).

Compliance with the requirement to assess impacts on resources is interpreted at the SRS as meaning that concentrations of radioactive contaminants at any points of compliance should not exceed standards for public drinking water supplies established by the United States Environmental Protection Agency (USEPA) (40 Code of Federal Regulations [CFR] Part 141).

Specific interpretation and application of the performance objectives listed above are further discussed below with respect to protection of the public, water resources, and inadvertent intruders.

1.3.1 Public Protection Performance Objective

Protection of the public according to the stated performance objectives requires that calculated annual dose to a hypothetical future member of the public shall not exceed 25 mrem total EDE from all exposure pathways. "All exposure pathways" includes all modes of exposure, including the air pathway, but excluding exposures to radon and short-lived progeny. Furthermore, calculated dose via the air pathway is not to exceed 10 mrem/yr total EDE, again excluding dose from radon and short-lived progeny. The point of compliance, as stated earlier, is the point of highest calculated dose beyond a 100-m buffer zone surrounding the waste.

Radon is addressed separately, with separate applicable limits. In most cases, the limit for radon should be an average ground surface emanation rate of 20 $pCi/m^2/s$, which applies in the E-Area PA. (An alternative limit may apply in special cases, which involve disposal of material that radiologically resembles uranium or thorium mill tailings, in which case an incremental increase in the air concentration of radon of 0.5 pCi/L at the point of public access [i.e., beyond a 100-m buffer zone surrounding the disposed waste] should be applied.)

1.3.2 Water Resource Impact Assessment Requirement

USDOE Order 435.1 does not specify either dose or concentration limits for radionuclides in water. Therefore, there is some ambiguity in applying the requirement even though, as described previously, the performance measure is interpreted as requiring that concentrations of contaminants in groundwater should not exceed values specified in USEPA standards for public drinking water supplies (40 CFR Part 141).

The SRS is one of the USDOE sites designated as being on the National Priorities List (NPL) by the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) (40 CFR 300). As a result, all groundwater at SRS is regulated by CERCLA. Under CERCLA, the maximum

contaminant levels (MCLs) promulgated under the Safe Drinking Water Act (40 CFR 141) are used as applicable or relevant and appropriate requirements (ARARs).

In the PA for the E-Area LLWF, three different options for specifying MCLs of radionuclides in groundwater were considered. In this analysis, the Primary Drinking Water Standards for radionuclides, promulgated on December 7, 2000, are used (USEPA 2000). The current 4 mrem/yr standard for beta and/or photon emitters in drinking water requires that MCLs be developed based on internal dosimetry data from International Commission on Radiological Protection (ICRP) Publication 2 (1959) and specified MCLs for ^{3}H and ^{90}Sr. National Bureau of Standards (NBS) Handbook 69 (U.S. Department of Commerce 1963) is used to derive MCLs. A listing of the resulting MCLs is available in the Implementation Guidance for Radionuclides (USEPA 2001). The WSRC Legal Department has determined that use of the MCLs is appropriate (Gough, 2002). There are several radionuclides in the present analysis for which MCLs are not available in this listing. For these radionuclides (^{79}Se, ^{93}Mo, ^{94}Nb, ^{126}Sn, ^{210}Pb, ^{225}Ra, ^{234}Th, and ^{242m}Am), an MCL is derived assuming a limit of 4 mrem/yr EDE and internal dosimetry based on ICRP Publication 30 (1979). This method is consistent with that used in the original PA for E Area and the PA revision (McDowell-Boyer et al. 2000).

1.3.3 Intruder Analysis Requirement

The assessment of impacts to hypothetical persons who are assumed to inadvertently intrude into the Naval Reactor Waste Disposal Pad requires that calculated annual total EDE to such individuals not exceed 100 mrem for chronic exposure scenarios. For acute exposure scenarios, calculated doses are not to exceed 500 mrem total EDE. Institutional controls are assumed to be effective in deterring intrusion for at least 100 years following closure of the facility. Passive controls, in the form of engineered barriers or features of the site, can be claimed as further deterrents to intrusion.

In general, the chronic exposure scenarios address all reasonable pathways. Consistent with USDOE guidance (USDOE 1996), consumption of groundwater and crop irrigation are exposure pathways that are excluded from the intruder analysis; impacts of groundwater contamination are evaluated separately in this analysis and the E-Area LLWF PA in the all-pathways analysis and the water resource protection analysis. Doses from short-lived radon progeny are not addressed for similar reasons.

2.0 THE NAVAL REACTOR WASTE DISPOSAL PAD

2.1 Major Design Philosophy

Specific design features of the Naval Reactor Waste Disposal Pad will enhance safe operation and disposal. First, the waste itself is contained in heavily-shielded steel disposal casks, thus protecting operations personnel from unacceptable radiation exposures. Second, the casks are designed to protect against the entrance or loss of water from the casks, such that releases of radionuclides before closure of E Area is unlikely. Third, the at-grade elevation of the gravel pads is such that the distance to the water table is at least 20 m, which increases the time of transport of any radionuclides released from the casks to the water table, thus diminishing resultant groundwater concentrations.

2.2 Facility Features

2.2.1 Layout of Facility

The Naval Reactor Waste Disposal Pad, an area covering approximately 11,000 m^2, provides capacity for 100 disposal casks. The details of cask placement in relation to other casks has not presently been determined, but a plan will evolve as Naval Reactor waste containers are received. After placement of all casks, infill between the casks will be done to stabilize the location of each cask, and sufficient backfill will be added over the top of the casks, to support the final cover and bring the final grade to at least 1.2 m above the top of the casks.

The Naval Reactor Waste Disposal Pad is located within the E-Area LLWF boundary, and therefore final closure of the pad will be similar to that for the entire facility. Briefly, an additional 0.9 m of backfill from the Burma Road borrow pit is assumed to be placed over the new grade, above which a laterally extensive moisture barrier is assumed to be placed. The moisture barrier consists of 0.76 m of clay overlain by 0.3 m of gravel and a geotextile fabric. Over this moisture barrier, 0.76 m of backfill is assumed, followed by 0.15 m of topsoil. As a result, a minimum of 4.1 m of backfill and cover material is assumed to overlie the Naval Reactor casks at closure.

2.2.2 Water Infiltration and Cover Integrity

Although normal infiltration in the vicinity of E Area is approximately 40 cm/yr (Section C.1.1, McDowell-Boyer et al. 2000), the presence of the cover, while intact, will considerably diminish the amount of water traveling through the Naval Reactor Waste Disposal Pad.

The slope on the final closure cover will be no more than 5%. This slope will allow runoff of water intercepted by the moisture barrier, such that infiltration into the Naval Reactor waste is limited. Vegetation and drainage ditches will be used together to stabilize soil against erosion, prevent pine tree growth, and divert excess water from the gravel layer of the closure cover.

2.2.3 Structural Stability and Inadvertent Intruder Barrier

Strength requirements of the Naval Reactor waste containers are such that they must remain intact while transporting Naval Reactor components, which have a density of 7,700-8,000 kg/m^3. Thus, it is expected that these containers will readily support a soil/clay overburden with a density of 960-1,600 kg/m^3. Container design will also deter water infiltration for a considerable period of time after institutional control has ceased.

A physical description of the waste form and disposal containers is given in the next section. The nature of the containers and waste form are such that intrusion by inadvertent intruders is not likely for a very long period of time. The steel containers and waste form are sufficiently thick that average corrosion would take thousands of years to allow contact with the radionuclides, which largely exist as activation products incorporated in the metal matrix of the waste components.

2.3 Physical Description of Waste and Containers

2.3.1 Waste Forms

Naval reactor waste consists of a variety of solid activated metal Naval nuclear reactor components, including core barrels/thermal shields (CB/TS), adapter flanges, closure heads, holddown barrels, pumps and other similar equipment. Activation product radionuclides are intimately dispersed in the metal matrix, and may be released only when the metal is corroded. Certain components are also covered with a thin layer of adherent corrosion products, referred to as "crud", which is rich in radionuclide contamination.

The Naval Reactor program has identified several types of Naval Reactor waste components that have been or are planned to be delivered to SRS for disposal. These waste components include CB/TS, holddown barrels, heads, adapter flanges, shrouds, and pumps. Volumes of the metal waste components range between 1.05 and 7.05 m^3 for each component. Most waste components also contain some water with the maximum amount averaging approximately 9.5×10^{-3} m^3. More detailed configurational descriptions of the Naval Reactor waste components are not available because of the classified nature of this information.

According to data supplied by the Naval Reactor program, a representative type of activated waste component is the Knolls Atomic Power Laboratory (KAPL) CB/TS unit. The six subcomponent parts of this unit are made of either Inconel or Zircaloy, both corrosion-resistant metal alloys. Because of the different constituents in Inconel and Zircaloy, activation product distributions between these two alloys are different, as are corrosion rates. A breakdown of fractional activity distribution between the two alloys for some radionuclides present in the activated waste components is shown in Table 2.3-1.

Table 2.3-1. Breakdown of "Activation" Inventories in Inconel and Zircaloy

Radionuclide	Fraction of Inventory in Inconel	Fraction of Inventory in Zircaloy
^{14}C	0.12	0.88
^{59}Ni	0.83	0.17
^{93}Zr	0.00	1.00
^{94}Nb	0.73	0.27
^{93}Mo	0.95	0.05
^{99}Tc	0.74	0.26
^{239}Pu	0.01	0.99
^{240}Pu	0.00	1.00

A representative type of Nuclear Reactor waste component with predominantly "crud" contamination is the KAPL Head unit. Composition and corrosion rates are not pertinent for these surface-contaminated components in this analysis, because release is not dependent on breakdown of the associated metal matrix. Of the 100 Naval Reactor waste casks that will be disposed in E Area, it is expected that approximately 50 will contain activated metal components, while the remaining 50 will contain surface-contaminated components.

2.3.2 Disposal Containers

There is no standard Naval Reactor waste container due to the variety of Naval Reactor waste components. The actual container configuration, thickness, material of construction and closure method may be tailored to the characteristics of the Naval Reactor waste component at the time of disposal. Planned or proposed containers for Naval Reactor waste disposal are mostly composed of a high-strength, corrosion-resistant steel and are closed by a weld. Many containers will also contain reinforced concrete providing additional shielding and structural integrity. Nominal thicknesses of steel walls and ends of the containers will be 35 cm. Nominal weld thicknesses will be on the order of 4 cm.

A simplified cross-section of a representative KAPL conceptual disposal container is illustrated in Figure 2.3-1. The interior volume of the shipping/disposal cask is approximately 23 m^3. The overall containerized waste volume is about 43 m^3.

2.4 Radiological Inventory of Waste

The Naval Reactor program provided radionuclide inventories of the two representative types of Naval Reactor waste components, the KAPL CB/TS and the KAPL Head units, shown in Table 2.4-1. The activity listed for each radionuclide in this table is considered a general upper bound case by KAPL personnel, suitable for use in the Special Analysis. A representative activity for the "average" component is about one-half the value listed in Table 2.4-1, according to Navy personnel. The listed curie inventories of 66 contaminants are further characterized as either "Activation" or "Crud" waste. Activation waste contains activation product radionuclides intimately dispersed in the metal matrix, which are released only as the metal is corroded. In contrast, crud waste resides in a thin surface coating and is more readily leached once containers are breached. Table 2.4-1 shows that the upper bound activity of a KAPL CB/TS cask is about 8×10^4 Ci of activation waste and 9 Ci of crud waste. The upper bound activity of a KAPL Head cask is about 2×10^{-2} Ci of activation waste and 40 Ci of crud waste. Thus, essentially all activation waste radionuclides reside in KAPL CB/TS casks. By contrast, the activity of crud waste radionuclides in KAPL Head casks is somewhat greater than the activity of crud-associated radionuclides in KAPL CB/TS casks.

Table 2.4-1 shows upper-bound activities per representative disposal cask. The Navy estimates that of the 100 casks destined for disposal at E Area, about one-half, or 50, are best represented by the CB/TS waste components, while the remaining 50 are best represented by the Head components. In order to project a representative total inventory of Naval Reactor waste, the upper-bound values in Table 2.4-1 are first halved to obtain average activities in the two representative types of waste. Next, these halved values are multiplied by 50 to obtain total inventories for each representative waste component (i.e., CB/TS or Head), since there are assumed to be 50 of each type of component. To obtain the total inventory of each radionuclide in "Activation" or "Crud" waste, the total inventories for each of the two types of representative waste components are then added together, keeping "Activation" and "Crud" waste categories separate. The final results of these operations are listed in Table 2.4-2. The inventory listed in Table 2.4-2 is the total projected inventory for the Naval Reactor waste components used in this assessment.

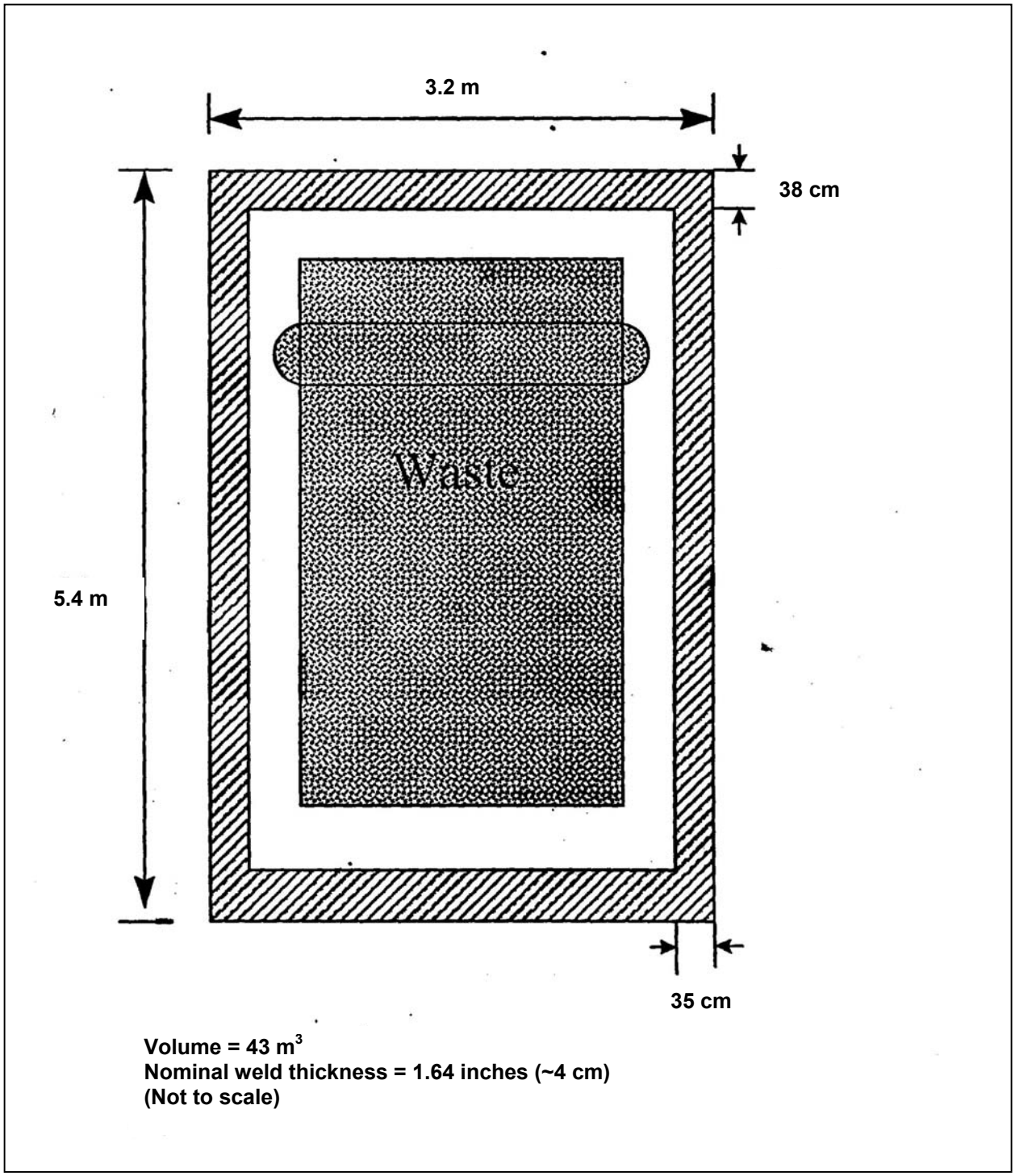

Figure 2.3-1. Conceptual Naval Reactor Waste Disposal Cask

Table 2.4-1. Radionuclide Inventory in Representative Naval Reactor Waste Component Casks to be Shipped to E Area

Radionuclide	KAPL CB/TS		KAPL Head	
	Activation [Curies]	Crud [Curies]	Activation [Curies]	Crud [Curies]
^{3}H	1.35×10^{1}	6.35×10^{-5}	2.02×10^{-5}	---
^{14}C	1.35×10^{1}	2.80×10^{-2}	7.92×10^{-8}	4.51×10^{-2}
^{51}Cr	7.49×10^{2}	8.16×10^{-4}	3.64×10^{-3}	1.13
^{54}Mn	1.37×10^{2}	1.54×10^{-1}	2.75×10^{-5}	4.51×10^{-1}
^{55}Fe	8.98×10^{3}	4.63	1.49×10^{-2}	9.02
^{59}Fe	7.49×10^{2}	1.05×10^{-2}	9.27×10^{-4}	1.13
^{58}Co	2.03×10^{3}	8.00×10^{-1}	3.09×10^{-5}	1.8×10^{1}
^{60}Co	9.82×10^{3}	2.54	1.34×10^{-3}	4.51
^{59}Ni	1.56×10^{2}	8.4×10^{-3}	1.22×10^{-6}	1.35×10^{-2}
^{63}Ni	1.8×10^{4}	8.36×10^{-1}	1.49×10^{-4}	1.35
^{79}Se	1.23×10^{-4}	4.20×10^{-9}	3.93×10^{-14}	6.77×10^{-9}
^{90}Sr	5.27×10^{-1}	1.10×10^{-3}	---	1.80×10^{-3}
^{90}Y	5.27×10^{-1}	1.10×10^{-3}	---	1.80×10^{-3}
^{93}Zr	7.49×10^{2}	5.59×10^{-6}	---	9.02×10^{-6}
^{95}Zr	6.18×10^{3}	1.51×10^{-2}	---	4.51×10^{-1}
^{93m}Nb	7.49×10^{2}	4.06×10^{-2}	---	6.76×10^{-2}
^{94}Nb	6.50×10^{-1}	5.60×10^{-4}	1.01×10^{-7}	9.02×10^{-4}
^{95}Nb	1.31×10^{4}	3.24×10^{-2}	---	9.7×10^{-1}
^{95m}Nb	1.31×10^{2}	1.28×10^{-4}	---	---
^{93}Mo	1.44×10^{-1}	---	---	---
^{99}Tc	1.43×10^{-2}	2.8×10^{-5}	7.04×10^{-9}	4.51×10^{-5}
^{106}Ru	4.20×10^{-2}	---	---	---
^{113m}In	4.89×10^{2}	---	---	---
^{113}Sn	4.89×10^{2}	---	---	---
^{119m}Sn	8.11×10^{3}	---	---	---
^{123}Sn	2.36×10^{2}	---	---	---
^{126}Sn	7.18×10^{-5}	1.26×10^{-6}	---	2.03×10^{-6}
^{125}Sb	4.08×10^{3}	2.32×10^{-2}	---	4.51×10^{-2}
^{125m}Te	2.55×10^{3}	6.45×10^{-3}	---	1.04×10^{-2}
^{129}I	8.50×10^{-8}	1.12×10^{-7}	---	1.80×10^{-7}
^{135}Cs	3.47×10^{-6}	---	---	---
^{137}Cs	5.17×10^{-1}	1.10×10^{-3}	---	1.80×10^{-3}

Table 2.4-1. Radionuclide Inventory in Representative Naval Reactor Waste Component Casks to be Shipped to E Area (cont.)

Radionuclide	KAPL CB/TS		KAPL Head	
	Activation [Curies]	Crud [Curies]	Activation [Curies]	Crud [Curies]
^{137m}Ba	5.17×10^{-1}	1.10×10^{-3}	---	1.80×10^{-3}
^{144}Ce	4.93×10^{-1}	---	---	---
^{147}Pm	2.98×10^{-1}	---	---	---
^{151}Sm	5.40×10^{-3}	---	---	---
^{154}Eu	6.75×10^{-3}	---	---	---
^{155}Eu	3.84×10^{-3}	---	---	---
^{181}Hf	7.49×10^{2}	3.39×10^{-3}	---	4.51×10^{-1}
^{182}Ta	1.77×10^{3}	---	2.55×10^{-3}	---
^{232}Th	2.20×10^{-16}	2.66×10^{-12}	---	4.29×10^{-12}
^{232}U	2.06×10^{-9}	4.15×10^{-8}	---	6.77×10^{-8}
^{234}U	2.78×10^{-5}	---	---	---
^{235}U	2.06×10^{-8}	---	---	---
^{236}U	4.22×10^{-5}	---	---	---
^{238}U	2.33×10^{-6}	---	---	---
^{237}Np	4.03×10^{-7}	8.40×10^{-10}	---	1.35×10^{-10}
^{238}Pu	2.69×10^{-2}	7.01×10^{-6}	---	1.13×10^{-5}
^{239}Pu	1.24×10^{-2}	1.12×10^{-6}	---	1.80×10^{-6}
^{240}Pu	1.11×10^{-2}	7.01×10^{-7}	---	1.13×10^{-6}
^{241}Pu	3.41	2.80×10^{-4}	---	4.51×10^{-4}
^{242}Pu	4.07×10^{-5}	8.40×10^{-9}	---	1.35×10^{-8}
^{244}Pu	2.77×10^{-12}	1.26×10^{-15}	---	2.03×10^{-15}
^{241}Am	3.52×10^{-2}	9.22×10^{-6}	---	1.58×10^{-5}
^{242m}Am	---	2.24×10^{-7}	---	9.02×10^{-8}
^{243}Am	2.41×10^{-4}	8.40×10^{-8}	---	1.35×10^{-7}
^{242}Cm	5.22×10^{-1}	5.78×10^{-5}	---	2.93×10^{-4}
^{243}Cm	---	7.00×10^{-8}	---	1.13×10^{-7}
^{244}Cm	1.92×10^{-2}	9.50×10^{-7}	---	1.58×10^{-5}
^{245}Cm	1.02×10^{-6}	7.01×10^{-10}	---	1.13×10^{-9}
^{246}Cm	3.93×10^{-7}	2.80×10^{-10}	---	4.51×10^{-10}
^{247}Cm	7.90×10^{-13}	8.40×10^{-16}	---	1.35×10^{-15}
^{248}Cm	1.86×10^{-12}	2.66×10^{-15}	---	4.29×10^{-15}
^{249}Cf	1.24×10^{-11}	1.40×10^{-14}	---	2.26×10^{-14}
^{251}Cf	2.64×10^{-13}	5.59×10^{-16}	---	9.02×10^{-16}
Total	8.00×10^{4}	9.13	2.36×10^{-2}	3.77×10^{1}

Table 2.4-2. Total Projected Inventory for Naval Reactor Waste (100 Disposal Casks)

Radionuclide	Total Activation [Curies]	Total Crud [Curies]
^{3}H	3.4×10^{2}	1.6×10^{-3}
^{14}C	3.4×10^{2}	1.8
^{51}Cr	1.9×10^{4}	2.8×10^{1}
^{54}Mn	3.4×10^{3}	1.5×10^{1}
^{55}Fe	2.2×10^{5}	3.4×10^{2}
^{59}Fe	1.9×10^{4}	2.9×10^{1}
^{58}Co	5.1×10^{4}	4.7×10^{2}
^{60}Co	2.5×10^{5}	1.8×10^{2}
^{59}Ni	3.9×10^{3}	5.5×10^{-1}
^{63}Ni	4.5×10^{5}	5.5×10^{1}
^{79}Se	3.1×10^{-3}	2.7×10^{-7}
^{90}Sr	1.3×10^{1}	7.3×10^{-2}
^{90}Y	1.3×10^{1}	7.3×10^{-2}
^{93}Zr	1.9×10^{4}	3.7×10^{-4}
^{95}Zr	1.5×10^{5}	1.2×10^{1}
^{93m}Nb	1.9×10^{4}	2.7
^{94}Nb	1.6×10^{1}	3.7×10^{-2}
^{95}Nb	3.3×10^{5}	2.5×10^{1}
^{95m}Nb	3.3×10^{3}	3.2×10^{-3}
^{93}Mo	3.6	---
^{99}Tc	3.6×10^{-1}	1.8×10^{-3}
^{106}Ru	1.1	---
^{113m}In	1.2×10^{4}	---
^{113}Sn	1.2×10^{4}	---
^{119m}Sn	2.0×10^{5}	---
^{123}Sn	5.9×10^{3}	---
^{126}Sn	1.8×10^{-3}	8.2×10^{-5}
^{125}Sb	1.0×10^{5}	1.7
^{125m}Te	6.4×10^{4}	4.2×10^{-1}
^{129}I	2.1×10^{-6}	7.3×10^{-6}
^{135}Cs	8.7×10^{-5}	---
^{137}Cs	1.3×10^{1}	7.3×10^{-2}

Table 2.4-2. Total Projected Inventory for Naval Reactor Waste (100 Disposal Casks)
(cont.)

Radionuclide	Total Activation [Curies]	Total Crud [Curies]
^{137m}Ba	1.3×10^{1}	7.3×10^{-2}
^{144}Ce	1.2×10^{1}	---
^{147}Pm	7.5	---
^{151}Sm	1.4×10^{-1}	---
^{154}Eu	1.7×10^{-1}	---
^{155}Eu	9.6×10^{-2}	---
^{181}Hf	1.9×10^{4}	1.1×10^{1}
^{182}Ta	4.4×10^{4}	---
^{232}Th	5.5×10^{-15}	1.7×10^{-10}
^{232}U	5.2×10^{-8}	2.7×10^{-6}
^{234}U	7.0×10^{-4}	---
^{235}U	5.2×10^{-7}	---
^{236}U	1.1×10^{-3}	---
^{238}U	5.8×10^{-5}	---
^{237}Np	1.0×10^{-5}	2.4×10^{-8}
^{238}Pu	6.7×10^{-1}	4.6×10^{-4}
^{239}Pu	3.1×10^{-1}	7.3×10^{-5}
^{240}Pu	2.8×10^{-1}	4.6×10^{-5}
^{241}Pu	8.5×10^{1}	1.8×10^{-2}
^{242}Pu	1.0×10^{-3}	5.5×10^{-7}
^{244}Pu	6.9×10^{-11}	8.2×10^{-14}
^{241}Am	8.8×10^{-1}	6.3×10^{-4}
^{242m}Am	---	7.9×10^{-6}
^{243}Am	6.0×10^{-3}	5.5×10^{-6}
^{242}Cm	1.3×10^{1}	8.8×10^{-3}
^{243}Cm	---	4.6×10^{-6}
^{244}Cm	4.8×10^{-1}	4.2×10^{-4}
^{245}Cm	2.6×10^{-5}	4.6×10^{-8}
^{246}Cm	9.8×10^{-6}	1.8×10^{-8}
^{247}Cm	2.0×10^{-11}	5.5×10^{-14}
^{248}Cm	4.7×10^{-11}	1.7×10^{-13}
^{249}Cf	3.1×10^{-10}	9.2×10^{-13}
^{251}Cf	6.6×10^{-12}	3.7×10^{-14}
Total	2.0×10^{6}	1.2×10^{3}

2.5 Waste Volume

Naval reactor core barrels and reactor components are to be disposed on at-grade gravel pads in the E-Area LLWF. The gravel pads have a total capacity of approximately 11,000 m^2. Up to 100 disposal containers may be disposed in the pad area. One hundred disposal containers, placed upright, would cover about 1000 m^2. One hundred disposal containers, placed on their sides, would cover about 1700 m^2. The metal volume of waste is approximately 3.5 m^3 per container, giving a maximum total waste volume of approximately 350 m^3 at capacity.

THIS PAGE INTENTIONALLY LEFT BLANK

3.0 ANALYSIS OF PERFORMANCE

The methods used to analyze the long-term performance of the Naval Reactor Waste Disposal Pad are described in this section. A conceptual model of the source term is described in Section 3.1, where potential mechanisms of loss of integrity of the casks and subsequent release of radionuclides are considered. The screening process used to identify potentially significant radionuclides in the Naval Reactor waste is also described in Section 3.1. In Section 3.2, the pathways to potential human exposure to radionuclides released from the Naval Reactor waste are considered. Finally, in Section 3.3, the methodology for quantifying release and transport of radionuclides from the Naval Reactor Waste Disposal Pad are described. Much of the information in this section derives directly from the E-Area LLWF Performance Assessment (McDowell-Boyer et al. 2000), and thus a summary of the relevant material is presented here.

3.1 Source Term

In this section, mechanisms and factors which affect the rate of potential release of radionuclides from the Naval Reactor Waste Disposal Pad to the environment are discussed. Screening techniques used to focus analyses on significant radionuclides and processes are also described, followed by a description of conceptual models for release from the waste forms and containers.

3.1.1 Mechanisms of Potential Radionuclide Release

The Naval Reactor Waste Disposal Pad is a subsurface facility, once closure is complete. The overlying soil and moisture barrier will extend approximately 4 m from the top of the disposal casks to the ground surface (Section 2.2.1). Release of radionuclides to the subsurface soil surrounding the waste units, as a result of corrosion of activated metal, diffusion, and convection processes, is probable. Direct release of radionuclides from the disposal casts to the overlying surface soil, air, or nearby surface water is highly unlikely, except for possible release of volatile radiological components of waste (i.e., tritium, ^{14}C, and radon). Uncovering the disposal casks by natural erosion processes during the first 10,000 years after disposal is not expected, nor is upward movement of radionuclides due to biointrusion, for reasons explained in Section 4.1.1 of the E-Area LLWF PA (McDowell-Boyer et al. 2000). Therefore, the source term analysis focuses on the mechanisms of release of radionuclides to the groundwater in the vadose, or unsaturated zone, and of volatile radionuclides to the air.

Release of radionuclides from Naval Reactor waste to the vadose zone is triggered by contact with water that has seeped into the waste disposal casks from the surrounding soil. The rate of water infiltration into the casks is a function of the infiltration rate of rainwater into the subsurface and the efficacy of engineered barriers that serve to divert water away from the waste in the disposal units.

Degradation of the infiltration barrier (clay/gravel drain system) is expected to occur by a number of natural processes. Potential processes include erosion, intrusion by plants and animals, external events such as settling or slumping or a seismic event. These processes will reduce the effectiveness of the cover in limiting the vertical moisture flux.

Release can only occur if radionuclides are present in corrosion products in the waste; radionuclides present as activation products in uncorroded metal are not available for release. Water can only contact the corrosion products when the disposal casks have degraded to a point that they are no longer watertight.

3.1.2 Screening of Radionuclide Inventory

Although a potentially large number of radionuclides may be disposed in the Naval Reactor Waste Disposal Pad (Table 2.4-1), a fraction of these radionuclides will contribute the majority of the dose. Shorter-lived and/or highly sorbing radionuclides may decay to insignificant levels before human exposures occur. In order to identify radionuclides that will be insignificant to the total potential dose associated with the Naval Reactor Waste Disposal Pad, and thereby focus the remaining analysis on the potentially significant radionuclides, the list of nuclides in the projected inventories was screened. The screening methodology addresses potential dose associated with releases of volatile radionuclides to air and water resource protection separately from dose to inadvertent intruders because pathways to exposure are different. The remainder of this discussion on screening is pertinent to water resource protection; screening of the projected inventory with respect to intruder doses is described in Section 5.

Many radionuclides decay significantly during transport in the subsurface, due to short half-lives or high sorption potential, such that allowable inventories greatly exceed any imaginable inventory in a disposal unit. Screening calculations, used to identify radionuclides that do not decay significantly during transport, were carried out on a large suite of 730 radionuclides that may be encountered during disposal operations. These radionuclides were selected because they represent all radionuclides having published dose conversion factors (DCFs) (USDOE 1988). A subset of these 730 radionuclides is listed in projected inventories for the Naval Reactor Waste Disposal Pad.

The two potentially volatile radionuclides in the Naval Reactor waste are ^{3}H and ^{14}C. Screening of the air pathway was not done; thus both of these radionuclides are included in the list of potentially significant radionuclides.

Screening of the groundwater pathway was accomplished with spreadsheet calculations, described in detail in Cook and Wilhite (1998), and summarized in Section 4.1.3.1.1 of the E-Area LLWF PA (McDowell-Boyer et al. 2000). The methodology described in the National Council on Radiation Protection and Measurements (NCRP) Report 123 (NCRP 1996), applicable to disposal of radionuclides in the ground, was followed. The depth of waste in the disposal units was assumed to be 4.7 m, consistent with Figure 2.3-1, and the area of waste in the disposal unit was conservatively assumed to be 1850 m^2. The at-grade facility will actually cover about 11,000 m^2, so the dilution of clean soil in between waste casks is not accounted for in the screening procedure.

The list of radionuclides in the Naval Reactor Waste Disposal Pad of potential significance with respect to air and water resource contamination, including those with potentially significant daughters noted by "+d", are as follows: ^{3}H, ^{14}C, ^{59}Ni, ^{79}Se, ^{90}Sr+d, ^{93}Mo, ^{93}Zr+d, ^{94}Nb, ^{99}Tc, ^{126}Sn+d, ^{129}I, ^{135}Cs, ^{232}Th+d, ^{232}U+d, ^{234}U+d, ^{235}U+d, ^{236}U, ^{237}Np, ^{238}Pu+d, ^{238}U+d, ^{239}Pu+d, ^{240}Pu+d, ^{241}Am+d, ^{241}Pu+d, ^{242}Pu+d, ^{243}Am+d, ^{244}Cm+d, ^{244}Pu, ^{245}Cm+d, ^{246}Cm, ^{247}Cm+d, ^{248}Cm+d, and ^{249}Cf+d. In a few cases, the radionuclide listed is not important due to a combination of its half-life, sorption characteristics, and/or DCF; however, the longer-lived daughters are potentially important. Radionuclides that fall into this category are ^{241}Pu, ^{241}Am, ^{244}Cm, and ^{249}Cf.

3.1.3 Conceptual Model of Release of Radionuclides to the Vadose Zone

The rate of release for any particular radionuclide to the subsurface will be a function of:

- the quantity of that radionuclide initially disposed (i.e., radionuclide inventory),
- the rate of water infiltration into the disposal facility,

- the corrosion rate of disposal casks,
- the corrosion rate of activated metal in the waste,
- the composition of infiltrating water contacting waste, and
- the physical form, sorptive, and diffusive behavior of the radionuclide (i.e., physical/chemical characteristics).

The quantities of radionuclides initially disposed in the Naval Reactor waste are given in Table 2.4-2. The other parameters listed above are discussed separately, as they relate to the conceptual model to simulate these processes affecting radionuclide release.

3.1.3.1 Rate of Water Infiltration into the Disposal Facility

The amount of water available for infiltration into the disposal facility is a function of the infiltration rate of rainwater into the subsurface and the efficiency of engineered barriers that serve to divert water away from the waste in the disposal units. The conceptual model of infiltration into the vicinity of the Naval Reactor disposal casks considers the average annual infiltration at the ground surface after closure, and the long-term integrity of the E-Area moisture barrier, described in the closure plan. A detailed discussion of how an average annual infiltration rate of 40 cm/yr was derived from past studies of infiltration at the SRS is provided in Appendix Section C.1 of the E-Area LLWF PA (McDowell-Boyer et al. 2000).

Long-term integrity of the engineered moisture barrier is discussed in Section 4.1.3.1.3 of the PA, and can be summarized as follows. Degradation of the infiltration barrier (clay/gravel drain system) is expected to occur by a number of natural processes, including erosion, intrusion by plants and animals, external events such as settling or slumping, or a seismic event.

According to the present closure concept (Cook et al. 2000), shallow-rooted bamboo will be planted on the disposal site and a system of drainage ditches will be constructed to handle surface runoff and divert infiltration. During the period of active institutional control, periodic site inspection would reveal any degradation of the overlying cover and drainage system and corrective actions would be taken.

Return of the SRS land to unrestricted use at the end of the active institutional control period may result in a usage conversion to agricultural practices, consistent with past and current land use in the SRS vicinity. As discussed in Section 3.1.4.1 of the E-Area LLWF PA, soil erosion rates for cropland in the vicinity of the SRS are on the order of 2 kg/m^2-yr, which are considerably higher than the applicable erosion rate with bamboo present. Based on the higher rate of erosion, the cover may be eroded down to the gravel layer in as little as 650 years after the bamboo vegetative layer is cleared. Erosion of the gravel layer is difficult to predict.

In this analysis, the moisture barrier is assumed to remain intact until failure of the Naval Reactor disposal casks, at 750 years. Assuming an earlier failure time for the moisture barrier will not affect the analysis of radionuclide release because water cannot contact the waste until the casks have failed. Subsidence is not expected before 750 years because the disposal casks themselves are expected to maintain their structural integrity for thousands of years due to the thickness and corrosion-resistance of the walls. The timing of the disposal cask failure is discussed in Section 3.1.3.2 below. At the time of moisture barrier failure, the cover is assumed to degrade to the properties of the top soil and backfill soil.

3.1.3.2 Corrosion Rate of Disposal Casks

The Naval Reactor waste casks are considered impervious to water until failure occurs. The earliest this will most likely occur is when the 4-cm thick welds fail, as the cask walls and ends are of a much thicker and more corrosion-resistant steel (Section 2.3.2). Using a corrosion rate of 4×10^{-3} cm/yr for carbon steel (Sullivan et al. 1988), the time for 75% of the weld to corrode would be 750 years, assuming the weld material corrodes at a rate similar to that of carbon steel. Before 750 years, the container is assumed to be intact and there is no contaminant release. After container breach, the container is assumed to provide no deterrent to leaching of the contents, and the waste is assumed to have the hydraulic properties of the top soil, which is more permeable than the surrounding native soil. The more permeable waste zone is assumed to occur due to the initial presence of voids within the disposal.

It has been shown that 3-cm thick welds, that would fail after about 500 years, would have very little effect on the result of this analysis, and would therefore be acceptable for disposal on the NR Pad (Cook 2002a).

The structural integrity of the disposal casks is assumed to be maintained much longer than 750 years after disposal. The minimum 35-cm thickness of corrosion-resistant steel should prevent crumbling of the casks for thousands of years. Using a corrosion rate of 4×10^{-3} cm/yr for carbon steel, which is more readily degraded than the more corrosion-resistant steel used for the disposal cask walls, a 35-cm wall can be assumed to be completely corroded through in 8750 years.

HY-80 metal, which is sometimes used for both casks and welding metal, has been shown to have a corrosion rate that is at least twice as low as that for generic carbon-steel (Dunn, 2002).

3.1.3.3 Corrosion Rate of Activated Metal Waste

Release of radionuclides from Naval Reactor waste will depend on the corrosion rates of the activated metal components in the waste, for radionuclides present in the activated metal matrix. In Section 2.3.1, it is noted that a representative type of activated waste component in the Naval Reactor waste is composed of subcomponents that are made of either Inconel or Zircaloy, which are corrosion-resistant metal alloys. Corrosion rates for these alloys are different. The corrosion rate for Inconel is conservatively estimated at 2.5×10^{-5} cm/yr, and for Zircaloy, 2.5×10^{-6} cm/yr. The corrosion rates are based on data from the Hanford site (Hanford 1993), and notification by the Naval Reactor program that Zircaloy corrosion rates in similar environments would be expected to be orders of magnitude below those identified for stainless steel and Inconel. Assuming that corrosion occurs on both sides of a metal plate, effective corrosion rates for Inconel and Zircaloy are estimated at 5×10^{-5} cm/yr and 5×10^{-6} cm/yr, respectively.

In Section 2.3.1, it is indicated that activation products are distributed unequally between the two types of alloys. From Table 2.3-1, for example, 88% of the ^{14}C activity in activated metal is estimated to be associated with Zircaloy, while the remaining 12% is in Inconel. Likewise, most of the ^{93}Mo (95%) can be found in Inconel, with only 5% in Zircaloy. Thus, the conceptual model of release of radionuclides from activated metals must account for this distribution. Because Zircaloy has a lower corrosion rate than that of Inconel, radionuclides associated with this alloy will be released at a slower rate.

Table 3.1-1. Characteristics and Corrosion Rates of the Naval Reactor Waste Component Parts

Component Part	Inventory Fraction	T_{actual} (in)	α	T_{eff} (in)	Fraction Corroded, C_f (yr^{-1})	Time till 100% Corroded (yr)
1 (Inconel)	0.6467	0.750	1.38	0.543	3.68×10^{-5}	2.72×10^{4}
2 (Inconel)	0.1617	0.095	1.00	0.095	2.11×10^{-4}	4.74×10^{3}
3 (Inconel)	0.1557	1.500	2.10	0.714	2.80×10^{-5}	3.57×10^{4}
4 (Inconel)	0.0359	2.250	3.10	0.726	2.76×10^{-5}	3.62×10^{4}
Total	**1.000**					
5 (Zircaloy)	0.4496	0.375	1.00	0.375	5.33×10^{-6}	1.88×10^{5}
6 (Zircaloy)	0.5504	0.150	1.00	0.150	1.33×10^{-5}	7.52×10^{4}
Total	**1.000**					

A further breakdown of the activity distribution of activated metal has been provided by the Naval Reactor program. For the typical waste component that is predominantly activated waste (KAPL CB/TS, Section 2.3.1), there are six component parts, four of which are made of Inconel, and two of which are made of Zircaloy. Inventory fractions are identified for these six components (Table 3.1-1) which describe the fraction of a radionuclide's activation inventory associated with each type of component part in each type of metal. For example, 64.67 percent of the ^{14}C associated with Inconel components (which is 12% of the total ^{14}C in activated waste) is associated with component one. The purpose of this additional breakdown is to allow consideration of the different inventories associated with component parts with different thicknesses and radionuclide spatial distributions.

The peaking factor, α, in Table 3.1-1, is defined as the ratio of peak flux at the surface to the average flux across the thickness of the metal component, and has a minimum value of 1. The factor t_{actual} is the actual thickness of the component.

Effective corrosion rates for each radionuclide in each of the six component parts may be calculated by combining the information on activity distribution between the two types of alloys (Table 2.3-1) and the information on distribution between the six different component parts. First, the fraction of a given component part corroded per year can be calculated by dividing the effective corrosion rate (5×10^{-5} cm/yr for Inconel and 5×10^{-6} cm/yr for Zircaloy) by the effective thickness of that part (Table 3.1-1). The inverse of the fractional corrosion rate is also reported in Table 3.1-1 as the time until 100% of the part is corroded.

The rate of release R of a given radionuclide from activated metal at time t, can be calculated from:

$$R(t) = I_A\left(f_I \sum_{i=1}^{i=4} I_f C_f + f_Z \sum_{i=5}^{i=6} I_f C_f \right) e^{-t \ln 2 / t_{1/2}} . \qquad [3\text{-}1]$$

The symbol I_A refers to the total activated metal inventory of the radionuclide, f_I is the fraction of I_A in Inconel, f_Z is the fraction of I_A in Zircaloy, I_f is the fraction of the radionuclide in the component part of interest, and C_f is the fraction of the component part corroded each year. Component part indices i = 1 to 4 are for Inconel; i = 5 and 6 are for Zircaloy. Half-life for the radionuclide is represented as $t_{1/2}$. The exponential term accounts for radioactive decay. Radionuclides in the projected inventory of the Naval Reactor waste (Table 2.4-2) for which fractional distribution between Inconel and Zircaloy (f_I and f_Z) are not provided in Table 2.3-1, are assumed to be associated with Inconel only. This assumption provides a conservative (i.e., potential overestimate) of release of these radionuclides, because corrosion characteristics of Inconel are such that release rates are higher than for Zircaloy.

3.1.3.4 Composition of Infiltrating Water Contacting the Waste

The composition of water infiltrating the Naval Reactor Waste Disposal Pad will potentially affect the solubility or sorptive characteristics of radionuclides in the wastes. Water that has infiltrated disposal casks containing cementitious waste forms will have a composition that reflects the interaction of concrete pore water with vadose zone water. The presence of CO_2 gas in the soil and calcite present as a weathering product in the cement will buffer the pH of the cask water to between 7 and 8. Water that has entered casks without cementitious components will have a composition more similar to that of vadose zone water. Corrosion products of metals that are present in the disposal units, arising either from waste containers or surface contaminated wastes, may also be present in the water contacting wastes.

3.1.3.5 Physical /Chemical Characteristics

The physical and chemical form of radionuclides present in the crud or released from the activated metal disposed in the Naval Reactor Waste Disposal Pad will vary according to the conditions in the casks which may cause a change in form over time. Waste records are not sufficiently complete to accurately assign a form to each radionuclide in all units at the time of disposal, and considerable uncertainty exists regarding conditions in the facility over time that may cause a change of form. Therefore, assumptions regarding chemical and physical form are based on limited available information derived from consideration of the probable water composition in the waste disposal units, discussed above. Conservative assumptions are generally made regarding physical and chemical form that tend to overestimate the mobility of radionuclides from the disposal units, and thus overestimate the concentrations in groundwater.

Radionuclides present in corrosion products are assumed to be leached according to laws governing solubility and sorption. The corrosion products are represented as porous media, thus maximizing the surface area of the waste potentially exposed to infiltrating water. For the analysis pertinent to water resource protection, radionuclides that have volatile forms (e.g., ^{3}H and ^{14}C) are assumed to be completely soluble in water, despite the fact that a separate analysis is done assuming a fraction of these radionuclides are volatile.

Radionuclides in the Naval Reactor waste that are available for leaching are most likely sorbed onto corrosion products of steel. Geochemical calculations of sorption coefficients (Kds) for many of the radionuclides in waste destined for the E-Area Low Activity Waste vaults are described in Appendix E of the E-Area LLWF PA. These coefficients are also applied to the Naval Reactor waste, as it is expected that there will be similarities in form of radionuclides due to the similar presence of corrosion products of steel.

The Kds assumed in the analyses of release from corrosion products present in the Naval Reactor waste are listed in Table 3.1-2. Selection of Kds was made according to the rationale described in the E-Area LLWF PA (Section 4.1.3.1.4).

The rate of diffusion is proportional to the concentration gradient and diffusivity. For a given porous material, it is reasonable to assume that apparent diffusion coefficients (molecular diffusion coefficients corrected for tortuosity of the porous medium) are similar for all radionuclides because molecular diffusivities in water do not vary significantly. Apparent diffusion coefficients will vary, however, between porous materials due to differences in tortuosity. Tortuosity cannot be measured directly, but apparent diffusion coefficients have been obtained empirically for conservative (nonsorbing, nonreactive, nondecaying) compounds. In this study, a diffusivity of 158 cm^2/yr (5×10^{-6} cm^2/s) is assumed for all materials in the vadose zone, including the degraded cap. This value is a typical value for soils (Freeze and Cherry 1979). Calculation of diffusional release is part of the mass transport simulations described in Section 3.3 below.

3.1.4 Conceptual Model of Release of Volatile Radionuclides

The rate of release of volatile radionuclides from the Naval Reactor waste is a function of the physical/chemical form of each radionuclide in the waste form, the rate of release from the container, and the diffusion rate of the volatile element or compound through the overlying porous soil and moisture barrier. As was done in the E-Area LLWF PA (McDowell-Boyer et al. 2000), a simplified approach is taken to estimate the upper-bound release rate for the two volatile elements identified in the Naval Reactor waste, ^{3}H and ^{14}C. The only difference between the estimates in the PA and in the

Table 3.1-2. Elemental Sorption Coefficients for Naval Reactor Waste Radionuclides

Kd (mL/g)				
Element	**Soil[a,b]**	**Corrosion Waste[c,d]**	**Gravel[a]**	**Clay[e]**
Ac	450	450[l]	450	2400
Am	1900	3700	1900	8400
C	2[f]	2	2[f]	1
Cf	510[g]	3700	510[g]	8400[m]
Cm	4000	3700	4000	6000
Cs	330[h]	330	330[h]	1900
H	0[g]	0	0[g]	0[l]
I	0.6[i]	0.6	0.6[i]	1
Mo	10	10[l]	10	90
Nb	160	160[l]	160	900
Ni	400	1200	400	650
Np	5	750	5	55
Pa	550	550[l]	550	2700
Pb	270	270[l]	270	550
Po	150	150[l]	150	3000
Pu	100[h]	2000	100[h]	5100
Ra	500	60	500	9100
Se	5[j]	170	5[j]	740
Sn	130	55	130	670
Sr	10[h]	3	10[h]	110
Tc	0.36[k]	0.36	0.36[k]	1
Th	3200	2200	3200	5800
U	35	6000	35	1600
Zr	600	600[l]	600	3300

[a] Values are for sand from Sheppard and Thibault (1990), unless otherwise noted.
[b] Values used for backfill and native soil.
[c] Values from Appendix E of E-Area LLWF PA (Geochemistry), unless otherwise noted.
[d] Values used for Naval Reactor waste forms.
[e] Values are for clay from Sheppard and Thibault (1990), unless otherwise noted.
[f] Site-specific value from McIntyre (1988).
[g] Value NCRP 1996.
[h] Site-specific value from Hoeffner (1985).
[i] Site-specific value from Hoeffner (1984).
[j] Value from Ticknor and Ruegger (1989).
[k] Site-specific value from Oblath (1982).
[l] Used value for "Soil".
[m] Assume same value as for Am.

present analysis is that a somewhat different inventory of radionuclides is assessed, and the volatile radionuclides in both crud and activated metal are considered, rather than just activated metal.

To estimate fluxes of potentially volatile ^{3}H and ^{14}C from the Naval Reactor waste, two approaches are taken. For ^{3}H in both activated metal and crud, it is conservatively assumed that all of the ^{3}H is in a volatile form, is available for transport, and that all is released in one year. Thus, the fractional release in one year is unity (i.e., 1 Ci/yr per Ci inventory) for all ^{3}H. For the crud-associated ^{14}C, a slow evolution to a volatile form is assumed. "Crud" is a solid, adherent corrosion film on the surface of the metal components. Carbon in crud is likely to be initially either an elemental or metallic carbide form, neither of which are volatile. As water contacts the waste, one might expect a slow formation of ^{14}CO or $^{14}CO_2$, as well as non-volatile carbonates, despite the preferential use of dissolved oxygen in formation of metallic oxides. For this analysis, it is assumed conservatively that 0.5%/yr of the initial ^{14}C inventory in crud becomes volatile and is released to the atmosphere. This assumption maximizes the fractional release at 100 years after closure, which corresponds to the time at which institutional control is assumed to be lost, and exposure at 100 m from the waste become relevant.

For ^{14}C in activated metal, the corrosion rate of the waste form is also taken into consideration. Corrosion rates of Inconel and Zircaloy, the two representative alloys found in the Naval Reactor wastes, are discussed in Section 3.1.3.3. Briefly, effective corrosion rates for Inconel and Zircaloy are 5×10^{-5} and 5×10^{-6} cm/yr, respectively. Six different types of typical components were analyzed, each with different thicknesses, and thus different fractional corrosion and release rates. Data are provided describing the distribution of ^{14}C between the two different alloys. Thus, it is possible to estimate the fractional release of ^{14}C as follows.

For Inconel, with which 12% of the ^{14}C in activated metal is associated (Table 2.2-1), the fractional release of ^{14}C to corrosion products is at most 2×10^{-4} yr^{-1} (Table 3.1-1). For Zircaloy, with which 88% of the ^{14}C in activated metal is associated (Table 2.2-1), the fractional release of ^{14}C to corrosion products is at most 1×10^{-5} yr^{-1} (Table 3.1-1). For simplicity, it is conservatively assumed that all of the ^{14}C in crud evolved from the activated metal after closure in a year is released in the same year. Thus, the maximum release of ^{14}C from activated metals in a given year based on this method of estimation is $0.12(2 \times 10^{-4})+0.88(1 \times 10^{-5})$, or 3×10^{-5} Ci/yr per Ci in activated metal.

3.2 Analysis of Pathways and Scenarios

The source term analysis in Section 3.1 will allow estimation of release of radionuclides from the Naval Reactor waste disposal casks to the immediate environment. These releases are considered in this section with respect to how radionuclides might be further dispersed and ultimately lead to exposures to off-site members of the public. In order to evaluate exposures in terms of performance measures stated (Section 1.3), significant pathways and scenarios relevant to estimating exposures for the all-pathways analysis, the air pathway analysis, and the water resource protection analysis must be identified.

3.2.1 All-Pathways Analysis

From Section 4.2 of the E-Area LLWF PA (McDowell-Boyer et al. 2000), an analysis of all pathways leading to human exposure as a result of subsurface disposal of low-level radioactive waste at the SRS concluded that two pathways are of potential importance to the analysis. These pathways include: 1) leaching of the waste form resulting in contamination of groundwater local to E Area, and both direct ingestion of that groundwater and ingestion of meat and milk arising from cattle that drink

the contaminated groundwater; and 2) release of volatile radionuclides to air, subsequent contamination of agricultural soil, crops, and animals, inhalation of air and ingestion of food products contaminated by volatile radionuclides. The assessment of the release of volatile radionuclides to air is addressed in Section 3.2.2. Therefore, the remaining pathway of concern in the all-pathways analysis is the pathway involving groundwater.

For the all-pathways analysis, the performance objective limiting off-site exposures to 25 mrem/yr EDE for all exposure pathways (Section 1.3) is relevant. This objective is assumed to apply for 10,000 years in this analysis. In Section 4.2.3 of the E-Area LLWF PA, an analysis was done comparing the potential significance of the meat and milk pathway to human exposure relative to that of direct ingestion of groundwater. The results of this analysis were that the dose from direct ingestion of drinking water is expected to be considerably greater than the dose from ingestion of milk and meat obtained from dairy and beef cattle that drink contaminated water from the same source. This conclusion is expected to apply to all radionuclides that could be present in the Naval Reactor Waste Disposal Pad. Thus, only the drinking water pathway needs to be considered for off-site releases of radionuclides in groundwater.

The nearest location from the disposal site for off-site members of the public depends on the time after disposal. During the period of active institutional control, i.e., for the first 100 years after facility closure, off-site members of the public are assumed to be located no closer to the disposal site than the present boundary of the SRS. However, after active institutional control ceases, off-site members of the public could be located as close as 100 m from the Naval Reactor Waste Disposal Pad. Because the Naval Reactor waste casks are considered impervious to water until 750 years (Section 3.1.3.2), the dose analysis for off-site members of the public can focus on exposure pathways resulting from use of contaminated groundwater at a distance of 100 m from the facility for the time period between 750 years and 10,000 years. Thus, in the dose analysis for the groundwater pathway, an off-site member of the public is assumed to use water from a well for domestic purposes, and the well is assumed to be at the location at least 100 m from the Naval Reactor Waste Disposal Pad where the maximum concentrations of radionuclides in groundwater are predicted to occur after the container is breached.

3.2.2 Air Pathways Analysis

The air pathway analysis for the Naval Reactor Waste Disposal Pad is similar to that for the entire E-Area LLWF (Section 4.2.2, McDowell-Boyer et al. 2000), except that only the source term of volatile radionuclides for the Naval Reactor Waste Disposal Pad is analyzed here. Human exposure calculations related to the air pathway include six pathways; plume shine, ground deposition, inhalation, vegetation consumption, milk consumption and meat consumption. Parameter selection was made to maximize the calculated dose. The parameters are given in Table 3.2-1.

Two points and times of exposure were used in the analysis. For the first 100 years, the minimum time for which institutional control will be maintained, the point of maximum exposure will be at the SRS boundary, about 11 km north of E Area. Doses were also calculated at a point 100 m from the E-Area LLWF, including all disposal units, at 100 years after closure.

The dose limit used in the air pathway analysis is 10 mrem/yr, EDE, as required by 40 CFR 61. This limit is assumed to apply for 10,000 years in this analysis.

Table 3.2-1. Parameters Used in Air Dose Calculations

Relative Concentration (X/Q):	1.70E-04 (s/m^3)
Decayed X/Q:	1.70E-04 (s/m^3)
Depleted X/Q:	1.70E-04 (s/m^3)
Relative Deposition (D/Q):	5.20E-07 ($1/m^2$)
Distance to Receptor:	100 (m)
Vegetable Consumption:	276 (kg/yr)
Leafy Vegetable Consumption:	43 (kg/yr)
Milk Consumption:	230 (L/yr)
Meat Consumption:	81 (kg/yr)
Origin of Milk:	Cow
Deposition Buildup Time:	25 (yr)
Breathing Rate:	8,000 (m^3/yr)
Elemental Iodine Fraction:	1
Absolute Humidity:	0.0114 (kg/m^3)
Tritium Plant-to-Air Ratio:	0.5
Shielding Factor:	0.23
Fraction of Year ^{14}C Released:	1
Retained Fraction (iodine):	1
Retained Fraction (particulates):	0.2
Weathering Rate Constant:	18.1 (1/yr)
Crop Exposure Time:	0.164 (yr)
Pasture Grass Exposure Time:	0.0822 (yr)
Pasture Grass Productivity:	1.8(kg/m^2)
Produce Productivity:	0.7(kg/m^2)
Surface Soil Density (15 cm):	240(kg/m^2)
Pasture Grass Holdup Time:	0(yrs)
Stored Feed Holdup Time:	0.247(yr)
Leafy Vegetable Holdup Time:	0.00274(yr)
Produce Holdup Time:	0.164(yr)
Milk Cattle Feed Consumption:	44(kg/day)
Beef Cattle Feed Consumption:	44(kg/day)
Feed-Milk-Man Transport Time:	0.00548 (yr)
Fraction of Year on Pasture (beef):	1
Fraction of Year on Pasture (milk):	1
Fraction Intake from Pasture (beef):	0.75
Fraction Intake from Pasture (milk):	0.56
Slaughter to Consumption Time:	0.0164(yr)
Fraction of Produce from Garden:	0.76
Fraction of Leafy Vegetables from Garden:	1

3.2.3 Water Resource Impacts Analysis

Pathways to human exposure related to the water resource impacts analysis are limited to consideration of direct ingestion of groundwater. The point of compliance, 100 m from the edge of the Naval Reactor Waste Disposal Pad, is the same as that described in Section 3.2.1 for the all-pathways analysis. At this compliance point, the water resource impact assessment requirement of USDOE Order 435.1 is addressed. For this analysis and the E-Area LLWF PA, this is interpreted as requiring that concentrations of groundwater contaminants should not exceed values specified in USEPA standards for public drinking water supplies (40 CFR Part 141).As described in Section 1.3.2 of this report, the current USEPA drinking water standards for specifying the water resource impact assessment requirement, or performance measure, are used in this analysis. The performance measure is assumed to apply for 10,000 years after disposal (Section 1.3).

The MCLs based on current USEPA drinking water standards, used to specify the performance measure for groundwater impacts (Section 1.3.2), are given in Table 3.2-2 for radionuclides and daughters in the Naval Reactor waste which are of potential significance with respect to water resource protection (Section 3.1.2). For all radionuclides except uranium, the MCLs are given in units of pCi/L, even though the primary standard for beta/gamma-emitting radionuclides is a dose limit rather than a limit on concentration. For uranium and all alpha-emitting radionuclides, the MCLs in Table 3.2-2 are obtained directly from current USEPA standards for radioactivity in drinking water (USEPA 2000, USEPA 2001). For naturally-occurring uranium with its normal isotopic abundances, the MCL of 30 μg/L corresponds to an activity concentration of 21 pCi/L. The MCLs for beta/gamma-emitting radionuclides are obtained as described in Section 1.3.2, using dose conversion factors from the USDOE (1988) for radionuclides not listed in Handbook 69 (U.S. Department of Commerce 1963).

As described in the present section and Section 3.2.1, concentrations of radionuclides in groundwater at any location more than 100 m from the location of disposal units are limited by two performance measures: 1) a maximum EDE of 25 mrem per year from all exposure pathways involving use of contaminated water and 2) USEPA standards for limiting dose from consumption of drinking water only or concentrations of radionuclides in groundwater. The first performance measure applies to the all-pathways analysis (Section 3.2.1), and assumes that use of contaminated groundwater is the only significant source of exposure for off-site members of the public. This assumption is justified in the discussion in Section 4.2.1 of the E-Area LLWF PA (McDowell-Boyer et al. 2000). The question then arises as to which of the two performance measures would be the more restrictive, i.e., would result in lower limits on acceptable concentrations of radionuclides in groundwater and, thus, in waste in the Naval Reactor Waste Disposal Pad.

In order to address the relative importance of the two performance measures given above that apply to use of contaminated groundwater, the dose from direct ingestion of contaminated water must be compared to the dose from the pathways involving ingestion of milk and meat from dairy and beef cattle that drink contaminated water. This was done in Section 4.2.3 of the E-Area LLWF PA. As noted in Section 3.2.1, the dose from direct ingestion of contaminated drinking water is expected to be considerably greater than the dose from ingestion of milk and meat obtained from dairy and beef cattle that drink contaminated water from the same source. Therefore, the limit of concern in both the all-pathways analysis and the water resource protection analysis is the concentration of radionuclides in groundwater.

Table 3.2-2. Maximum Contaminant Levels for Radionuclides in Groundwater for Groundwater Protection Requirement

Radionuclide	MCL pCi/L[a,b]
^{3}H	20,000
^{14}C	2,000
^{59}Ni	300
^{79}Se	700[c]
^{90}Sr	8
^{93}Mo	4,000[c]
^{93}Zr	2,000
^{94}Nb	1,000[c]
^{99}Tc	900
^{126}Sn	300[c]
^{129}I	1
^{135}Cs	900
^{210}Pb	1[c]
^{226}Ra	5
^{227}Ac	1[c]
^{234}Th	400[c]
^{239}Np	300
Uranium	30[d]
^{241}Pu	300
Other Alpha[e]	15

[a] From USEPA 2001, unless otherwise noted.
[b] Values are in units of pCi/L, unless otherwise noted.
[c] Radionuclide not listed in Handbook 69. Value calculated using ICRP Publication 30 internal dosimetry, assuming 2L/d water intake and a 4 mrem/yr EDE limit (see Section 1.3.2).
[d] Value is in units of μg/L.
[e] Adjusted gross alpha emitters (excluding radon and uranium).

The comparative analysis from Section 4.2.3 of the E-Area LLWF PA (McDowell-Boyer et al. 2000) may be summarized as follows. First, for all beta/gamma-emitting radionuclides, the performance measure for protection of groundwater resources — i.e., a dose limit of either 4 mrem per year to whole body or any organ or 4 mrem per year EDE from the drinking water pathway only — should be more restrictive than the performance objective for protection of off-site members of the public — i.e., a dose limit of 25 mrem per year EDE from all exposure pathways — because the dose from the drinking water pathway only is expected to be greater than the dose from the milk and meat pathways combined. Thus, if the performance measure for groundwater protection is met, then the performance objective for protection of off-site individuals also will be met without the need for analysis of the dose from exposure pathways other than drinking water.

Second, for alpha-emitting radionuclides, the performance measure for protection of groundwater resources, which is expressed in terms of concentration limits rather than limits on dose equivalent, may result in doses from the drinking water pathway only that exceed the performance objective for protection of off-site members of the public. For example, the current MCL for ^{239}Pu in groundwater of 15 pCi/L (see Table 3.2-2) corresponds to an EDE of nearly 50 mrem per year, assuming consumption of 2 L/d of water and the ingestion DCF for ^{239}Pu given in Table C.3-2 in the E-Area LLWF PA (McDowell-Boyer et al. 2000). In these cases, the dose limit in the performance objective for off-site individuals would be more restrictive and, in principle, the contributions to the dose from exposure pathways other than drinking water would need to be considered in demonstrating compliance with the performance objective. However, the contribution from the meat and milk pathways is expected to be no more than a few tens of percent of the contribution from ingestion of groundwater, and should be much less for many radionuclides, in which case the other pathways essentially can be neglected in estimating dose. That is, in cases where the performance objective for off-site individuals from all potentially-significant exposure pathways applies, demonstration of compliance with the performance objective reasonably needs to take into account only the dose from the drinking water pathway. In other cases where the MCL for an alpha-emitting radionuclide corresponds to a dose less than the performance objective for off-site individuals, compliance with the MCL would ensure that the dose limit for all exposure pathways would be met without need for further analysis.

Therefore, the general conclusion from this analysis is that only the drinking water pathway needs to be considered for off-site releases of radionuclides in groundwater. In cases where the MCL in groundwater corresponds to a dose equivalent less than the performance objective for off-site individuals of 25 mrem per year from all exposure pathways, compliance with the MCL would ensure that the dose to off-site individuals would be substantially less than the performance objective. In cases where the MCL in groundwater corresponds to a dose equivalent greater than the performance objective for off-site individuals, the dose from all potentially-significant exposure pathways other than drinking water would be insignificant compared with the dose from the drinking water pathway, particularly when the uncertainties in estimating maximum concentrations of radionuclides in groundwater at locations more than 100 m from the Naval Reactor Waste Disposal Pad are taken into account.

3.3 Transport Analysis Methodology

The conceptual models developed for analyzing potential transport for radionuclides from the Naval Reactor Waste Disposal Pad to the point of compliance associated with the performance objectives of this analysis (Section 1.3) are described in this section. Methods used to implement these models, assumptions made in implementation, and justification for the assumptions are also discussed.

3.3.1 All-Pathways Analysis

Based on the discussions in Section 3.2.1, the only exposure pathway of concern for the all-pathways analysis, in addition to the air pathway analysis in Section 3.3.2, is the pathway involving direct ingestion of groundwater. In large part, the analysis of this pathway is identical to that for the water-resource protection analysis, and thus is described in Section 3.3.3. Doses to the off-site members of the public resulting from ingestion of contaminated groundwater beyond the 100-m buffer zone around the disposal facility were not directly estimated. Rather, comparisons of maximum predicted groundwater concentrations with the more restrictive of either MCLs (Table 3.2-2) or allowable concentrations based on the 25-mrem per year performance objective were made. The allowable concentrations were calculated by dividing the 25 mrem per year value by the EDE per unit concentration in drinking water (Table C.3-5, Appendix C.3 of the E-Area LLWF PA). A composite listing of MCLs and allowable concentrations based on the 25 mrem/yr limit is given in Table 3.3-1.

3.3.2 Air Pathway Analysis

Doses for unit releases of ^{3}H and ^{14}C were calculated at the site boundary (Simpkins and Carlton 1998) using the MAXIGASP computer program (Hamby 1995). Doses for unit releases at the E-Area boundary were calculated (Simpkins 1998) using the MAXINE spreadsheet (Hamby 1994) that implements the MAXIGASP methodology. The results are shown in Table 3.3-2.

Table 3.3-2. Dose for Unit Release via the Air Pathway

	Dose for Unit Release (mrem/Ci)	
Radionuclide	100 m	SRS Boundary
^{3}H (oxide)	8.5×10^{-3}	2.4×10^{-6}
^{14}C	3.8	1.0×10^{-3}

3.3.3 Water Resource Protection Analysis

The analysis for water resource protection requires that transport of radionuclides from the waste disposal units to the compliance points for groundwater protection (Section 1.3) be simulated. To carry out these simulations, conceptual transport models of two types were developed: models describing vadose, or unsaturated, zone transport (which include transport through disposal facility barriers); and a model describing transport through the saturated zone beneath the water table to the point of compliance. The conceptual models define how features of the disposal units and the subsurface environment are represented in the numerical models used to carry out the transport simulations. The key assumptions and values of parameters assumed for this analysis are described below.

Implementation of the conceptual models required the use of a computer code that can solve 3-dimensional flow and mass transport equations simultaneously. The code selected for this study is PORFLOW Version 4.00.7 (Analytic and Computational Research, Inc. [ACRI] 1998), which is an updated version of that (Version 3.0) used for the E-Area LLWF PA (McDowell-Boyer et al. 2000). The basis for code selection and the theoretical framework of PORFLOW is discussed in Appendix B of the E-Area PA. Model implementation involved simulation of flow and transport in the vadose zone, and simulation of transport only in the saturated zone. A steady-state saturated zone flow field provided for this study was simulated using the FACT code (Hamm et al. 1996), and is essentially the

Table 3.3-1. Comparison of MCLs and Allowable Groundwater Concentrations Based on the 25 mrem per Year Performance Objective for Off-site Individuals

Radionuclide	MCL,[a] pCi/L	Allowable Concentration Based on 25 mrem per Year, pCi/L[b]
^{3}H	20,000	540,000
^{14}C	2,000	16,000
^{59}Ni	300	160,000
^{79}Se	700	4,100
^{90}Sr	8	250
^{93}Mo	4,000	26,000
^{93}Zr	2,000	16,000
^{94}Nb	1,000	6,800
^{99}Tc	900	26,000
^{126}Sn	300	1,900
^{129}I	1	130
^{135}Cs	900	4,800
^{210}Pb	1	6.8
^{210}Po	15	21
^{223}Ra	15	63
^{224}Ra	15	93
^{226}Ra	5	31
^{227}Ac	1	2.5
^{227}Th	15[c]	960
^{228}Ra	5	28
^{228}Th	15[c]	89
^{229}Th	15[c]	9.6
^{230}Th	15[c]	64
^{231}Pa	15[c]	3.1
^{232}Th	15[c]	13
^{232}U	640,000,000 (30 µg/L)	26
^{234}Th	400	2,600
^{234}U	190,000 (30 µg/L)	130
^{235}U	65 (30 µg/L)	140
^{236}U	1,900 (30 µg/L)	140
^{237}Np	15[c]	8.9
^{238}U	10 (30 µg/L)	150
^{238}Pu	15[c]	8.9
^{239}Np	300	12,000

Table 3.3-1. Comparison of MCLs and Allowable Groundwater Concentrations Based on the 25 mrem per Year Performance Objective for Off-site Individuals (cont.)

Radionuclide	MCL,[a] pCi/L	Allowable Concentration Based on 25 mrem per Year, pCi/L[b]
^{239}Pu	15[c]	8.1
^{240}Pu	15[c]	8.1
^{241}Pu	300	400
^{242}Pu	15[c]	8.3
^{241}Am	15[c]	7.6
^{243}Am	15[c]	7.6
^{244}Pu	15[c]	8.6
^{244}Cm	15[c]	15
^{245}Cm	15[c]	7.6
^{246}Cm	15[c]	7.6
^{247}Cm	15[c]	8.3
^{248}Cm	15[c]	2.1
^{249}Cf	15[c]	7.4

[a] From Table 3.2-2.
[b] Calculated from Table C.3-5 in E-Area LLWF PA; potentially-significant short-lived daughters are accounted for.
[c] Alpha-emitter other than radon or uranium.

same flow field used for the E-Area LLWF PA except that a smaller portion of the simulation grid is used for the present study.

3.3.3.1 Vadose Zone Flow and Transport

The conceptual flow model of the unsaturated, or vadose, zone is represented by a two-dimensional vertical cross-section of the Naval Reactor Waste Disposal Pad (Figure 3.3-1). Analysis in 2-dimensions is sufficient for this problem, because releases along the length of the Naval Reactor Waste Disposal Pad are expected to be uniform except at each end of the pad.

Releases from the end-planes of the pad are expected to be insignificant relative to releases from the combined areas represented by the bottom and sides of the pad, and thus no corrections are made to account for these releases. The results in 2-dimensions, which assume a unit depth in the third dimension, are readily adapted to 3-dimensions. The modeling domain is 200 ft in the horizontal direction and 90 ft in the vertical direction. The choice of a 200-ft width is arbitrary, as the exact placement of the individual containers on the pad, and shape of the pad, has yet to be determined (Section 2.2.1). However, a 200-ft domain width, and 140-ft width assumed for the waste (Figure 3.3-1), allows for a waste placement that maximizes the flux over a given area, as this assumes the containers are placed close together in a square pattern. Only half of the horizontal distance is simulated, taking advantage of symmetry.

As noted earlier (Section 2.1), the Naval Reactor waste containers will be disposed at grade. In the waste disposal area, the average water table is approximately 70 ft below the ground surface. However, 60 ft is assumed for the depth to the water table in the model, which is conservative.

The source of water into the modeling domain comes from infiltration at the ground surface, or top of the domain. A 40 cm/yr net infiltration rate is assumed, which is an average for E Area (Appendix C.1.1, E-Area LLWF PA). A closure cap will be constructed after E-Area LLWF is filled (Section 2.2.1) and will likely reduce water infiltration into the waste for several hundred years. As noted earlier (Section 3.1.3.1), failure of the moisture barrier is not critical to this analysis because water cannot contact the waste until the casks have failed, which is not expected before 750 years. After this period, the cap is degraded and it is assumed here that the entire degraded closure cap has the flow properties of the backfill soil. Because the Naval Reactor waste casks are assumed to be intact for 750 years, long after the cap failure, a single steady-state flow field was constructed for the vadose zone analysis, which assumes the cap has failed. Contaminant transport modeling starts at 750 years using this flow field.

In order to solve the flow equations, it was necessary to specify boundary conditions and hydraulic properties of the materials present in the simulation domain. The top of the domain is a constant flux boundary, where the net infiltration rate of 40 cm/yr is applied. The bottom of the domain is a constant head boundary, maintained by the presence of the water table. The left and right boundaries are no flow boundaries, where flow is essentially vertical and parallel to these boundaries, as described above. Pressure at the water table is arbitrarily set to zero. Flow velocity at a modeling node is proportional to the pressure gradient and not the absolute pressure.

Because water saturation at the water table is 1.0, by definition, the capillary pressure (P_C) is also zero. In the modeling domain, water saturation decreases with elevation and capillary pressure increases. Capillary pressure is a function of saturation. The capillary pressure curve [Yu et al. 1993] used for soil in the vicinity of the Naval Reactor Waste Disposal Pad is depicted in Figure 3.3-2.

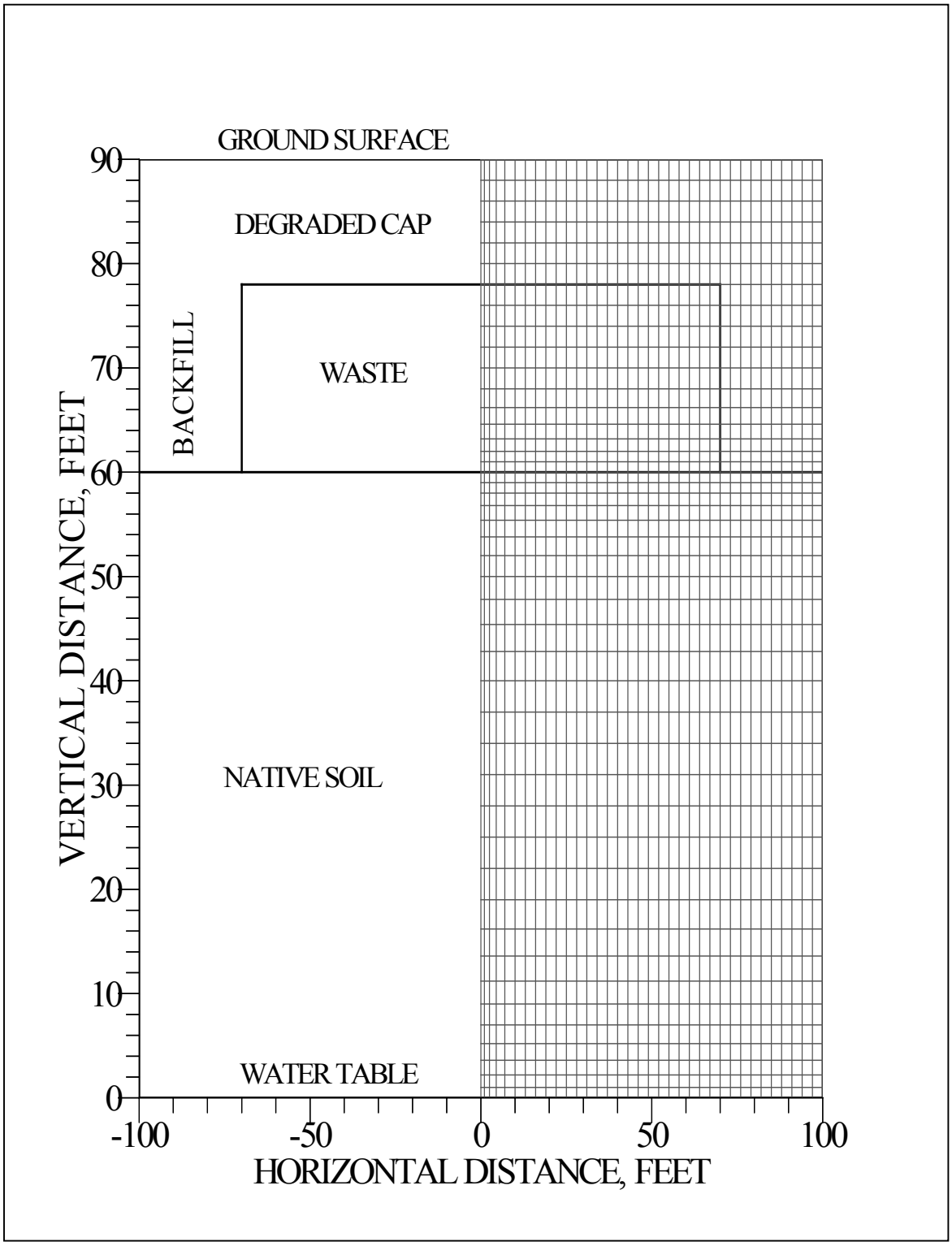

Figure 3.3-1. Conceptual Model and Modeling Grid for the Naval Reactor Waste Disposal Pad

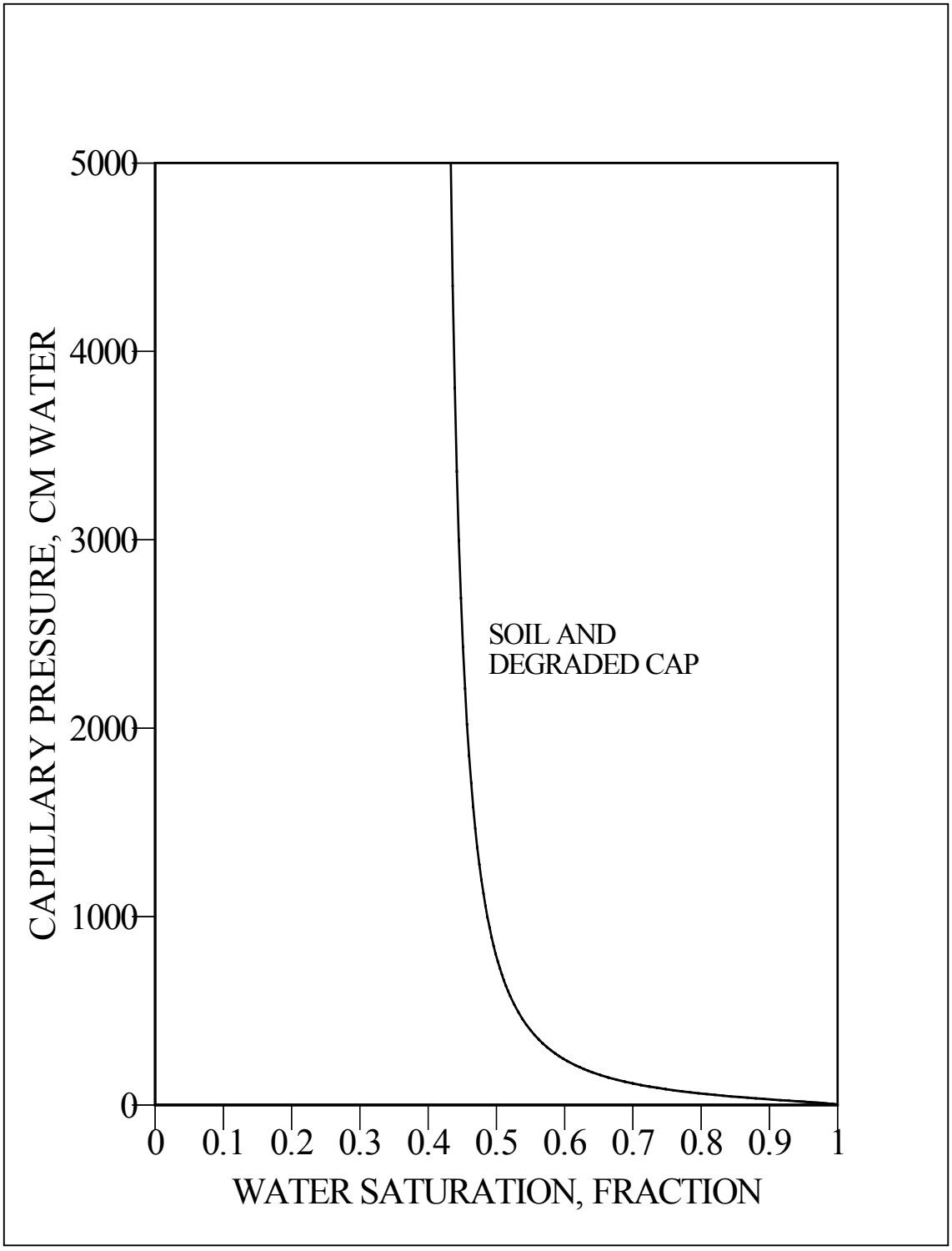

Figure 3.3-2. Capillary Pressure Curve Used for Unsaturated-Zone Modeling

When the soil is not saturated, its relative permeability (conductivity) to water is also reduced. Relative permeability is the permeability to water at reduced saturation divided by that at $S_w = 1$ (i.e., at saturation). Figure 3.3-3 depicts the relative permeability as a function of water saturation used in simulating the vadose zone flow field. As indicated, when water saturation = 1.0, relative permeability = 1.0, by definition. The saturated permeability for soil and degraded cap used is 315.4 cm/yr (10^{-5} cm/s).

In the E-Area PA, the saturated permeability of the degraded cap and waste was assumed to be 3.154×10^4 cm/yr (10^{-3} cm/s), while the underlying soil was assumed to be 315.4 cm/yr (10^{-5} cm/s). In this former scenario, the less permeable underlying soil controls the rate of movement of the water and contaminants from the waste zone to the water table. Thus, the current change in assumed permeability does not significantly affect the rate of flow and transport from the waste zone.

With user-defined assumptions, boundary conditions, initial conditions, material properties, and numerical schemes, PORFLOW allows calculation of water saturation and velocity at every grid node in the modeling domain. In this study, it is assumed that the flow is at steady state. This means the saturation and flow velocity at each node do not change with time. A steady-state solution is obtained by running the transient model for 2,000 years. As expected, the velocity at every node is 40 cm/year vertically toward the water table.

Mass transport simulations were carried out in this study using the steady-state flow field described above. The end result of these computations is a time-dependent flux of radionuclides out of the vadose zone into the saturated zone. These simulations consider contaminant release to the vadose zone from the waste, adsorption onto porous materials, convection and diffusion through the porous media, and radioactive decay. The release rate developed in Section 3.1.3, quantifying release of radionuclides from activated metal in the Naval Reactor Waste Disposal Pad, is incorporated into the PORFLOW mass transport simulations.

Adsorption and diffusion coefficients described in Section 3.1.3 are used in the modeling specifications. Longitudinal and transverse dispersivities are assumed to be zero, as was assumed and justified in the E-Area LLWF PA (Section 4.3.3.1). Radioactive decay and ingrowth of radioactive daughter products are accounted for explicitly in the PORFLOW code, using half-lives tabulated by Kocher (1981). Boundary conditions on mass assumed for transport modeling are as follows. At the top and bottom of the domain, a constant concentration of zero is assigned, which corresponds to the assumption that contaminants reaching either boundary are rapidly swept away from the vicinity of contact. This maximizes the diffusive flux out of the bottom of the domain by maximizing the concentration gradient. Thus, this assumption is conservative. At the vertical boundaries, diffusive fluxes are set to zero due to symmetry. Convective fluxes at the vertical boundaries are also zero, due to the no flow boundaries imposed at these locations within the simulation domain.

Estimated fluxes at the water table are reported as Ci/yr at the water table per Ci initially present in either “crud” or “activation” waste in the Naval Reactor Waste Disposal Pad. The list of radionuclides for which results are provided are those listed in Section 3.1-2, which includes only those radionuclides that are considered potentially significant with respect to air and water resource contamination, according to the screening methods described in Section 4.1.3.1.1 of the E-Area LLWF PA. Peak fractional fluxes to the water table for radionuclides present in “crud” waste in the Naval Reactor Waste Disposal Pad are given in Table 3.3-3; for radionuclides present in “activation” waste, peak fractional fluxes are given in Table 3.3-4.

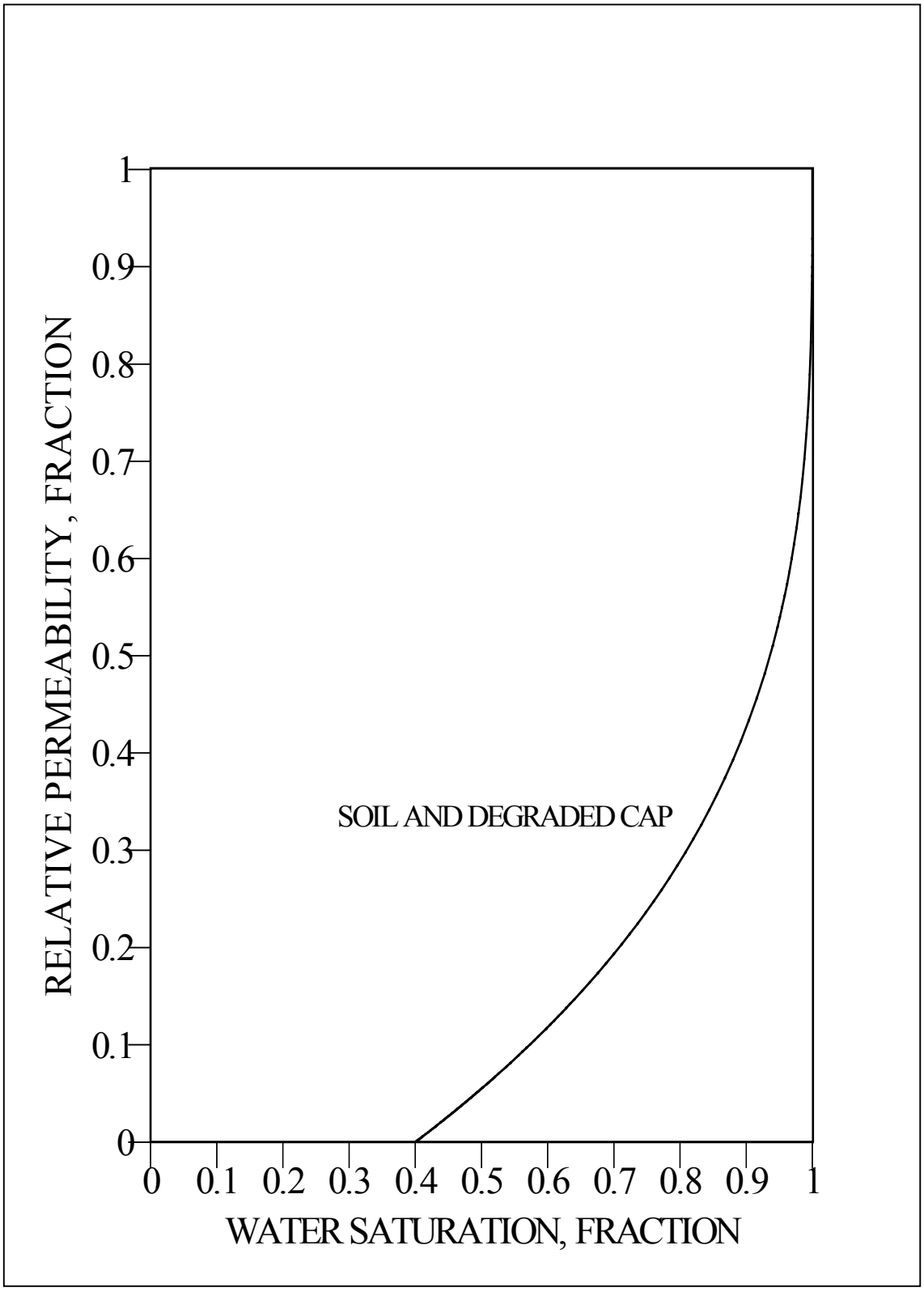

Figure 3.3-3. Relative Permeability Curve Used for Unsaturated-Zone Modeling

Table 3.3-3. Predicted Peak Fluxes and Peak Times for the "Crud" Unsaturated Zone Transport Runs

Radionuclides[a]	Peak Flux Fraction/Year	Peak Time Years
^{3}H	2.97×10^{-20}	763
^{14}C	1.57×10^{-2}	865
^{59}Ni	6.85×10^{-8}	10000
^{79}Se	4.50×10^{-4}	1202
$^{90}Sr+d$	3.05×10^{-15}	1036
^{93}Mo	2.95×10^{-3}	1276
^{93}Zr	2.06×10^{-10}	10000
^{93m}Nb	8.71×10^{-10}	10000
^{94}Nb	1.84×10^{-4}	8787
^{99}Tc	6.21×10^{-2}	782
$^{126}Sn+d$	3.24×10^{-4}	6808
^{129}I	4.53×10^{-2}	794
^{135}Cs	3.10×10^{-6}	10000
^{232}Th	7.95×10^{-28}	10000
^{228}Ra	5.81×10^{-27}	10000
^{228}Th	9.08×10^{-28}	10000
^{224}Ra	5.82×10^{-27}	10000
^{232}U	9.85×10^{-15}	1822
^{228}Th	1.09×10^{-16}	1822
^{224}Ra	6.96×10^{-16}	1822
^{234}U	1.28×10^{-5}	3635
^{230}Th	9.41×10^{-9}	10000
^{226}Ra	4.42×10^{-8}	10000
^{210}Pb	8.14×10^{-8}	10000
^{210}Po	1.46×10^{-7}	10000
^{235}U	1.29×10^{-5}	3931
^{231}Pa	1.32×10^{-7}	9999
^{227}Ac	1.60×10^{-7}	10000
^{227}Th	2.25×10^{-8}	10000
^{223}Ra	1.44×10^{-7}	10000
^{236}U	1.29×10^{-5}	3977
^{238}U	1.29×10^{-5}	10000
^{234}Th	1.43×10^{-7}	4079
^{234}U	3.60×10^{-7}	10000
^{237}Np	1.03×10^{-4}	1258
^{238}Pu	1.86×10^{-18}	2747
^{234}U	4.59×10^{-9}	3626
^{239}Pu	3.08×10^{-5}	7538

Table 3.3-3. Predicted Peak Fluxes and Peak Times for the "Crud" Unsaturated Zone Transport Runs (cont.)

Radionuclides[a]	Peak Flux Fraction/Year	Peak Time Years
^{235}U	2.28×10^{-10}	9999
^{240}Pu	1.78×10^{-5}	6835
^{236}U	4.30×10^{-9}	6586
^{241}Pu	6.68×10^{-48}	1267
^{241}Am	1.02×10^{-30}	10000
^{237}Np	6.94×10^{-10}	3607
^{242}Pu	3.81×10^{-05}	8547
^{238}U	4.36×10^{-11}	10000
^{244}Pu	3.88×10^{-5}	10000
^{241}Am	2.94×10^{-29}	10000
^{237}Np	2.08×10^{-8}	3644
^{243}Am	1.09×10^{-22}	10000
^{239}Np	3.97×10^{-20}	10000
^{239}Pu	5.36×10^{-6}	10000
^{244}Cm	1.68×10^{-85}	1480
^{240}Pu	4.93×10^{-8}	6835
^{236}U	1.19×10^{-11}	6576
^{245}Cm	5.92×10^{-31}	10000
^{241}Pu	1.89×10^{-25}	10000
^{241}Am	7.07×10^{-26}	10000
^{237}Np	2.11×10^{-7}	9980
^{246}Cm	3.11×10^{-31}	10000
^{247}Cm	1.34×10^{-30}	10000
^{243}Am	2.04×10^{-23}	10000
^{239}Np	7.39×10^{-21}	10000
^{239}Pu	2.22×10^{-6}	10000
^{248}Cm	1.31×10^{-30}	10000
^{244}Pu	2.60×10^{-9}	10000
^{249}Cf	1.50×10^{-18}	9499
^{245}Cm	5.68×10^{-20}	10000
^{241}Pu	3.02×10^{-18}	10000
^{241}Am	1.70×10^{-19}	10000
^{237}Np	8.62×10^{-9}	10000

[a]Indented entries are potentially radiologically-significant decay products of parent radionuclides.

Table 3.3-4. Predicted Peak Fluxes and Peak Times for the "Activation" Unsaturated Zone Transport Runs

Radionuclide[a]	Peak Flux Fraction/Year	Peak Time Years
^{3}H	9.63×10^{-24}	768
^{14}C	1.45×10^{-5}	1008
^{59}Ni	2.19×10^{-9}	10000
^{79}Se	6.00×10^{-5}	4699
^{90}Sr+d	4.85×10^{-18}	1036
^{93}Mo	2.95×10^{-3}	1276
^{93}Zr	9.38×10^{-13}	10000
^{93m}Nb	4.01×10^{-12}	10000
^{94}Nb	2.48×10^{-5}	10000
^{99}Tc	4.92×10^{-5}	833
^{126}Sn+d	5.71×10^{-5}	9610
^{129}I	6.33×10^{-5}	2608
^{135}Cs	1.37×10^{-7}	10000
^{232}Th	1.72×10^{-29}	10000
^{228}Ra	1.26×10^{-28}	10000
^{228}Th	1.97×10^{-29}	10000
^{224}Ra	1.26×10^{-28}	10000
^{232}U	1.71×10^{-16}	1988
^{228}Th	1.88×10^{-18}	1988
^{224}Ra	1.20×10^{-17}	1988
^{234}U	4.83×10^{-6}	9999
^{230}Th	2.12×10^{-9}	10000
^{226}Ra	8.37×10^{-9}	10000
^{210}Pb	1.53×10^{-8}	10000
^{210}Po	2.76×10^{-8}	10000
^{235}U	4.97×10^{-6}	10000
^{231}Pa	3.01×10^{-8}	10000
^{227}Ac	3.65×10^{-8}	10000
^{227}Th	5.13×10^{-9}	9999
^{223}Ra	3.29×10^{-8}	10000
^{236}U	4.96×10^{-6}	9999
^{238}U	4.97×10^{-6}	10000
^{234}Th	5.47×10^{-8}	9999
^{234}U	1.36×10^{-7}	10000
^{237}Np	3.81×10^{-5}	10000
^{238}Pu	5.65×10^{-21}	2941
^{234}U	3.36×10^{-10}	10000
^{239}Pu	1.34×10^{-6}	10000

Table 3.3-4. Predicted Peak Fluxes and Peak Times for the "Activation" Unsaturated Zone Transport Runs (cont.)

Radionuclide[a]	Peak Flux Fraction/Year	Peak Time Years
^{235}U	1.48×10^{-11}	10000
^{240}Pu	6.28×10^{-7}	9999
^{236}U	2.77×10^{-10}	10000
^{241}Pu	1.30×10^{-50}	1415
^{241}Am	4.48×10^{-33}	10000
^{237}Np	5.50×10^{-11}	9999
^{242}Pu	1.74×10^{-6}	10000
^{238}U	2.80×10^{-12}	10000
^{244}Pu	1.77×10^{-6}	10000
^{241}Am	8.46×10^{-31}	10000
^{237}Np	7.71×10^{-9}	9971
^{243}Am	2.51×10^{-24}	10000
^{239}Np	9.10×10^{-22}	10000
^{239}Pu	1.33×10^{-6}	10000
^{244}Cm	2.46×10^{-87}	1628
^{240}Pu	1.09×10^{-8}	9999
^{236}U	4.11×10^{-12}	10000
^{245}Cm	1.28×10^{-32}	10000
^{241}Pu	9.77×10^{-27}	10000
^{241}Am	3.53×10^{-27}	10000
^{237}Np	9.23×10^{-8}	10000
^{246}Cm	6.76×10^{-33}	10000
^{247}Cm	2.86×10^{-32}	10000
^{243}Am	5.12×10^{-25}	10000
^{239}Np	1.86×10^{-22}	10000
^{239}Pu	4.83×10^{-7}	10000
^{248}Cm	2.80×10^{-32}	10000
^{244}Pu	6.26×10^{-10}	9999
^{249}Cf	6.33×10^{-20}	10000
^{245}Cm	1.90×10^{-21}	10000
^{241}Pu	1.02×10^{-19}	10000
^{241}Am	5.48×10^{-21}	10000
^{237}Np	3.69×10^{-9}	9999

[a]Indented entries are potentially radiologically-significant decay products of parent radionuclides.

Predicted fractional release rates for "crud" and "activation" ^{14}C and ^{59}Ni are depicted in Figures 3.3-4 and 3.3-5, respectively. For ^{14}C, the peak fluxes occur around 1000 years. For ^{59}Ni, peak fluxes in the 10,000-year time frame occur at 10,000 years. The true peak for ^{59}Ni occurs after 20,000 years, but is not of concern for this study. Although the "crud" peak is greater for both radionuclides, these are fractional fluxes, and thus are not directly indicative of the contribution of both types of waste forms to the flux. For ^{14}C, the total projected "crud" inventory is approximately 200 times less than the "activation" inventory (Table 2.4-2); thus, the peak flux due to the "crud" component is only about a factor of five higher than the peak flux due to the "activation" component. For ^{59}Ni, the total projected "crud" inventory is approximately 7,000 times less than the "activation" inventory (Table 2.4-2); thus, the peak flux due to the "crud" component is actually about a factor of 200 lower than the peak flux due to the "activation" component ^{14}C.

3.3.3.2 Saturated Zone Flow and Transport

The conceptual flow model of the saturated zone in the vicinity of the Naval Reactor Waste Disposal Pad is represented by a three-dimensional model developed by the Environmental Sciences Section, Savannah River Technology Center (SRTC). A comprehensive description of the model is given in Section 4.3.3.7.2 of the E-Area LLWF PA. Briefly, the SRTC model simulates groundwater flow in the General Separations Area of the SRS, within the area bounded by Fourmile Branch to the south, Upper Three Runs to the north, F Area to the west, and McQueen Branch to the east (Figure 3.1-2 of the E-Area LLWF PA). The upper boundary of the model domain is the ground surface, and the lower boundary is the bottom of the Gordon aquifer. The FACT code (Appendix B of the E-Area LLWF PA) was used to implement the model and derive the steady-state flow field used in this study.

The FACT code uses a $109 \times 78 \times 21$ mesh for the model area. In the E-Area LLWF PA, a smaller mesh of $51 \times 49 \times 21$ is used for the entire LLWF. In the present study of the Naval Reactor Waste Disposal Pad, the mesh is further reduced to $26 \times 26 \times 13$, a grid size sufficient for estimating peak groundwater concentrations of radionuclides potentially released from the facility at the compliance point, located at least 100 m from the edge of the pad (Section 1.3). In other words, the grid for this study represents a subset of the grid developed for the entire General Separations Area, but still provides the same resolution, or grid element size of 200 ft by 200 ft. Testing was done of the reduced grid to verify that changing the grid size did not significantly affect results.

Transport of radionuclides is simulated using PORFLOW and the flow field generated by the FACT code. Although the FACT code can simulate transport, it presently does not have the capability of addressing radioactive decay and daughter ingrowth during transport as does PORFLOW. The coordinates and flow rates of the FACT-generated flow field are first translated into a format appropriate for use in PORFLOW, requiring the use of FORTRAN programs developed for this purpose (see appendices). The final PORFLOW mesh is $27 \times 27 \times 14$, reflecting an increase of one node in each direction due to the PORFLOW convention of treating each boundary node as a "half node".

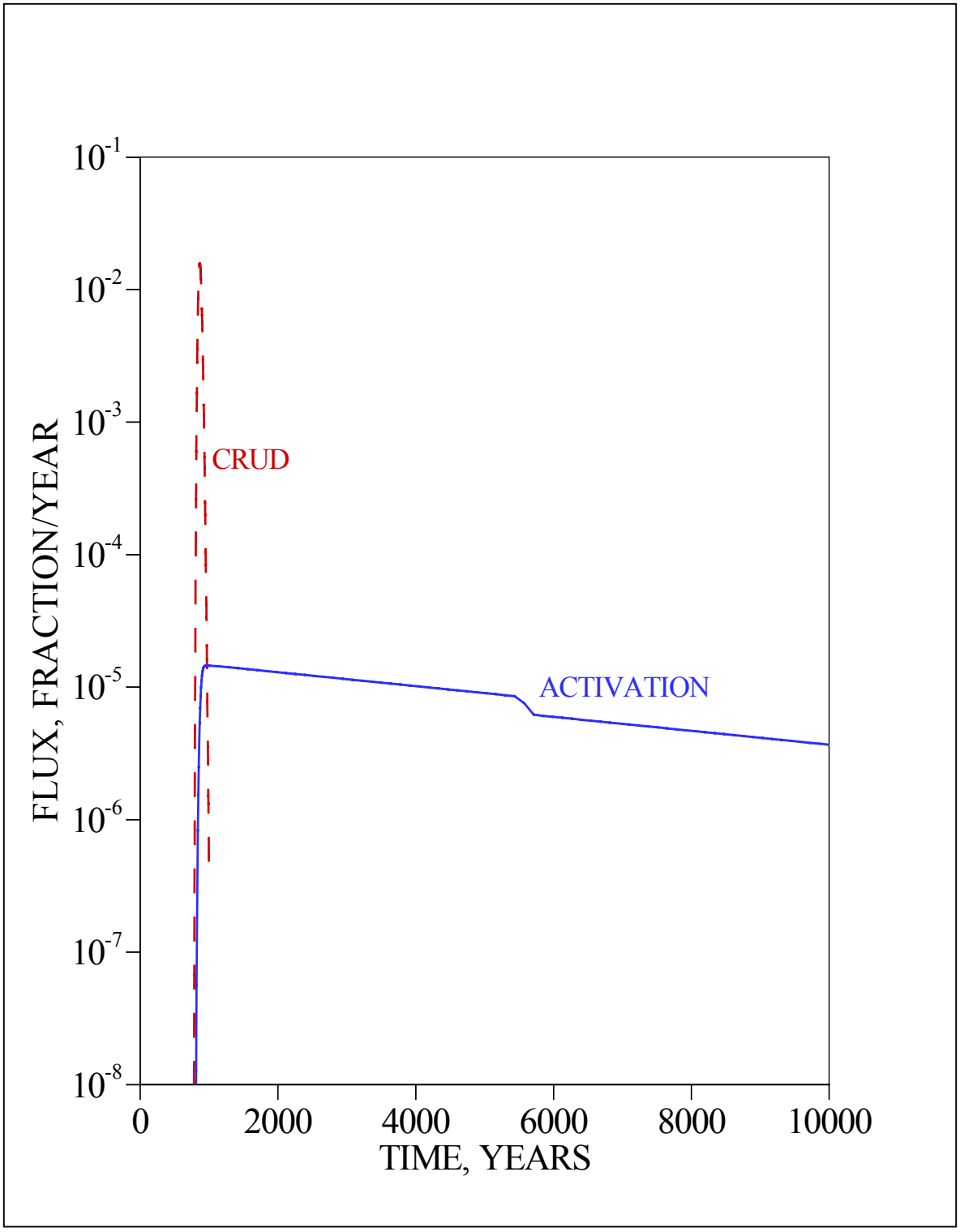

Figure 3.3-4. Comparison of ^{14}C "Crud" and "Activation" Fluxes to the Water Table

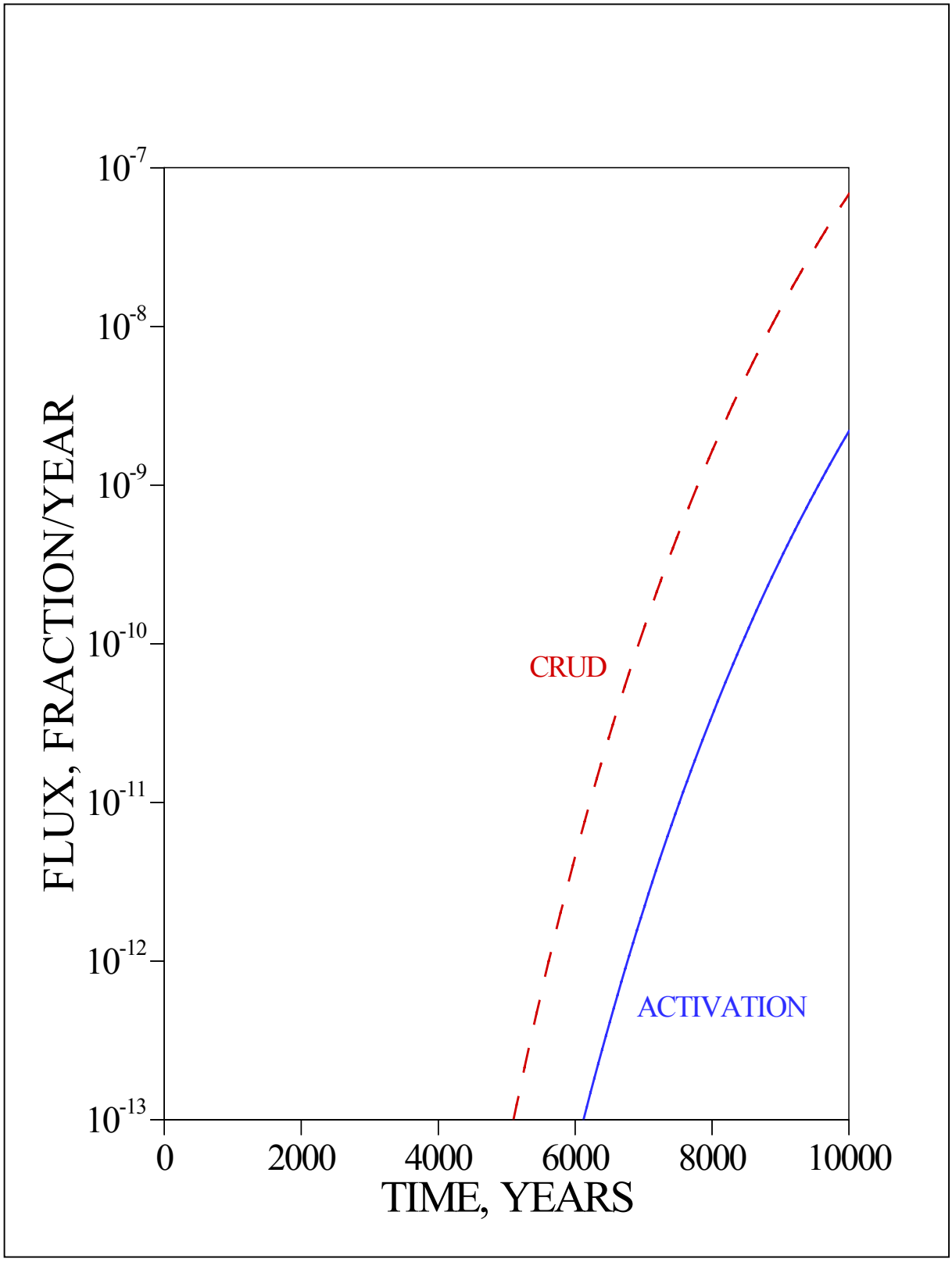

Figure 3.3-5. Comparison of ^{59}Ni "Crud" and "Activation" Fluxes to the Water Table

In Figure 3.3-6, a sub-domain of the 27 × 27 horizontal mesh is shown in relation to the LLWF boundaries and a conceptual Naval Reactor Waste Disposal Pad. As noted earlier, each block in the mesh is 200 × 200 ft. The grid blocks chosen to represent the Naval Reactor Waste Disposal Pad are in the location planned for this facility and are L-shaped, although the exact configuration of the final pad, based on cask placement, is not known at this time (Section 2.2.1). The area covered by the three grid blocks, which represents the area of source input in the model, is an area covering $3 \times (200)^2$ ft^2, or 120,000 ft^2. The source nodes are the nodes into which the time-dependent fluxes of radionuclides calculated in the vadose zone modeling are input at the level of the water table. The water table elevation corresponds to the 5th node from the bottom of the modeling domain for the saturated zone.

As with the vadose zone transport simulations, transport of radionuclides introduced to the saturated zone under E Area occurs as a result of advective and dispersive processes, but is hindered by sorptive and radioactive decay processes. Dispersion, caused by diffusion of solute molecules and mechanical mixing, is simulated by PORFLOW in the advection-dispersion equation. Required inputs for addressing dispersion and diffusion are discussed in Section 4.3.3.7.3 of the E-Area LLWF PA. Calculation of groundwater concentration at each node of the modeling domain is accomplished in PORFLOW after taking into account radioactive decay, radioactive daughter ingrowth, sorption, and diffusion. Half-lives used for these calculations are taken from Kocher (1981). Element-specific Kds used for these calculations are identical to those used for the source term analysis (Table 3.1-2). The value for particle density of the media, required by PORFLOW for calculating retardation of sorbing contaminants, is taken from the E-Area LLWF PA, and is 2650 kg/m^3. A diffusional porosity of 0.25 was assumed for all lithologies, and is also taken from the PA.

At the location of the source nodes, shown in Figure 3.3-6, the general groundwater flow is north, east, and downward. To find the point of peak groundwater concentration at least 100 m from the edge of the facility, horizontal node locations of (11,15), (12,15), and (13,15), designated as "compliance nodes" in Figure 3.3-6, were monitored over time. These nodes are approximately 100 m from the edge of the Naval Reactor Waste Disposal Pad. The peak is expected to occur at one of these lateral locations, at a depth dependent on the strength of the downward gradient in this area. Vertical nodes 2 through 6 from the bottom of the domain were monitored to determine the location of the node in which peak concentration node occurs at the compliance locations. Locating the peak concentration requires searching the computed concentrations at each node at each time step, and was accomplished with a FORTRAN program developed for that purpose.

Transport simulations accomplished with PORFLOW are done assuming the flow field is at steady state throughout the time period of interest, up to 10,000 years. The steady-state flow field from FACT specifies fluxes at each node. By disabling the flow calculation in PORFLOW, these fluxes are held steady. Thus, flow boundary conditions are immaterial. However, contaminant transport may be affected by diffusion boundary conditions. For the groundwater transport simulations, all six external faces of the 3-dimensional modeling domain are designated as zero concentration boundaries. This is a conservative choice for this problem but has very little effect on the results of this study since the compliance nodes are sufficiently far from the domain boundaries.

Optimal time steps for the PORFLOW simulations were selected on a trial-and-error basis, the goal being to preserve numerical stability and satisfy mass balance requirements on a grid element-by-grid element basis. Output of the simulations is in terms of normalized groundwater concentration at the point of maximum concentration for each radionuclide simulated, which was found to be the node (12,15,5). Groundwater concentrations are normalized to one Ci of activity of each radionuclide disposed in the Naval Reactor Waste Disposal Pad. Because "crud" and "activation" activities were run separately, separate peak groundwater concentrations are presented in Tables 3.3-5 and 3.3-6.

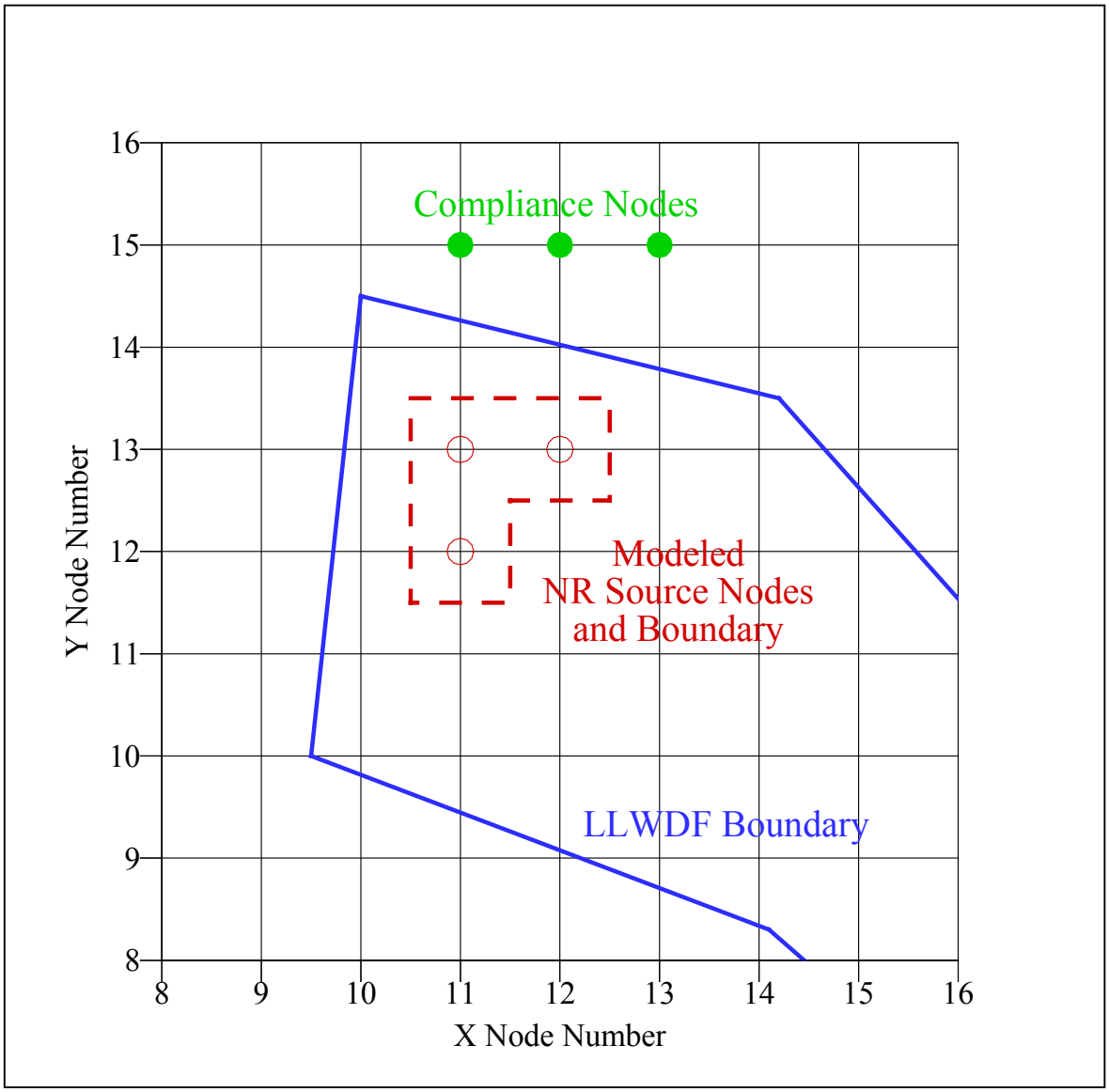

Figure 3.3-6. Locations of Naval Reactor Source and Compliance Nodes

Table 3.3-5. Predicted Groundwater Peak Concentrations Based on 1.0 Ci Initial "Crud" Inventory of the Parent Radionuclide

Radionuclide[a]	Peak Concentration pCi/L-Ci	Peak Time Years
^{3}H	9.85×10^{-16}	765
^{14}C	4.70×10^{2}	904
^{59}Ni	1.48×10^{-5}	10000
^{79}Se	1.95×10^{1}	1463
$^{90}Sr+d$	5.65×10^{-12}	1114
^{93}Mo	8.38×10^{1}	1456
^{93}Zr	8.90×10^{-9}	10000
^{93m}Nb	3.87×10^{-8}	10000
^{94}Nb	3.68	10000
^{99}Tc	2.04×10^{3}	791
$^{126}Sn+d$	8.73	9120
^{129}I	1.43×10^{3}	808
^{135}Cs	1.54×10^{-3}	10000
^{232}Th	1.23×10^{-28}	10000
^{228}Ra	9.12×10^{-28}	10000
^{228}Th	1.43×10^{-28}	10000
^{224}Ra	9.14×10^{-28}	10000
^{232}U	1.05×10^{-11}	2028
^{228}Th	1.15×10^{-13}	2033
^{224}Ra	7.38×10^{-13}	2033
^{234}U	5.54×10^{-1}	5373
^{230}Th	3.69×10^{-4}	10000
^{226}Ra	1.67×10^{-3}	10000
^{210}Pb	3.08×10^{-3}	10000
^{210}Po	5.54×10^{-3}	10000
^{235}U	5.63×10^{-1}	6453
^{231}Pa	5.20×10^{-3}	10000
^{227}Ac	6.31×10^{-3}	10000
^{227}Th	8.88×10^{-4}	10000
^{223}Ra	5.68×10^{-3}	10000
^{236}U	5.63×10^{-1}	6479
^{238}U	5.63×10^{-1}	6347
^{234}Th	6.17×10^{-3}	6347
^{234}U	1.57×10^{-2}	10000
^{237}Np	4.50	1573
^{238}Pu	2.94×10^{-16}	3060
^{234}U	1.99×10^{-4}	5376
^{239}Pu	1.19	10000

Table 3.3-5. Predicted Groundwater Peak Concentrations Based on 1.0 Ci Initial "Crud" Inventory of the Parent Radionuclide (cont.)

Radionuclide[a]	Peak Concentration pCi/L-Ci	Peak Time Years
^{235}U	1.22×10^{-5}	10000
^{240}Pu	5.64×10^{-1}	9329
^{236}U	2.22×10^{-4}	8546
^{241}Pu	7.66×10^{-48}	1329
^{241}Am	8.25×10^{-31}	10000
^{237}Np	3.02×10^{-5}	3839
^{242}Pu	1.56	10000
^{238}U	2.35×10^{-6}	10000
^{244}Pu	1.59	10000
^{241}Am	2.39×10^{-29}	10000
^{237}Np	9.04×10^{-4}	4360
^{243}Am	8.86×10^{-23}	10000
^{239}Np	3.31×10^{-20}	10000
^{239}Pu	1.50×10^{-1}	10000
^{244}Cm	6.69×10^{-90}	1560
^{240}Pu	1.56×10^{-3}	9330
^{236}U	6.12×10^{-7}	8555
^{245}Cm	4.55×10^{-32}	10000
^{241}Pu	2.14×10^{-25}	10000
^{241}Am	8.22×10^{-26}	10000
^{237}Np	9.16×10^{-3}	10000
^{246}Cm	2.39×10^{-32}	10000
^{247}Cm	1.03×10^{-31}	10000
^{243}Am	1.53×10^{-23}	10000
^{239}Np	5.73×10^{-21}	10000
^{239}Pu	4.22×10^{-2}	10000
^{248}Cm	1.01×10^{-31}	10000
^{244}Pu	6.80×10^{-5}	10000
^{249}Cf	1.16×10^{-16}	10000
^{245}Cm	2.47×10^{-18}	10000
^{241}Pu	1.35×10^{-16}	10000
^{241}Am	6.74×10^{-18}	10000
^{237}Np	3.74×10^{-4}	10000

[a]Indented entries are potentially radiologically-significant decay products of parent radionuclides.

Table 3.3-6. Predicted Groundwater Peak Concentrations Based on 1.0 Ci Initial "Activation" Inventory of the Parent Radionuclide

Radionuclide[a]	Peak Concentration pCi/L-Ci	Peak Time Years
^{3}H	1.40×10^{-19}	771
^{14}C	2.50×10^{-1}	1064
^{59}Ni	2.53×10^{-7}	9999
^{79}Se	1.04	4858
$^{90}Sr+d$	4.69×10^{-15}	1121
^{93}Mo	7.63×10^{-1}	1906
^{93}Zr	2.60×10^{-11}	10000
^{93m}Nb	1.14×10^{-10}	10000
^{94}Nb	1.05×10^{-1}	10000
^{99}Tc	8.52×10^{-1}	874
$^{126}Sn+d$	6.17×10^{-1}	10000
^{129}I	1.10	2628
^{135}Cs	3.20×10^{-5}	10000
^{232}Th	2.26×10^{-30}	10000
^{228}Ra	1.69×10^{-29}	10000
^{228}Th	2.64×10^{-30}	10000
^{224}Ra	1.69×10^{-29}	10000
^{232}U	9.98×10^{-14}	2178
^{228}Th	1.10×10^{-15}	2178
^{224}Ra	7.02×10^{-15}	2178
^{234}U	7.90×10^{-2}	10000
^{230}Th	3.08×10^{-5}	10000
^{226}Ra	1.16×10^{-4}	10000
^{210}Pb	2.13×10^{-4}	10000
^{210}Po	3.83×10^{-4}	10000
^{235}U	8.13×10^{-2}	10000
^{231}Pa	4.42×10^{-4}	10000
^{227}Ac	5.35×10^{-4}	10000
^{227}Th	7.52×10^{-5}	10000
^{223}Ra	4.82×10^{-4}	10000
^{236}U	8.12×10^{-2}	10000
^{238}U	8.13×10^{-2}	10000
^{234}Th	8.91×10^{-4}	10000
^{234}U	2.22×10^{-3}	10000
^{237}Np	6.58×10^{-1}	10000
^{238}Pu	5.94×10^{-19}	3224
^{234}U	5.26×10^{-6}	10000
^{239}Pu	1.34×10^{-2}	10000

Table 3.3-6. Predicted Groundwater Peak Concentrations Based on 1.0 Ci Initial "Activation" Inventory of the Parent Radionuclide (cont.)

Radionuclide[a]	Peak Concentration pCi/L-Ci	Peak Time Years
^{235}U	2.52×10^{-7}	10000
^{240}Pu	6.29×10^{-3}	10000
^{236}U	4.71×10^{-6}	10000
^{241}Pu	1.38×10^{-50}	1474
^{241}Am	2.95×10^{-33}	10000
^{237}Np	9.45×10^{-7}	10000
^{242}Pu	1.74×10^{-2}	10000
^{238}U	4.74×10^{-8}	10000
^{244}Pu	1.77×10^{-2}	10000
^{241}Am	5.57×10^{-31}	10000
^{237}Np	1.33×10^{-4}	10000
^{243}Am	1.66×10^{-24}	10000
^{239}Np	6.20×10^{-22}	10000
^{239}Pu	1.06×10^{-2}	10000
^{244}Cm	9.91×10^{-92}	1706
^{240}Pu	1.12×10^{-4}	10000
^{236}U	7.41×10^{-8}	10000
^{245}Cm	8.46×10^{-34}	10000
^{241}Pu	1.00×10^{-26}	10000
^{241}Am	3.64×10^{-27}	10000
^{237}Np	1.58×10^{-3}	10000
^{246}Cm	4.48×10^{-34}	10000
^{247}Cm	1.89×10^{-33}	10000
^{243}Am	3.12×10^{-25}	10000
^{239}Np	1.17×10^{-22}	10000
^{239}Pu	2.88×10^{-3}	10000
^{248}Cm	1.86×10^{-33}	10000
^{244}Pu	4.77×10^{-6}	10000
^{249}Cf	3.05×10^{-18}	10000
^{245}Cm	5.58×10^{-20}	10000
^{241}Pu	3.09×10^{-18}	10000
^{241}Am	1.48×10^{-19}	10000
^{237}Np	6.31×10^{-5}	10000

[a]Indented entries are potentially radiologically-significant decay products of parent radionuclides.

To develop total peak groundwater concentrations attributable to both the "crud" waste and "activation" waste, the normalized concentrations as a function of time (up to 10,000 years after closure) for each type of waste are multiplied by the respective "crud" or "activation" projected inventories for each radionuclide. This results in two curves (concentration as a function of time) for each radionuclide: one for "crud" waste and one for "activation" waste. The two curves are then summed to obtain a total groundwater concentration curve, from which the peak concentration is then determined. The peak groundwater concentrations, in pCi/L, based on total projected "crud" and "activation" inventory for each radionuclide, are given in Table 3.3-7.

Table 3.3-7. Predicted Groundwater Peak Concentrations Based on Total Projected "Crud" and "Activation" Inventory of the Parent Radionuclide

Radionuclide[a]	Peak Concentration pCi/L	Peak Time Years
^{3}H	4.76×10^{-17}	771
^{14}C	8.93×10^{2}	905
^{59}Ni	9.93×10^{-4}	10000
^{79}Se	3.20×10^{-3}	4861
$^{90}Sr+d$	4.71×10^{-13}	1123
^{93}Mo	2.74	1871
^{93}Zr	4.87×10^{-7}	10000
^{93m}Nb	2.13×10^{-6}	10000
^{94}Nb	1.85	10000
^{99}Tc	3.72	791
$^{126}Sn+d$	1.76×10^{-3}	10000
^{129}I	1.05×10^{-2}	808
^{135}Cs	2.77×10^{-9}	10000
^{232}Th	2.13×10^{-38}	10000
^{228}Ra	1.59×10^{-37}	10000
^{228}Th	2.48×10^{-38}	10000
^{224}Ra	1.59×10^{-37}	10000
^{232}U	2.86×10^{-17}	2029
^{228}Th	3.14×10^{-19}	2033
^{224}Ra	2.01×10^{-18}	2033
^{234}U	5.49×10^{-5}	10000
^{230}Th	2.14×10^{-8}	10000
^{226}Ra	8.07×10^{-8}	10000
^{210}Pb	1.44×10^{-7}	9918
^{210}Po	2.59×10^{-7}	9918
^{235}U	4.18×10^{-8}	10000
^{231}Pa	2.28×10^{-10}	10000
^{227}Ac	2.75×10^{-10}	10000
^{227}Th	3.87×10^{-11}	10000
^{223}Ra	2.48×10^{-10}	10000
^{236}U	8.61×10^{-5}	10000
^{238}U	4.74×10^{-6}	10000
^{234}Th	5.20×10^{-8}	10000
^{234}U	1.30×10^{-7}	10000
^{237}Np	6.71×10^{-6}	10000
^{238}Pu	5.28×10^{-19}	3179
^{234}U	3.63×10^{-6}	10000
^{239}Pu	4.23×10^{-3}	10000

Table 3.3-7. Predicted Groundwater Peak Concentrations Based on Total Projected "Crud" and "Activation" Inventory of the Parent Radionuclide (cont.)

Radionuclide[a]	Peak Concentration pCi/L	Peak Time Years
^{235}U	7.89×10^{-8}	10000
^{240}Pu	1.77×10^{-3}	9329
^{236}U	1.32×10^{-6}	10000
^{241}Pu	1.18×10^{-48}	1475
^{241}Am	2.67×10^{-31}	10000
^{237}Np	8.10×10^{-5}	10000
^{242}Pu	1.80×10^{-5}	9906
^{238}U	4.96×10^{-11}	10000
^{244}Pu	1.32×10^{-12}	9906
^{241}Am	5.05×10^{-31}	10000
^{237}Np	1.18×10^{-4}	10000
^{243}Am	1.05×10^{-26}	10000
^{239}Np	3.92×10^{-24}	10000
^{239}Pu	6.45×10^{-5}	10000
^{244}Cm	4.91×10^{-92}	1684
^{240}Pu	5.46×10^{-5}	10000
^{236}U	3.58×10^{-8}	10000
^{245}Cm	2.37×10^{-38}	10000
^{241}Pu	2.65×10^{-31}	10000
^{241}Am	9.66×10^{-32}	10000
^{237}Np	4.07×10^{-8}	10000
^{246}Cm	4.84×10^{-39}	10000
^{247}Cm	4.31×10^{-44}	10000
^{243}Am	7.02×10^{-36}	10000
^{239}Np	2.62×10^{-33}	10000
^{239}Pu	5.94×10^{-14}	10000
^{248}Cm	1.04×10^{-43}	10000
^{244}Pu	2.34×10^{-16}	10000
^{249}Cf	1.01×10^{-27}	9906
^{245}Cm	1.81×10^{-29}	9906
^{241}Pu	1.08×10^{-27}	10000
^{241}Am	5.20×10^{-29}	10000
^{237}Np	1.99×10^{-14}	10000

[a]Indented entries are potentially radiologically-significant decay products of parent radionuclides.

4.0 RESULTS OF ANALYSIS

In this section, the results of the analysis of performance of the Naval Reactor Waste Disposal Pad in the E-Area LLWF are presented. As described in Section 3.2, the analysis focused on ingestion of radionuclides in groundwater for the all-pathways and water resource protection analysis, and on inhalation of volatile radionuclides for the air pathways analysis. In Section 4.1, inventory limits used in evaluating performance with respect to the performance objective for protection of the public and the assessment requirement for water resources (Section 1.3) are presented. These inventory limits are based on normalized groundwater concentrations presented in Tables 3.3-5 and 3.3-6. In Section 4.2, calculated doses for the air pathway analysis are provided.

4.1 Results of All-Pathways and Water Resource Impacts Analyses

It was established in Section 3.2 that the only significant pathway of concern for the all-pathways analysis is groundwater ingestion, with the endpoint of the analysis being evaluation of dose. For the water resource protection analysis, the endpoint of the analysis is estimated groundwater concentrations at the same compliance point as that used in evaluating protection of the public; the point of highest concentration outside a 100-m buffer zone around disposal units. The performance objective of 25 mrem/yr for protection of the public can readily be converted to an allowable groundwater concentration limit, by dividing the 25 mrem/yr limit by the EDE of any given radionuclide per unit concentration in groundwater (Table 4.1-1). Thus, the endpoint of the all-pathways analysis can be converted to a groundwater concentration, and the endpoints of both of these analyses can be expressed in terms of groundwater concentration. The following discussion addresses the results of the all-pathways analysis and water resource protection analysis together, as the same predicted groundwater concentrations can be used to evaluate the performance objective for protection of the public and the assessment requirement for water resources.

The methods for predicting groundwater concentrations at the compliance point for protection of the public and water resources were described in Section 3.3. Briefly, groundwater concentrations at the compliance point were predicted by using the vadose zone models (simulated with PORFLOW) to predict contaminant flux to the water table as a function of time. Fractional fluxes (Ci/yr per Ci disposed) were estimated, and used as the source term to the saturated zone model. Groundwater concentrations (pCi/L per Ci disposed) were then calculated using the PORFLOW computer code. These concentrations were calculated for two different source terms – that originating from leaching of “crud” waste (metal corrosion products) present in the waste originally, and that originating from the corrosion of “activation” waste, and subsequent leaching of the corrosion products.

The total peak groundwater concentrations (pCi/L) presented in Table 3.3-7 of the previous section represent the contribution from both “crud” and “activation” waste. In order to derive inventory limits, the total groundwater concentration for each radionuclide is normalized to the total activity of each radionuclide by dividing this peak concentration by the sum of the “crud” and “activation” inventories for each radionuclide (Table 2.4-2). The resulting normalized peak groundwater concentration has units of pCi/L-Ci.

Table 4.1-1. Annual EDE's for Drinking Water Pathway per Unit Concentration of Radionuclides in Water

Radionuclide[a,b]	EDE rem/yr per μCi/L
^{3}H	4.6×10^{-2}
^{14}C	1.5
^{59}Ni	1.5×10^{-1}
^{79}Se	6.1
^{90}Sr+d	1.0×10^{2}
^{93}Mo	9.5×10^{-1}
^{93}Zr+d	1.6
^{94}Nb	3.7
^{99}Tc	9.5×10^{-1}
^{126}Sn+d	1.3×10^{1}
^{129}I	2.0×10^{2}
^{135}Cs	5.2
^{230}Th	3.9×10^{2}
^{226}Ra+d	8.0×10^{2}
^{210}Pb	3.7×10^{3}
^{210}Po	1.2×10^{3}
^{232}Th	2.0×10^{3}
^{228}Ra+d	8.8×10^{2}
^{228}Th	2.8×10^{2}
^{224}Ra+d	2.7×10^{2}
^{232}U	9.5×10^{2}
^{228}Th	2.8×10^{2}
^{224}Ra+d	2.7×10^{2}
^{234}U	1.9×10^{2}
^{230}Th	3.9×10^{2}
^{226}Ra+d	8.0×10^{2}
^{210}Pb	3.7×10^{3}
^{210}Po	1.2×10^{3}
^{235}U+d	1.8×10^{2}
^{231}Pa	8.0×10^{3}
^{227}Ac	1.0×10^{4}
^{227}Th	2.6×10^{1}
^{223}Ra+d	4.0×10^{2}
^{236}U	1.8×10^{2}
^{238}U	1.7×10^{2}
^{234}Th+d	9.5
^{237}Np+d	2.8×10^{3}
^{238}Pu	2.8×10^{3}
^{234}U	1.9×10^{2}
^{239}Pu	3.1×10^{3}

Table 4.1-1. Annual EDE's for Drinking Water Pathway per Unit Concentration of Radionuclides in Water (cont.)

Radionuclide[a,b]	EDE rem/yr per μCi/L
^{235}U	1.8×10^{2}
^{240}Pu	3.1×10^{3}
^{236}U	1.8×10^{2}
^{241}Pu	6.3×10^{1}
^{241}Am	3.3×10^{3}
^{237}Np+d	2.8×10^{3}
^{242}Pu	3.0×10^{3}
^{238}U	1.7×10^{2}
^{244}Pu+d	2.9×10^{3}
^{241}Am	3.3×10^{3}
^{237}Np+d	2.8×10^{3}
^{242m}Am+d	3.1×10^{3}
^{238}Pu	2.8×10^{3}
^{234}U	1.9×10^{2}
^{243}Am+d	3.3×10^{3}
^{239}Pu	3.1×10^{3}
^{244}Cm	1.7×10^{3}
^{240}Pu	3.1×10^{3}
^{236}U	1.8×10^{2}
^{245}Cm	3.3×10^{3}
^{241}Pu	6.3×10^{1}
^{241}Am	3.3×10^{3}
^{237}Np+d	2.8×10^{3}
^{246}Cm	3.3×10^{3}
^{247}Cm+d	3.0×10^{3}
^{243}Am	3.3×10^{3}
^{239}Np	2.1
^{239}Pu	3.1×10^{3}
^{248}Cm	1.2×10^{4}
^{244}Pu	2.7×10^{3}
^{249}Cf	3.4×10^{3}
^{245}Cm	3.3×10^{3}
^{241}Pu	6.3×10^{1}
^{241}Am	3.3×10^{3}
^{237}Np+d	2.8×10^{3}

[a] Indented entries are decay products of parent radionuclide that are potentially radiologically-significant.
[b] The notation "+d" indicates that short-lived daughters were incorporated into the EDE. Dose from other daughters listed is calculated explicitly by adding EDE's.

The more restrictive of either the MCLs (in pCi/L) or allowable concentrations (in pCi/L) based on the 25 mrem/yr performance objective (Table 3.3-1) can be divided by the normalized total groundwater concentrations (in pCi/L-Ci) to derive an inventory limit. The resulting inventory limits are not only specific to each radionuclide, but also specific to the assumed distribution of "crud" and "activation" activity (Table 2.4-2) for each radionuclide. Calculated inventory limits, based on the peak total groundwater concentrations, are presented in Table 4.1-2 of this section.

4.2 Results of Air Pathway Analysis

Table 4.2-1 presents the results of the air pathway analysis described in Sections 3.1.4 and 3.2.2. Inventory limits based on the air pathway are derived in the following manner. Dose factors in Table 3.3-2 are multiplied by the normalized fractional releases discussed in Section 3.1.4 to obtain dose rate per Ci inventory (mrem/yr-Ci). Inventory limits are then calculated by dividing the performance objective for the air pathway of 10 mrem/yr (Table 1.3-1) by the dose rate per Ci inventory.

For ^{3}H, both the "crud" and "activation" inventories are conservatively assumed to be in a volatile form, either at closure of the facility, when the compliance point is the SRS boundary, or at 100 years post closure, when the compliance point is 100 m from the E-Area LLWF boundary. At closure of the facility, it is assumed that none of the projected inventory of ^{3}H has decayed radiologically, giving a dose rate per Ci inventory at the site boundary of 2.4×10^{-6} mrem/yr-Ci, and a corresponding inventory limit of 4.2×10^{6} Ci. At 100 years post closure, 0.4% of the original inventory remains after radiological decay, and thus the normalized dose rate and corresponding inventory limit for ^{3}H are calculated to be $(0.004)(8.5 \times 10^{-3})$, or 3.4×10^{-5} mrem/yr-Ci, and 2.9×10^{5} Ci, respectively.

For ^{14}C, the "crud" inventory is conservatively assumed to be only partially volatile, as described in Section 3.1.4. Thus the dose rate per Ci "crud" inventory is $(0.005/yr)(1.0 \times 10^{-3}$ mrem/Ci), or 5.0×10^{-6} mrem/yr-Ci at the site boundary at closure. The dose rate per Ci "crud" inventory is (0.5)(0.005/yr)(3.8 mrem/Ci), or 9.5×10^{-3} mrem/yr-Ci, at 100 m from the E-Area LLWF boundary at 100 years, where 0.5 represents the remaining fraction of ^{14}C after loss due to volatilization and radioactive decay in the first 100 years. For the activated metal inventory of ^{14}C, the dose rate per Ci "activation" inventory is $(3 \times 10^{-5}/yr)(1.0 \times 10^{-3}$ mrem/Ci), or 3.0×10^{-8} mrem/yr-Ci at the site boundary at closure. One hundred years after closure, the dose rate per Ci "activation" inventory is (0.99) $(3 \times 10^{-5}/yr)$ (3.8 mrem/Ci), or 1.1×10^{-4} mrem/yr-Ci at 100 m from the E-Area LLWF boundary, where 0.99 is the fraction of ^{14}C remaining at 100 years after loss due to decay. The total normalized dose rate for ^{14}C at each time and point of compliance is calculated by multiplying the dose rate per Ci inventory of "crud" by the projected "crud" inventory, adding that to the dose rate per Ci inventory of "activation" multiplied by the projected "activation" inventory, and then dividing the sum by the total projected inventory (summed "crud" plus "activation" inventory). At the site boundary at closure, the total normalized ^{14}C dose rate is calculated to be 5.6×10^{-8} mrem/yr-Ci, and the corresponding inventory limit is 1.8×10^{8} Ci. At 100 m from the E-Area LLWF boundary, at 100 years post closure, the total normalized ^{14}C dose rate is calculated to be 1.6×10^{-4} mrem/yr-Ci, and the corresponding inventory limit is 6.3×10^{4} Ci.

Table 4.1-2. Calculated Inventory Limits for the Naval Reactor Waste Disposal Pad

Radionuclide[a,b]	Concentration Limit[c] pCi/L	Calculated Inventory Limit[d] Ci/Pad
^{3}H	20,000	$>1 \times 10^{20}$
^{14}C	2,000	7.7×10^{2}
^{59}Ni	300	1.2×10^{9}
^{79}Se	700	6.8×10^{2}
^{90}Sr+d	8	2.2×10^{14}
^{93}Mo	4,000	5.2×10^{3}
^{93}Zr+d	2,000	8.8×10^{12}
^{94}Nb	1,000	8.7×10^{3}
^{99}Tc	900	8.8×10^{1}
^{126}Sn+d	300	3.2×10^{2}
^{129}I	1	9.0×10^{-4}
^{135}Cs	900	2.8×10^{7}
^{232}Th	13	$>1 \times 10^{20}$
^{228}Ra	5	[d]
^{228}Th	15	[d]
^{224}Ra	15	[d]
^{232}U	26	2.5×10^{12}
^{228}Th	15	[d]
^{224}Ra	15	[d]
^{234}U	130	1.7×10^{3}
^{230}Th	15	[d]
^{226}Ra	5	[d]
^{210}Pb	1	[d]
^{210}Po	15	[d]
^{235}U	65	8.1×10^{2}
^{231}Pa	3.1	[d]
^{227}Ac	1	[d]
^{227}Th	15	[d]
^{223}Ra	15	[d]
^{236}U	140	1.8×10^{3}
^{238}U	10	1.2×10^{2}
^{234}Th	400	[d]
^{234}U	130	[d]
^{237}Np	8.9	1.3×10^{1}
^{238}Pu	8.9	2.4×10^{7}
^{234}U	130	[d]
^{239}Pu	8.1	5.9×10^{2}
^{235}U	65	[d]
^{240}Pu	8.1	1.3×10^{3}
^{236}U	140	[d]
^{241}Pu	300	9.3×10^{6}
^{241}Am	7.6	[d]

Table 4.1-2. Calculated Inventory Limits for the Naval Reactor Waste Disposal Pad (cont.)

Radionuclide[a,b]	Concentration Limit[c] pCi/L	Calculated Inventory Limit[d] Ci/Pad
^{237}Np	8.9	[d]
^{242}Pu	8.3	4.6×10^{2}
^{238}U	10	[d]
^{244}Pu	8.6	4.5×10^{2}
^{241}Am	7.6	6.6×10^{4}
^{237}Np	8.9	[d]
^{243}Am	7.6	7.5×10^{2}
^{239}Np	300	[d]
^{239}Pu	8.1	[d]
^{244}Cm	15	7.1×10^{4}
^{240}Pu	8.1	[d]
^{236}U	140	[d]
^{245}Cm	7.6	5.7×10^{3}
^{241}Pu	300	[d]
^{241}Am	7.6	[d]
^{237}Np	8.9	[d]
^{246}Cm	7.6	$>1 \times 10^{20}$
^{247}Cm	8.3	2.7×10^{3}
^{243}Am	7.6	[d]
^{239}Np	300	[d]
^{239}Pu	8.1	[d]
^{248}Cm	2.1	1.7×10^{6}
^{244}Pu	8.6	[d]
^{249}Cf	7.4	1.4×10^{5}
^{245}Cm	7.6	[d]
^{241}Pu	300	[d]
^{241}Am	7.6	[d]
^{237}Np	8.9	[d]

[a] "+d" indicates short- and/or long-lived radioactive daughters are considered in the analysis.
[b] Indented entries are decay products of parent radionuclides that are potentially significant.
[c] The more restrictive of either the MCL or the allowable concentration based on a 25 mrem/yr performance objective (Table 3.3-1)
[d] Calculated by dividing the "Concentration limit" by the "Peak normalized groundwater concentration". For radionuclides with potentially significant daughters listed, the lower limit calculated by this method for all radionuclides in the decay chain is the one reported as the inventory limit for the parent radionuclide.

Table 4.2-1. Results of Air Pathway Analysis

Radionuclide	Inventory Limit Based on Location at SRS Boundary before 100-Year Institutional Control Period[a] (Ci/Pad)	Inventory Limit Based on Location 100 Meters from E-Area LLWF after 100-Year Institutional Control Period[b] (Ci/Pad)	Inventory Limit Based on Air Pathway (Ci/Pad)
^{3}H	4.2×10^{6}	2.9×10^{5}	2.9×10^{5}
^{14}C	1.8×10^{8}	6.3×10^{4}	6.3×10^{4}

[a] Calculated at 0 years post closure, as this minimizes the limit by minimizing radiological decay.
[b] Calculated at 100 years post closure, as this minimizes the limit by minimizing radiological decay.

THIS PAGE INTENTIONALLY LEFT BLANK

5.0 INADVERTENT INTRUDERS

As described in Section 1.3, disposal of low-level radioactive waste in the E-Area LLWF must meet a performance measure for protection of inadvertent intruders onto the disposal site. In particular, after loss of active institutional control at 100 years after facility closure, the EDE to an intruder should not exceed 100 mrem per year for scenarios involving continuous exposure or 500 mrem for scenarios involving a single acute exposure (Section 1.3). These dose limits apply to the sum of dose equivalents from all exposure pathways that are assumed to occur in a given exposure scenario for an inadvertent intruder.

The time period covered by the intruder analysis for this performance analysis for the Naval Reactor Waste Disposal Pad extends beyond 1,000 years after loss of institutional control, as it did in the E-Area LLWF PA. Some intruder scenarios cannot reasonably occur within the 1,000-year time period put forth as the compliance period for PA in USDOE's interim guidance (Section 1.3), and ingrowth of radioactive daughters for some radionuclides may be more significant to dose beyond this time period. Therefore, analytical results for the first 10,000 years after loss of active institutional control (and beyond for a few radionuclides) are used to evaluate performance of the Naval Reactor Waste Disposal Pad with respect to inadvertent intruders.

In Section 5.1 below, inadvertent intruder exposure scenarios selected for analysis are summarized. The selection of scenarios is based on a review of the intruder scenarios described in the E-Area LLWF PA. This review indicates that the dominant scenarios for an at-grade Naval Reactor Waste Disposal Pad are the same as those for a below-grade pad, which was evaluated in the E-Area LLWF PA, due the fact that the planned cover depth and design is identical for both modes of disposal. Access to the Naval Reactor disposal casks by an inadvertent intruder is not expected to occur earlier in the present analysis.

Screening of the list of radionuclides projected for disposal in the Naval Reactor Waste Disposal Pad (Table 2.4-1) is described in Section 5.2. Screening allows for selection of radionuclides of potential significance to estimate inadvertent intruder dose. Radionuclides not considered potentially significant are those that decay to insignificant levels before human exposures may occur.

Finally, in Section 5.3, analysis of dose to inadvertent intruders for the potentially significant scenarios and radionuclides is described. Results of the dose calculations are also presented in that section.

5.1 Exposure Scenarios for Inadvertent Intruders

The focus in development of exposure scenarios for inadvertent intruders is on selecting reasonable events that may occur, giving consideration to regional customs and construction practices. An important assumption in all scenarios is that an intruder has no prior knowledge of the existence of a waste disposal facility at the site. Therefore, after active institutional control ceases, certain exposure scenarios are assumed to be precluded only by the physical state of the disposal facility, i.e., the integrity of the engineered barriers used in facility construction. Passive institutional controls, such as permanent marker systems at the disposal site and public records of prior land use, also could prevent inadvertent intrusion after active institutional control ceases, but the use of passive institutional controls is not assumed in this analysis.

Intruder exposure scenarios described below do not include consumption of groundwater and crop irrigation with groundwater because impacts associated with these exposure routes are evaluated

separately in the water resource protection analysis. A more in-depth justification for this neglect is provided in Section 6.1.1 of the E-Area LLWF PA. Pathways of exposure to volatile forms of ^{3}H and ^{14}C, are considered separately in the air pathway analysis (Section 3.2.2).

Several chronic and acute exposure scenarios for inadvertent intruders were initially considered for use in the E-Area LLWF PA (McDowell-Boyer et al. 2000). However, analyses of how these scenarios would apply to the E-Area LLWF indicated that only three chronic exposure scenarios need to be included in the PA:

- an agriculture scenario involving direct intrusion into disposal units at times after the engineered barriers above the waste have lost their structural and physical integrity and can be penetrated by the types of excavation procedures normally used at the SRS;
- a resident scenario involving permanent residence in a home located either on top of an intact concrete roof or other engineered barrier, which first could occur upon loss of active institutional control at 100 years after facility closure, or on top of intact but essentially exposed waste at times after the engineered barriers have lost their integrity; and
- a post-drilling scenario involving exhumation of waste from a disposal unit at times after drilling through a disposal unit becomes credible.

In Section 6.3.3 of the E-Area LLWF PA, it is indicated that the post-drilling scenario is potentially relevant for any disposal unit for which drilling into the waste may occur before the agriculture scenario becomes credible. As noted in the Section 6.1.1.5 of the E-Area PA, previous analyses of the agriculture and post-drilling scenarios have shown that the dose to an intruder per unit concentration of radionuclides in excavated material should be considerably greater for the agriculture scenario than for the post-drilling scenario, provided the assumptions for the exposure pathways in the two scenarios are reasonably consistent. The principal reasons for the greater doses in the agriculture scenario are 1) the greater volume of waste exhumed during construction of a foundation for a home compared with the volume of waste exhumed during drilling of a well, which results in greater concentrations of radionuclides in contaminated soil in the intruder's vegetable garden, and 2) the doses from external and inhalation exposure while residing in a home on the disposal site, which contribute to the dose for the agriculture scenario but are not relevant for the post-drilling scenario. For the Naval Reactor Waste Disposal Pad, there is no reason to believe that drilling may occur before agriculture scenario becomes credible; thus, the post-drilling scenario is not relevant for this analysis.

5.2 Screening of Radionuclides for Intruder Dose Analysis

Many radionuclides decay significantly before an inadvertent intruder may come into contact with them according to the exposure scenarios described in Section 5.1. For these radionuclides, calculated allowable inventories greatly exceed any imaginable inventory in the Naval Reactor Waste Disposal Pad. Screening calculations were therefore carried out on a large suite of 730 radionuclides that may be encountered during disposal operations, in order to eliminate radionuclides that are negligible contributors to dose from further analysis. These calculations are the same calculations carried out for the E-Area LLWF PA, and described in Section 6.2 of that document (McDowell-Boyer et al. 2000). The list of 730 radionuclides was selected because it represents all radionuclides having published DCFs (USDOE 1988).

A subset of these 730 radionuclides are listed in the projected inventory (Table 2.4-1) for the Naval Reactor Waste Disposal Pad. This section presents the results of the simple screening analysis for selecting the radionuclides of potential concern for protection of inadvertent intruders.

The lists of radionuclides in the Naval Reactor Waste Disposal Pad disposal facility of potential significance with respect to inadvertent intruders, including those with potentially significant daughters noted by "+d", are as follows: ^{3}H, ^{14}C, ^{59}Ni, ^{60}Co, ^{63}Ni, ^{79}Se, ^{90}Sr+d, ^{93}Mo, ^{93}Zr+d, ^{94}Nb, ^{99}Tc, ^{126}Sn+d, ^{129}I, ^{135}Cs, ^{137}Cs+d, ^{151}Sm, ^{154}Eu, ^{232}Th+d, ^{232}U+d, ^{234}U+d, ^{235}U+d, ^{236}U, ^{237}Np, ^{238}U+d, ^{238}Pu+d, ^{239}Pu+d, ^{240}Pu+d, ^{241}Pu+d, ^{242}Pu+d, ^{244}Pu, ^{241}Am+d, ^{242m}Am+d, ^{243}Am+d, ^{243}Cm, ^{244}Cm+d, ^{245}Cm+d, ^{246}Cm, ^{247}Cm+d, ^{248}Cm+d, ^{249}Cf+d, and ^{251}Cf.

5.3 Dose Analysis for Inadvertent Intruders

In this section of the Naval Reactor Waste Disposal Pad analysis, radionuclide-specific doses to inadvertent intruders and inventory limits for chronic exposures are calculated. Acute exposure scenarios are not evaluated, as chronic exposure scenarios are shown, in Section 6.1 of the E-Area LLWF PA (McDowell-Boyer et al. 2000), to be the dose-limiting scenarios for all E-Area LLWF disposal facilities. Dose calculations are based on the total projected inventory given in Table 2.4-2. Inventory limits are based on the EDE limit of 100 mrem/yr, as required for compliance with the performance measure for protection of inadvertent intruders (Section 1.3). By comparing radionuclide-specific doses to the 100-mrem/yr limit, or inventory limits to projected inventory for the pad, compliance with the performance measure can be determined.

As described in Section 1.3, compliance with the performance measure for protection of inadvertent intruders is assumed to be required for 1,000 years after disposal. However, because maximum dose to an inadvertent intruder is often predicted to occur beyond 1,000 years after disposal, the calculations are carried out for 10,000 years to provide additional information and perspective on the performance of the Naval Reactor Waste Disposal Pad. Consideration of potential doses beyond 1,000 years is particularly important because one of the two credible scenarios for this facility, the agriculture scenario, cannot reasonably occur for approximately 10,000 years due to the slow corrosion rate of the disposal casks.

In Section 5.1, it is established that the two exposure scenarios of concern for this analysis are the resident scenario and the agriculture scenario. The agriculture scenario involves direct intrusion into disposal units at any time after the concrete vaults and any other engineered barriers above the waste have lost their structural and physical integrity and excavation into the waste becomes credible. The resident scenario involves permanent residence in a home located immediately above an intact concrete roof or other engineered barrier at any time after loss of active institutional control. The exposure pathways of concern for these two scenarios are described in Section 6.1 of the E-Area LLWF PA. Doses to inadvertent intruders resulting from use of contaminated groundwater obtained from a well on the disposal site are evaluated separately, in the water resource protection analysis, and are not calculated for the intruder, in accordance with the reasoning put forth in the Format and Content Guide for PAs (USDOE 1996).

The models for estimating dose for the two chronic exposure scenarios for inadvertent intruders considered in this analysis are presented in Appendix C.3 of the E-Area LLWF PA. In this appendix, scenario dose conversion factors (SDCF) were developed that estimate EDE per unit concentration in the waste zone (see Figure 4.2). In the PA, the doses for the agriculture scenario conservatively assumed that all of the activity remaining in the waste zone at the time the scenario was deemed credible was in an available form, and not tied up in activated metal. Thus, there was one SDCF for the agriculture scenario, and one SDCF for the resident scenario. In the present analysis, the amount of "available" (i.e., corrosion product) activity and total activity were tracked separately, so that pathways requiring available activity could be evaluated more accurately. Therefore, in this analysis, there is one SDCF for the internal exposure pathways in the agriculture scenario (i.e., ingestion of vegetables, ingestion of soil, and inhalation) that require available radionuclides, and a second SDCF

for the agriculture scenario for the external exposure pathways that apply to the total inventory of radionuclides, regardless of form. For the resident scenario, one SDCF exists as external exposure is the only route of exposure defined for this scenario. The SDCF's for the resident scenario are used for screening only in this analysis (see Section 5.3.2); thus, shielding provided by the waste and disposal casks is conservatively ignored. The SDCF's developed according to the procedures described in Appendix C.3 of the E-Area LLWF PA are given in Tables 5.3-1 and 5.3-2 for the agriculture and resident scenarios, respectively.

To calculate dose from the SDCF's, the maximum concentrations of radionuclides in the disposal facility at any time after the scenario is first assumed to be credible must be specified. The concentrations of radionuclides in disposal units over time are estimated using the initial concentrations at disposal corrected for radioactive decay. Depletion of radionuclide inventories in disposal units due to corrosion of activated waste, and removal of "available" radionuclides by infiltrating water also is considered in most cases.

Ingrowth of radiologically-significant decay products at times long after disposal is potentially important in the intruder dose analysis, and is accounted for in the analysis. The calculation of radionuclide inventory and concentration limits in disposal facilities on the basis of an initial unit activity of each radionuclide in disposed waste is described below.

For an assumed exposure scenario and particular type of disposal unit, the EDE (rem/yr) to an inadvertent intruder from exposure to a given radionuclide can be expressed as

$$H = (I_0/V) \times SDCF \times G \times F , \qquad \text{(Eq. 5.3-1)}$$

where

I_0	=	inventory of radionuclide in disposal units at time of disposal (μCi),
V	=	volume of waste in Naval Reactor Waste Disposal Pad (m^3),
SDCF	=	scenario dose conversion factor for radionuclide (rem/yr per μCi/m^3),
G	=	geometrical correction factor (dimensionless), and
F	=	fraction of initial inventory of radionuclide remaining in disposal units at time intrusion occurs.

The volume of waste in the Naval Reactor Waste Disposal Pad is considered to be the volume of the projected 100 waste disposal casks (43 m^3 each). The correction factors G and F take into account the fact that the average radionuclide concentration encountered by an inadvertent intruder would be less than the initial average concentration in disposed waste and that the dose would be correspondingly reduced.

The geometrical correction factor, *G*, in Eq. 5.3-1 takes into account that a large-scale excavation into the Naval Reactor Waste Disposal Pad, as assumed in the agriculture and resident scenarios, would involve exposure to uncontaminated material between individual disposal units and disposed waste itself. This correction factor assumes that excavation into disposal units would occur at random locations. Therefore, the geometrical correction factor is given by the fraction of the land area encompassed by the disposal units of a particular type that contains waste. The assumed value of the geometrical correction factor for the agriculture and resident scenarios in this analysis is 0.08. The correction factor was estimated by dividing the area corresponding to the disposal casks by the area of the total facility. The area taken up by disposal casks is approximately 800 m^2, as the diameter of each cask is 3.2 m (Figure 2.3-1). The total land area of the Naval Reactor Waste Disposal Pad is 11,000 m^2 (Section 2.2.1).

Table 5.3-1. Scenario Dose Conversion Factors for all Exposure Pathways for the Agriculture Scenario

Radionuclide[a,b]	EDE for Internal Exposure Pathways (rem/yr per μCi/m^3)	EDE for External Exposure Pathways (rem/yr per μCi/m^3)
^{3}H	3.9×10^{-6}	0.0
^{14}C	1.5×10^{-5}	3.0×10^{-9}
^{60}Co	1.1×10^{-6}	3.5×10^{-3}
^{59}Ni	6.8×10^{-8}	0.0
^{63}Ni	1.8×10^{-7}	0.0
^{79}Se	1.2×10^{-6}	4.2×10^{-9}
^{90}Sr+d	1.8×10^{-4}	5.4×10^{-6}
^{93}Mo	1.7×10^{-6}	1.3×10^{-7}
^{93}Zr+d	4.1×10^{-8}	2.3×10^{-8}
^{94}Nb	1.8×10^{-7}	2.1×10^{-3}
^{99}Tc	1.1×10^{-5}	2.7×10^{-8}
^{126}Sn+d	8.9×10^{-7}	2.6×10^{-3}
^{129}I	8.1×10^{-5}	2.9×10^{-6}
^{135}Cs	1.2×10^{-6}	8.4×10^{-9}
^{137}Cs+d	8.6×10^{-6}	7.7×10^{-4}
^{151}Sm	1.0×10^{-8}	2.1×10^{-10}
^{154}Eu	2.5×10^{-7}	1.7×10^{-3}
^{232}Th	4.9×10^{-5}	1.2×10^{-7}
^{228}Ra+d	1.7×10^{-5}	1.3×10^{-3}
^{228}Th	1.1×10^{-5}	1.8×10^{-6}
^{224}Ra+d	6.9×10^{-6}	2.2×10^{-3}
^{232}U	5.5×10^{-5}	2.0×10^{-7}
^{228}Th	1.1×10^{-5}	1.8×10^{-6}
^{224}Ra+d	6.9×10^{-6}	2.2×10^{-3}
^{234}U	1.1×10^{-5}	8.8×10^{-8}
^{230}Th	1.1×10^{-5}	2.6×10^{-7}
^{226}Ra+d	1.5×10^{-5}	2.4×10^{-3}
^{210}Pb	2.8×10^{-4}	5.3×10^{-7}
^{210}Po	1.2×10^{-5}	1.2×10^{-8}
^{235}U+d	1.0×10^{-5}	1.6×10^{-4}
^{231}Pa	1.1×10^{-4}	4.2×10^{-5}
^{227}Ac	3.0×10^{-4}	1.1×10^{-7}
^{227}Th	6.4×10^{-7}	1.1×10^{-4}
^{223}Ra+d	7.7×10^{-6}	2.6×10^{-4}
^{236}U	1.0×10^{-5}	4.6×10^{-8}
^{238}U	9.8×10^{-6}	2.3×10^{-8}
^{234}Th+d	7.6×10^{-8}	2.9×10^{-5}
^{234}U	1.1×10^{-5}	8.8×10^{-8}

Table 5.3-1. Scenario Dose Conversion Factors for all Exposure Pathways for the Agriculture Scenario (cont.)

Radionuclide[a,b]	EDE for Internal Exposure Pathways (rem/yr per μCi/m^3)	EDE for External Exposure Pathways (rem/yr per μCi/m^3)
^{237}Np+d	2.5×10^{-4}	2.4×10^{-4}
^{238}Pu	3.5×10^{-5}	3.3×10^{-8}
^{234}U	1.1×10^{-5}	8.8×10^{-8}
^{239}Pu	3.9×10^{-5}	6.3×10^{-8}
^{235}U	1.0×10^{-5}	1.6×10^{-4}
^{240}Pu	3.9×10^{-5}	3.2×10^{-8}
^{236}U	1.0×10^{-5}	4.6×10^{-8}
^{241}Pu	7.7×10^{-7}	1.3×10^{-9}
^{241}Am	4.6×10^{-5}	9.5×10^{-6}
^{237}Np+d	2.5×10^{-4}	2.4×10^{-4}
^{242}Pu	3.7×10^{-5}	2.8×10^{-8}
^{238}U	9.8×10^{-6}	2.3×10^{-8}
^{244}Pu+d	3.6×10^{-5}	4.6×10^{-4}
^{241}Am	4.6×10^{-5}	9.5×10^{-6}
^{237}Np+d	2.5×10^{-4}	2.4×10^{-4}
^{242m}Am+d	4.5×10^{-5}	1.1×10^{-5}
^{238}Pu	3.5×10^{-5}	3.3×10^{-8}
^{234}U	1.1×10^{-5}	8.8×10^{-8}
^{243}Am+d	4.6×10^{-5}	2.0×10^{-4}
^{239}Pu	3.9×10^{-5}	6.3×10^{-8}
^{242}Cm	1.1×10^{-6}	0.0
^{238}Pu	3.5×10^{-5}	3.3×10^{-8}
^{234}U	1.1×10^{-5}	8.8×10^{-8}
^{243}Cm	2.6×10^{-5}	1.3×10^{-4}
^{244}Cm	2.0×10^{-5}	2.8×10^{-8}
^{240}Pu	3.9×10^{-5}	3.2×10^{-8}
^{236}U	1.0×10^{-5}	4.6×10^{-8}
^{245}Cm	4.0×10^{-5}	7.4×10^{-5}
^{241}Pu	7.7×10^{-7}	1.3×10^{-9}
^{241}Am	4.6×10^{-5}	9.5×10^{-6}
^{237}Np	2.5×10^{-4}	2.4×10^{-4}
^{246}Cm	4.0×10^{-5}	2.6×10^{-8}
^{247}Cm+d	3.7×10^{-5}	4.0×10^{-4}
^{243}Am+d	4.6×10^{-5}	2.0×10^{-4}
^{239}Pu	3.9×10^{-5}	6.3×10^{-8}
^{248}Cm	1.4×10^{-4}	1.9×10^{-8}
^{244}Pu+d	3.6×10^{-5}	4.6×10^{-4}
^{249}Cf	4.1×10^{-5}	4.2×10^{-4}

Table 5.3-1. Scenario Dose Conversion Factors for all Exposure Pathways for the Agriculture Scenario (cont.)

Radionuclide[a,b]	EDE for Internal Exposure Pathways (rem/yr per μCi/m^3)	EDE for External Exposure Pathways (rem/yr per μCi/m^3)
^{245}Cm	4.0×10^{-5}	7.4×10^{-5}
^{241}Pu	7.7×10^{-7}	1.3×10^{-9}
^{241}Am	4.6×10^{-5}	9.5×10^{-6}
^{237}Np	2.5×10^{-4}	2.4×10^{-4}
^{251}Cf	4.1×10^{-5}	1.2×10^{-4}

[a] Indented entries are decay products of parent radionuclide that are potentially radiologically-significant.
[b] The notation "+d" indicates that short-lived daughters were incorporated into the EDE. Dose from other daughters listed is calculated explicitly by adding EDE's.

Table 5.3-2. Scenario Dose Conversion Factors for the External Exposure Pathway of the Resident Scenario

Radionuclide[a,b]	EDE (rem/yr per μCi/m^3)[c]
^{3}H	0.0
^{14}C	2.9×10^{-9}
^{59}Ni	0.0
^{60}Co	3.5×10^{-3}
^{63}Ni	0.0
^{79}Se	4.2×10^{-9}
^{90}Sr+d	5.4×10^{-6}
^{93}Mo	1.3×10^{-7}
^{93}Zr+d	2.3×10^{-8}
^{94}Nb	2.1×10^{-3}
^{99}Tc	2.7×10^{-8}
^{126}Sn+d	2.6×10^{-3}
^{129}I	2.8×10^{-6}
^{135}Cs	8.4×10^{-9}
^{137}Cs+d	7.6×10^{-4}
^{242}Sm	2.1×10^{-10}
^{154}Eu	1.7×10^{-3}
^{232}Th	1.2×10^{-7}
^{228}Ra+d	1.3×10^{-3}
^{228}Th	1.8×10^{-6}
^{224}Ra+d	2.2×10^{-3}
^{232}U	2.0×10^{-7}
^{228}Th	1.8×10^{-6}
^{224}Ra+d	2.2×10^{-3}
^{234}U	8.8×10^{-8}
^{230}Th	2.6×10^{-7}
^{226}Ra+d	2.4×10^{-3}
^{210}Pb	5.3×10^{-7}
^{210}Po	1.2×10^{-8}
^{235}U+d	1.6×10^{-4}
^{231}Pa	4.2×10^{-5}
^{227}Ac	1.1×10^{-7}
^{227}Th	1.1×10^{-4}
^{223}Ra+d	2.6×10^{-4}
^{236}U	4.6×10^{-8}
^{238}U	2.2×10^{-8}
^{234}Th+d	2.9×10^{-5}
^{237}Np+d	2.4×10^{-4}
^{238}Pu	3.3×10^{-8}

Table 5.3-2. Scenario Dose Conversion Factors for the External Exposure Pathway of the Resident Scenario (cont.)

Radionuclide[a,b]	EDE (rem/yr per μCi/m^3) [c]
^{234}U	8.8×10^{-8}
^{239}Pu	6.3×10^{-8}
^{235}U	1.5×10^{-4}
^{240}Pu	3.2×10^{-8}
^{236}U	4.6×10^{-8}
^{241}Pu	1.3×10^{-9}
^{241}Am	9.4×10^{-6}
^{237}Np	2.4×10^{-4}
^{242}Pu	2.8×10^{-8}
^{238}U	2.2×10^{-8}
^{244}Pu+d	4.6×10^{-4}
^{241}Am	9.5×10^{-6}
^{237}Np	2.4×10^{-4}
^{242m}Am+d	1.1×10^{-5}
^{238}Pu	3.3×10^{-8}
^{234}U	8.8×10^{-8}
^{243}Am+d	2.0×10^{-4}
^{239}Pu	6.3×10^{-8}
^{243}Cm	1.3×10^{-4}
^{244}Cm	2.8×10^{-8}
^{240}Pu	3.2×10^{-8}
^{236}U	4.6×10^{-8}
^{245}Cm	7.4×10^{-5}
^{241}Pu	1.3×10^{-9}
^{241}Am	9.4×10^{-6}
^{237}Np	1.7×10^{-5}
^{246}Cm	2.6×10^{-8}
^{247}Cm+d	4.0×10^{-4}
^{243}Am	3.1×10^{-5}
^{239}Np	1.6×10^{-4}
^{239}Pu	6.3×10^{-8}
^{248}Cm	1.9×10^{-8}
^{244}Pu+d	4.2×10^{-4}
^{249}Cf	4.2×10^{-4}
^{245}Cm	7.4×10^{-5}
^{241}Pu	1.3×10^{-9}

Table 5.3-2. Scenario Dose Conversion Factors for the External Exposure Pathway of the Resident Scenario (cont.)

Radionuclide[a,b]	EDE (rem/yr per μCi/m^3) [c]
^{241}Am	9.4×10^{-6}
^{237}Np	2.4×10^{-4}
^{251}Cf	1.2×10^{-4}

[a] Indented entries are decay products of parent radionuclide that are potentially radiologically-significant.
[b] The notation "+d" indicates that short-lived daughters were incorporated into the EDE. Dose from other daughters listed is calculated explicitly by adding EDE's.
[c] Results assume disposal unit has lost engineered barriers that provide shielding above the waste.

The parameter F in Eq. 5.3-1 takes into account that, between the time of disposal and the time that a scenario for inadvertent intrusion is assumed to occur, the initial inventories of radionuclides in the disposal units would be reduced by radioactive decay and by mobilization and transport in water which infiltrates the disposal unit. For most radionuclides in the various disposal units, the value of the parameter F as a function of time after disposal is obtained from calculations performed by the PORFLOW computer code. The fraction of the initial inventory remaining in disposal units at any time after disposal takes into account radioactive decay and removal from the disposal units by infiltrating water. For any radiologically-significant long-lived decay products, ingrowth is calculated explicitly by PORFLOW. Two radionuclides (^{60}Co and ^{242m}Am+d) were not evaluated with the PORFLOW code, because they were not important to the water resource analysis. Only radioactive decay was considered in developing F factors for these radionuclides. Ingrowth of potentially significant radioactive daughters of ^{242m}Am was evaluated in the following manner. For progeny with decay half-lives shorter or comparable to that of ^{242m}Am (i.e., ^{242}Am, ^{242}Cm, and ^{238}Pu), the daughter activities are assumed to be equivalent to that of the parent throughout the analysis. For the ^{235}U daughter of ^{242m}Am, which is longer-lived than ^{242m}Am, the fraction of ^{235}U remaining in the waste was estimated by assuming that ^{242m}Am decays completely to the ^{235}U in the disposal unit at time zero. This assumption neglects the loss of the parent through leaching and assumes instantaneous ingrowth of ^{235}U, thereby maximizing the value of the fraction remaining term for that daughter. The initial activity of the daughter, ^{235}U, per Ci of original inventory of the parent (^{242m}Am), was calculated from:

$$A_{D0} = A_{P0}\frac{\lambda_d}{\lambda_p} = \frac{\lambda_D}{\lambda_P} \qquad (5.3\text{-}2)$$

where

A_{D0} = initial activity of the daughter, Ci,
A_{P0} = initial activity of the parent, (= 1 Ci),
λ_D = radioactive decay constant (0.693/half-life) of the daughter (yr), and
λ_P = radioactive decay constant of the parent (yr).

To calculate the fraction of the daughter remaining at time t, an exponential decay term ($e^{-\lambda_D t}$) was multiplied by the estimated initial activity of the daughter per Ci of the parent, A_{D0}. This produces an estimate of the activity (Ci) of the daughter at time t, per Ci of the parent radionuclide.

Since the objective of the intruder dose analysis is to establish limits on inventories of radionuclides for disposal, Eq. 5.3-1 can be rearranged to give

$$I_0 = (H \times V)/(SDCF \times G \times F). \qquad (5.3\text{-}3)$$

Therefore, for a dose limit for inadvertent intruders of 100 mrem (0.1 rem) per year and using a conversion factor for activity of 10^{-6} Ci/μCi, the inventory limit for a radionuclide in a particular type of disposal unit for a given exposure scenario is given by

$$I_0(Ci) = [10^{-7} \times V(m^3)]/[SDCF(rem\text{-}m^3/\mu Ci\text{-}y) \times G \times F]. \qquad (Eq.\ 5.3\text{-}4)$$

The SDCF depends only on the radionuclide and exposure scenario, G is the same for both exposure scenarios considered here, and F depends on the time after disposal at which intrusion is assumed to occur and on the radionuclide.

Alternatively, the waste acceptance criteria (WAC) could be expressed in terms of limits on average concentrations of radionuclides in disposed waste. From Eq. 5.3-4, the limit on average concentration of a radionuclide is given by

$$C_0(\mu Ci/m^3) = 0.1/[SDCF(rem\text{-}m^3/\mu Ci\text{-}y) \times G \times F]. \quad \text{(Eq. 5.3-5)}$$

The models in Eqs. 5.3-4 and 5.3-5 for estimating limits on inventories or average concentrations of radionuclides in the different types of disposal units for the different exposure scenarios for inadvertent intruders are implemented in the following sections.

5.3.1 Dose Analysis for the Agriculture Scenario

Application of the models in Eqs. 5.3-1 through 5.3-5 to the Naval Reactor Waste Disposal Pad produced the results of the dose analysis for intruder-agriculture scenario, which are discussed in this section. The results provided exclude the contributions from radon.

The agriculture scenario involves direct excavation into disposal units at times after the engineered barriers above the waste have lost their structural and physical integrity and can be penetrated by normal excavation procedures at SRS. At some distant time in the future, it is conceivable that the corrosion-resistant steel disposal casks will have degraded by corrosion and that the waste could be accessed by excavation, resulting in exposures of inadvertent intruders according to the agriculture scenario described in Section 6.1.1.1 of the E-Area LLWF PA (McDowell-Boyer et al. 2000).

On the basis of the design of the conceptual waste cask described in Section 2.3.2 and estimates of corrosion rates of carbon steel, excavation into the waste would not be a credible occurrence for a very long time after disposal. For excavation into the waste to occur, the top of the casks would need to be degraded by corrosion. Assuming the thickness of steel on top of the disposal cask is 38 cm (Section 2.3.2), and using a corrosion rate of 4×10^{-3} m/yr for carbon steel (Sullivan et al. 1988), a conservative estimate of the time for complete corrosion of the steel top is about 9,500 years. This estimate is conservative because the corrosion-resistant steel used in the casks will likely delay corrosion beyond that of carbon steel.

The Naval Reactor components are constructed of highly corrosion-resistant Inconel and Zircaloy metals and hold the vast majority of radionuclide contaminants as activation products in the metal matrices. The Naval Reactor Inconel and Zircaloy waste forms are expected to maintain their structural and physical integrity for 30,000 years or more (Appendix C, Table C.1-7). Thus, unless the corrosion rates for Inconel or Zircaloy have been underestimated by more than an order of magnitude, which is not likely, it seems reasonable to conclude that direct intrusion into the waste by excavation is not a credible occurrence for about the first 40,000 years after disposal which is well beyond the assumed time period for compliance with the performance measure for protection of inadvertent intruders (1000 years). Therefore, on the basis of the agriculture scenario, no limits on average concentrations or inventories of radionuclides in waste casks need to be imposed to provide protection of future inadvertent intruders.

Simply for the purpose of providing a perspective on inventory limits based on the agriculture scenario, a dose analysis for this scenario at 10,000 years, the minimum time the steel casks can be expected to completely corrode, was performed for the radionuclides listed in Section 5.2 for Naval Reactor components. Based on this analysis, limits on concentration and inventory were calculated (Eq. 5.3-4 and 5.3-5) for the Naval Reactor waste casks, and are presented in Table 5.3-3.

Table 5.3-3. Intruder-Based Radionuclides Disposal Limits for the Agriculture Scenario at 10,000 Years

Radionuclide[a,b]	Fraction Remaining Available[c]	Fraction Remaining Total[d]	Concentration Limit[e] ($\mu Ci/m^3$)	Inventory Limit[f] (Ci/pad)
^{3}H	0.0	0.0	$>1 \times 10^{20}$	$>1 \times 10^{20}$
^{14}C	5.9×10^{-5}	2.6×10^{-1}	7.6×10^{8}	3.2×10^{6}
^{60}Co	0.0[g]	0.0[g]	$>1 \times 10^{20}$	$>1 \times 10^{20}$
^{59}Ni	2.2×10^{-1}	8.7×10^{-1}	8.3×10^{7}	3.6×10^{5}
^{63}Ni	0.0	0.0	$>1 \times 10^{20}$	$>1 \times 10^{20}$
^{79}Se	3.1×10^{-2}	6.5×10^{-1}	3.1×10^{7}	1.3×10^{5}
$^{90}Sr+d$	0.0	0.0	$>1 \times 10^{20}$	$>1 \times 10^{20}$
^{93}Mo	3.1×10^{-4}	9.4×10^{-2}	9.8×10^{7}	4.2×10^{5}
$^{93}Zr+d$	3.9×10^{-2}	9.5×10^{-1}	5.4×10^{7}	2.3×10^{5}
^{94}Nb	2.0×10^{-2}	5.5×10^{-1}	1.19×10^{3}	4.6
^{99}Tc	1.1×10^{-4}	7.2×10^{-1}	6.0×10^{7}	2.6×10^{5}
$^{126}Sn+d$	1.0×10^{-2}	6.4×10^{-1}	7.5×10^{2}	3.2
^{129}I	4.0×10^{-5}	1.5×10^{-1}	2.9×10^{6}	1.3×10^{4}
^{135}Cs	7.1×10^{-2}	7.5×10^{-1}	1.3×10^{7}	5.7×10^{4}
$^{137}Cs+d$	0.0	0.0	$>1 \times 10^{20}$	$>1 \times 10^{20}$
^{151}Sm	0.0	0.0	$>1 \times 10^{20}$	$>1 \times 10^{20}$
^{154}Eu	0.0	0.0	$>1 \times 10^{20}$	$>1 \times 10^{20}$
^{232}Th	7.1×10^{-1}	7.1×10^{-1}	3.1×10^{2}	1.3
$^{228}Ra+d$	7.1×10^{-1}	7.1×10^{-1}	[h]	[h]
^{228}Th	7.1×10^{-1}	7.1×10^{-1}	[h]	[h]
$^{224}Ra+d$	7.1×10^{-1}	7.1×10^{-1}	[h]	[h]
^{232}U	0.0	0.0	$>1 \times 10^{20}$	$>1 \times 10^{20}$
^{228}Th	0.0	0.0	[h]	[h]
$^{224}Ra+d$	0.0	0.0	[h]	[h]
^{234}U	3.9×10^{-1}	1.0	9.7×10^{3}	4.2×10^{1}
^{230}Th	3.0×10^{-2}	8.7×10^{-2}	[h]	[h]
$^{226}Ra+d$	4.9×10^{-3}	5.0×10^{-2}	[h]	[h]
^{210}Pb	4.9×10^{-3}	5.1×10^{-2}	[h]	[h]
^{210}Po	4.9×10^{-3}	5.1×10^{-2}	[h]	[h]
$^{235}U+d$	4.0×10^{-1}	1.0	5.0×10^{3}	2.1×10^{1}
^{231}Pa	4.0×10^{-2}	1.7×10^{-1}	[h]	[h]
^{227}Ac	4.0×10^{-2}	1.7×10^{-1}	[h]	[h]
^{227}Th	4.0×10^{-2}	1.7×10^{-1}	[h]	[h]
^{223}Ra	4.0×10^{-2}	1.7×10^{-1}	[h]	[h]
^{236}U	3.8×10^{-1}	1.0	3.1×10^{5}	1.3×10^{3}
^{238}U	4.0×10^{-1}	1.0	3.8×10^{4}	1.6×10^{2}
^{234}Th	4.0×10^{-1}	1.0	[h]	[h]
^{234}U	1.1×10^{-2}	2.9×10^{-2}	[h]	[h]
$^{237}Np+d$	2.0×10^{-1}	8.8×10^{-1}	4.8×10^{3}	2.0×10^{1}

Table 5.3-3. Intruder-Based Radionuclides Disposal Limits for the Agriculture Scenario at 10,000 Years (cont.)

Radionuclide[a,b]	Fraction Remaining Available[c]	Fraction Remaining Total[d]	Concentration Limit[e] ($\mu Ci/m^3$)	Inventory Limit[f] (Ci/pad)
^{238}Pu	0.0	0.0	3.5×10^{9}	1.5×10^{7}
^{234}U	3.0×10^{-5}	3.4×10^{-4}	[h]	[h]
^{239}Pu	5.5×10^{-2}	7.4×10^{-1}	5.7×10^{5}	2.5×10^{3}
^{235}U	6.8×10^{-7}	8.4×10^{-6}	[h]	[h]
^{240}Pu	2.6×10^{-2}	3.4×10^{-1}	1.2×10^{6}	5.2×10^{3}
^{236}U	1.5×10^{-5}	1.8×10^{-4}	[h]	[h]
^{241}Pu	0.0	0.0	7.6×10^{8}	3.3×10^{6}
^{241}Am	4.1×10^{-10}	3.8×10^{-9}	[h]	[h]
$^{237}Np+d$	3.3×10^{-7}	6.5×10^{-6}	[h]	[h]
^{242}Pu	7.4×10^{-2}	9.9×10^{-1}	4.3×10^{5}	1.9×10^{3}
^{238}U	1.3×10^{-7}	1.6×10^{-6}	[h]	[h]
$^{244}Pu+d$	7.5×10^{-2}	1.0	2.7×10^{3}	1.2×10^{1}
^{241}Am	5.6×10^{-8}	1.2×10^{-7}	2.4×10^{7}	1.0×10^{5}
$^{237}Np+d$	4.0×10^{-5}	1.7×10^{-4}	[h]	[h]
$^{242m}Am+d$	0.0[g]	0.0[g]	2.3×10^{8}	9.8×10^{5}
^{238}Pu	0.0[g]	0.0[g]	[h]	[h]
^{234}U	5.0×10^{-4}[g]	5.0×10^{-4}[g]	[h]	[h]
^{243}Am	1.5×10^{-1}	4.2×10^{-1}	1.4×10^{4}	5.9×10^{1}
^{239}Np	1.5×10^{-1}	4.2×10^{-1}	[h]	[h]
^{239}Pu	5.5×10^{-2}	1.4×10^{-1}	[h]	[h]
^{243}Cm	0.0	0.0	$>1 \times 10^{20}$	$>1 \times 10^{20}$
^{244}Cm	0.0	0.0	9.6×10^{7}	4.1×10^{5}
^{240}Pu	3.3×10^{-4}	9.8×10^{-4}	[h]	[h]
^{236}U	1.9×10^{-7}	5.4×10^{-7}	[h]	[h]
^{245}Cm	1.7×10^{-1}	4.6×10^{-1}	2.3×10^{4}	9.9×10^{1}
^{241}Pu	1.7×10^{-1}	4.6×10^{-1}	[h]	[h]
^{241}Am	1.8×10^{-1}	5.0×10^{-1}	[h]	[h]
$^{237}Np+d$	5.0×10^{-4}	1.9×10^{-3}	[h]	[h]
^{246}Cm	9.2×10^{-2}	2.4×10^{-1}	3.4×10^{5}	1.5×10^{3}
^{247}Cm	3.8×10^{-1}	1.0	2.2×10^{3}	9.7
^{243}Am	2.3×10^{-1}	6.5×10^{-1}	[h]	[h]
^{239}Np	2.3×10^{-1}	6.5×10^{-1}	[h]	[h]

Table 5.3-3. Intruder-Based Radionuclides Disposal Limits for the Agriculture Scenario at 10,000 Years (cont.)

Radionuclide[a,b]	Fraction Remaining Available[c]	Fraction Remaining Total[d]	Concentration Limit[e] ($\mu Ci/m^3$)	Inventory Limit[f] (Ci/pad)
^{239}Pu	3.2×10^{-2}	1.0×10^{-1}	[h]	[h]
^{248}Cm	3.8×10^{-1}	1.0	2.3×10^{4}	9.9×10^{1}
^{244}Pu	2.8×10^{-5}	8.5×10^{-5}	[h]	[h]
^{249}Cf	1.4×10^{-9}	3.1×10^{-9}	5.3×10^{5}	2.3×10^{3}
^{245}Cm	7.4×10^{-3}	2.0×10^{-2}	[h]	[h]
^{241}Pu	7.4×10^{-3}	2.1×10^{-2}	[h]	[h]
^{241}Am	7.7×10^{-3}	2.1×10^{-2}	[h]	[h]
$^{237}Np+d$	2.1×10^{-5}	7.7×10^{-5}	[h]	[h]
^{251}Cf	0.0	0.0	$>1 \times 10^{20}$	$>1 \times 10^{20}$

[a] Indented entries are decay products of parent radionuclide that are potentially radiologically-significant.
[b] The notation "+d" indicates that short-lived daughters were incorporated into the EDE. Dose from other daughters listed is calculated explicitly.
[c] "Available" refers to the portion of the waste that has been corroded from the activated waste. Fractions shown for radioactive daughters reflect fraction of initial activity of parent. Considers radioactive decay and leaching unless otherwise noted.
[d] "Total" refers to both corroded and activated metal waste. Fractions shown for radioactive daughters reflect fraction of initial activity of parent. Considers radioactive decay and leaching unless otherwise noted.
[e] Limit on average concentration in disposed waste; obtained from Eq. 5.3-5
[f] Limit on inventory per Naval Reactor Waste Disposal Pad; obtained from Eq. 5.3-4, assuming a volume of 4.3 x 10^3 m^3 per 100-container pad.
[g] Only radioactive decay accounted for.
[h] Daughter contributions are accounted for in parent radionuclide's limit.

5.3.2 Dose Analysis for the Resident Scenario

In the intruder-resident scenario, the intruder is assumed to reside in a home located immediately on top of an intact concrete roof or other engineered barrier above a disposal unit, and the scenario is assumed to be credible immediately following loss of active institutional control at 100 years after disposal. The intruder is assumed not to excavate in the waste itself, as this is not credible at this time for the Naval Reactor Waste Disposal Pad, given the corrosion resistance of the disposal containers. Thus, the only pathway of concern for this scenario is external exposure to photon-emitting radionuclides in the waste while residing in the home.

The Naval Reactor components are encased in high-strength, low-alloy, corrosion-resistant steel waste disposal containers, or casks, which are closed by a weld (Section 2.3.2). These casks are disposed in the Naval Reactor Waste Disposal Pad, which may ultimately contain 100 casks. There is no standard disposal container due to the variety of components stored in the containers; therefore, precisely accounting for shielding afforded by the waste form or cask is not possible.

Before a detailed analysis of potential dose to a resident intruder was made, the list of radionuclides from Section 5.2 was further screened for potential significance by calculating external dose arising from exposure to radionuclides in unshielded waste. While this is not a credible scenario, it is useful in limiting the number of radionuclides for which a more detailed analysis is required. Applying Eq. 5.3-4 above, and using the SDCF's in Table 5.3-2, the inventory limits for each radionuclide were calculated and compared to the projected inventory in Table 2.4-2. The value of F, the fraction of initial radioactivity remaining at the time the scenario is assumed credible, was calculated by assuming decay only depleted the original activity in the waste for 100 years. The results are presented in Table 5.3-4. The results of this screening analysis indicate that even under these assumptions of no shielding, only the projected inventory of ^{94}Nb exceeds the calculated inventory limit (i.e., ratio of limit to projected inventory is less than one), and thus needs further analysis. The calculated inventory limits for all other radionuclides exceed the projected inventory, under the extremely conservative conditions assumed.

For the analysis of the resident scenario for the Naval Reactor waste federal requirements for transportation of hazardous materials, specified in 49 CFR Part 173, were relied upon for estimating exposure to photon emissions from radionuclides projected to be present in the Naval Reactor waste destined for disposal at E Area. These regulations were useful in estimating an upper bound on the external dose rate at the surface of the casks.

According to 49 CFR Part 173, the external dose rate at the surface of a waste cask may not exceed 0.2 rem/hr (200 mrem/hr). Measurements that have been done on selected Naval Reactor waste casks indicate that the external dose rate is less than 15 mrem/hr, and averages less than 2 mrem/hr. The following discussion will show that shielding provided by the waste casks and waste components must be significant to satisfy the federal requirements, based on the projected inventories provided in Table 2.4-2.

Table 5.3-4. Intruder-Based Radionuclides Disposal Limits for the Resident Scenario At 100 Years

(Used for screening only – scenario not credible because no shielding of the waste is assumed)

Radionuclide[a,b]	Fraction Remaining[c,d]	Concentration Limit[e] ($\mu Ci/m^3$)	Inventory Limit[f] (Ci/pad)	Ratio of Inventory Limit to Projected Inventory
^{3}H	3.6×10^{-3}	$>1 \times 10^{20}$	$>1 \times 10^{20}$	$>1 \times 10^{20}$
^{14}C	9.9×10^{-1}	4.4×10^{8}	1.9×10^{6}	5.4×10^{3}
^{60}Co	1.9×10^{6}[g]	1.9×10^{8}	7.9×10^{5}	3.1
^{59}Ni	1.0	$>1 \times 10^{20}$	$>1 \times 10^{20}$	$>1 \times 10^{20}$
^{63}Ni	5.1×10^{-1}	$>1 \times 10^{20}$	$>1 \times 10^{20}$	$>1 \times 10^{20}$
^{79}Se	1.0	3.1×10^{8}	1.3×10^{6}	4.1×10^{8}
$^{90}Sr+d$	8.9×10^{-2}	2.7×10^{6}	1.1×10^{4}	8.6×10^{2}
^{93}Mo	9.8×10^{-1}	1.0×10^{7}	4.2×10^{4}	1.2×10^{4}
$^{93}Zr+d$	1.0	5.6×10^{7}	2.4×10^{5}	1.2×10^{1}
^{94}Nb	1.0	6.1×10^{2}	2.6	1.6×10^{-1}
^{99}Tc	1.0	4.7×10^{7}	2.0×10^{5}	5.4×10^{5}
$^{126}Sn+d$	1.0	4.9×10^{2}	2.1	1.1×10^{5}
^{129}I	1.0	$>1 \times 10^{20}$	$>1 \times 10^{20}$	$>1 \times 10^{20}$
^{135}Cs	1.0	1.5×10^{8}	6.4×10^{5}	7.4×10^{9}
$^{137}Cs+d$	1.0×10^{-1}	1.7×10^{4}	7.0×10^{1}	5.3
^{151}Sm	4.6×10^{-1}	1.3×10^{10}	5.4×10^{7}	3.9×10^{8}
^{154}Eu	3.8×10^{-4}	2.0×10^{6}	8.4×10^{3}	5.0×10^{4}
^{232}Th	1.0	1.3	5.4×10^{3}	3.2×10^{7}
$^{228}Ra+d$	1.0	[h]	[h]	[h]
^{228}Th	1.0	[h]	[h]	[h]
$^{224}Ra+d$	1.0	[h]	[h]	[h]
^{232}U	3.8×10^{-1}	1.5×10^{3}	6.2	2.2×10^{6}
^{228}Th	4.0×10^{-1}	[h]	[h]	[h]
$^{224}Ra+d$	4.0×10^{-1}	[h]	[h]	[h]
^{234}U	1.0	9.5×10^{6}	4.0×10^{4}	5.7×10^{9}
^{230}Th	9.0×10^{-4}	[h]	[h]	[h]
$^{226}Ra+d$	1.9×10^{-5}	[h]	[h]	[h]
^{210}Pb	1.1×10^{-5}	[h]	[h]	[h]
^{210}Po	1.1×10^{-5}	[h]	[h]	[h]
$^{235}U+d$	1.0	8.0×10^{3}	3.4×10^{1}	6.5×10^{7}
^{231}Pa	2.1×10^{-3}	[h]	[h]	[h]
^{227}Ac	1.5×10^{-3}	[h]	[h]	[h]
^{227}Th	1.5×10^{-3}	[h]	[h]	[h]
^{223}Ra	1.5×10^{-3}	[h]	[h]	[h]
^{236}U	1.0	2.8×10^{7}	1.2×10^{5}	1.1×10^{10}
^{238}U	1.0	4.4×10^{4}	1.9×10^{2}	3.2×10^{6}
^{234}Th	1.0	[h]	[h]	[h]
^{234}U	2.8×10^{-4}	[h]	[h]	[h]
$^{237}Np+d$	1.0	5.3×10^{3}	2.2×10^{1}	2.2×10^{6}

Table 5.3-4. Intruder-Based Radionuclides Disposal Limits for the Resident Scenario at 100 Years (cont.)

(Used for screening only – scenario not credible because no shielding of the waste is assumed)

Radionuclide[a,b]	Fraction Remaining[c,d]	Concentration Limit[e] ($\mu Ci/m^3$)	Inventory Limit[f] (Ci/pad)	Ratio of Inventory Limit to Projected Inventory
^{238}Pu	4.5×10^{-1}	8.5×10^{7}	3.6×10^{5}	5.3×10^{5}
^{234}U	2.0×10^{-4}	[h]	[h]	[h]
^{239}Pu	1.0	2.0×10^{7}	8.6×10^{4}	2.8×10^{5}
^{235}U	9.8×10^{-8}	[h]	[h]	[h]
^{240}Pu	9.9×10^{-1}	4.0×10^{7}	1.7×10^{5}	6.0×10^{5}
^{236}U	3.0×10^{-6}	[h]	[h]	[h]
^{241}Pu	8.1×10^{-3}	4.7×10^{6}	2.0×10^{4}	2.3×10^{2}
^{241}Am	2.9×10^{-2}	[h]	[h]	[h]
$^{237}Np+d$	8.0×10^{-7}	[h]	[h]	[h]
^{242}Pu	1.0	4.6×10^{7}	1.9×10^{5}	1.9×10^{8}
^{238}U	1.6×10^{-8}	[h]	[h]	[h]
$^{244}Pu+d$	1.0	2.8×10^{3}	1.2×10^{1}	1.7×10^{11}
^{241}Am	8.5×10^{-1}	1.6×10^{5}	6.7×10^{2}	7.6×10^{2}
$^{237}Np+d$	3.0×10^{-5}	[h]	[h]	[h]
$^{242m}Am+d$	6.3×10^{-1g}	1.8×10^{5}	7.6×10^{2}	9.6×10^{7}
^{238}Pu	3.5×10^{-1g}	[h]	[h]	[h]
^{234}U	6.1×10^{-5g}	[h]	[h]	[h]
^{243}Am	9.9×10^{-1}	3.6×10^{3}	1.5×10^{1}	2.5×10^{3}
^{239}Np	9.9×10^{-1}	[h]	[h]	[h]
^{239}Pu	2.9×10^{-3}	[h]	[h]	[h]
^{243}Cm	8.8×10^{-2}	1.2×10^{5}	4.9×10^{2}	1.1×10^{8}
^{244}Cm	2.2×10^{-2}	1.8×10^{9}	7.5×10^{6}	1.6×10^{7}
^{240}Pu	2.7×10^{-3}	[h]	[h]	[h]
^{236}U	6.1×10^{-9}	[h]	[h]	[h]
^{245}Cm	9.9×10^{-1}	1.7×10^{4}	7.3×10^{1}	2.8×10^{6}
^{241}Pu	9.9×10^{-1}	[h]	[h]	[h]
^{241}Am	1.2×10^{-1}	[h]	[h]	[h]
$^{237}Np+d$	1.7×10^{-6}	[h]	[h]	[h]
^{246}Cm	9.9×10^{-1}	5.1×10^{7}	2.1×10^{5}	2.2×10^{10}
^{247}Cm	1.0	3.2×10^{3}	1.3×10^{1}	6.6×10^{11}
^{243}Am	9.4×10^{-3}	[h]	[h]	[h]
^{239}Np	9.4×10^{-3}	[h]	[h]	[h]
^{239}Pu	1.3×10^{-5}	[h]	[h]	[h]
^{248}Cm	1.0	6.5×10^{7}	2.8×10^{5}	5.8×10^{15}
^{244}Pu	8.4×10^{-7}	[h]	[h]	[h]
^{249}Cf	8.2×10^{-1}	3.7×10^{3}	1.6×10^{1}	5.0×10^{10}
^{245}Cm	7.4×10^{-3}	[h]	[h]	[h]

Table 5.3-4. Intruder-Based Radionuclides Disposal Limits for the Resident Scenario at 100 Years (cont.)

(Used for screening only – scenario not credible because no shielding of the waste is assumed)

Radionuclide[a,b]	Fraction Remaining[c,d]	Concentration Limit[e] ($\mu Ci/m^3$)	Inventory Limit[f] (Ci/pad)	Ratio of Inventory Limit to Projected Inventory
^{241}Pu	6.0×10^{-3}	[h]	[h]	[h]
^{241}Am	3.9×10^{-4}	[h]	[h]	[h]
$^{237}Np+d$	3.8×10^{-9}	[h]	[h]	[h]
^{251}Cf	9.3×10^{-1}	1.2×10^{4}	5.0×10^{1}	7.6×10^{12}

[a] Indented entries are decay products of parent radionuclide that are potentially radiologically-significant.
[b] The notation "+d" indicates that short-lived daughters were incorporated into the EDE. Dose from other daughters listed is calculated explicitly.
[c] Includes both corrosion products and activated metal waste. Considers radioactive decay and leaching unless otherwise noted.
[d] Fractions shown for radioactive daughters reflect fraction of initial activity of parent.
[e] Limit on average concentration in disposed waste; obtained from Eq. 5.3-5
[f] Limit on inventory per Naval Reactor Waste Disposal Pad; obtained from Eq. 5.3-4, assuming a volume of 4.3 x 10^3 m^3 per 100-container pad.
[g] Only radioactive decay accounted for.
[h] Daughter contributions are accounted for in parent radionuclide's limit.

Based on the projected radionuclide inventory in Table 2.4-2 and the known spectrum of photons emitted in the decay of the various radionuclides, the most important photon-emitting radionuclide in the waste appears to be ^{60}Co. That is, for an average waste cask at the present time, a substantial fraction of the external dose rate at the surface of an intact waste package should be due to the inventory of ^{60}Co. From Table 2.4-2, the projected inventory of ^{60}Co in 100 waste casks is reported as 2.5×10^5 Ci, which is an average of 2.5×10^3 Ci per cask. In reality, some casks may contain more than 2.5×10^3 Ci, and some may contain less. But all casks must conform to the external dose rate requirements of less than or equal to 0.2 rem/hr at the surface. Assuming, for simplicity, that the average activity is in the form of a point source at a distance of 1 m from the surface of the waste cask, the external dose rate at the surface of the cask can be estimated from the known spectrum of photons emitted in the decay of ^{60}Co and the amount of shielding provided by the walls of the cask. At a distance of 1 m from a point source, the unshielded external dose rate in air per unit activity of ^{60}Co has been calculated from the decay data of Kocher (1981) as 1.4 rem/hr per Ci (Unger and Trubey 1982). The minimum container closure weld thickness of the casks is 4 cm (Section 2.3.2), although the casks themselves are much thicker (35 to 38 cm). From Figure 13 of NCRP (1976), a thickness of 4 cm of iron, which is a reasonable approximation for the shielding provided by carbon steel, reduces the external dose rate from ^{60}Co by a factor of 0.4. Thus, the estimated dose rate for the simplified case considered here is $(2.5 \times 10^3$ Ci) × (1.4 rem/hr per Ci) × (0.4) = 1400 rem/hr. This estimate far exceeds the 0.2 rem/hr acceptance criterion for the waste casks. If one assumes the effective shielding is closer to the thickness of the cask walls or top, the estimated dose rate would be $(2.5 \times 10^3$ Ci) × (1.4 rem/hr per Ci) × (6×10^{-4}) = 2.1 rem/hr, which is still much higher than the average measured rate of approximate .002 rem/hr, but closer to the required maximum surface rate of 0.2 rem/hr.

There are two possible explanations for the high estimate of external dose given above. The first is that the stated inventory of ^{60}Co in the waste is a highly conservative inventory. If such a waste inventory were truly to be shipped, the effective shielding provided by the disposal container, or the shipping/storage/disposal container package, must be made much greater than the 4 cm minimum thickness at the weld. Indeed, to achieve 200 mrem/hr at the container surface, a reduction factor of 1.4×10^{-4} is needed, if credit is not taken for self-attenuation by the activation product matrix, or for a thicker container. The second explanation for the high estimated dose rate based on the projected inventory in Appendix A is that the Naval Reactor waste is mostly in the form of solid activated metals, and the shielding provided by the waste form matrix was not taken into account in the simple calculation given above. Per 10 CFR Part 61, NRC recommends using a reduction in dose by a factor of 10 for activated metals due to their inherent shielding properties. Taking credit for the activation waste form using this factor still does not achieve a credible dose rate, suggesting that self-attenuation and effective thickness of the containers are likely much greater than assumed for this calculation. Recent detailed shielding calculations were conducted by Rosenberger (1997) for the Naval Reactor waste casks. Assuming waste radionuclide inventories are uniformly distributed inside a 2.5 m x 4.6 m (D x H) waste cask of 4 cm thickness, the results showed that the dose rate 30 cm from the top of the cask would be about 170 rem/hr, or about 1/8th of that calculated by the simplified model above, but still much higher than the measured rates.

A simplified approach is used to obtain a conservative, upper-bound estimate of the external dose that could be received by an inadvertent intruder at 100 years after disposal, the presumed end of institutional control, which is the earliest time that exposures could occur. This approach is described below and will show that a container thickness sufficient to keep initial dose below 200 mrem/hr at the container surface will be adequate for residential intruder protection at the end of institutional control.

Conservative, upper-bound estimates of external dose ^{94}Nb at 100 years after disposal are obtained as follows. First, a conservative assumption is made that the external dose rate at the surface of a waste cask at the present time is 200 mrem/hr, which is the limit for any cask specified in 49 CFR Part 173. Next, it is assumed that the dose rate of 200 mrem/hr is entirely due to ^{60}Co at the time of disposal, due to its relatively high projected inventory (Table 2.4.2) and high energies and intensities of the emitted photons (Kocher 1981). This assumption also is conservative (i.e., the dose rate from ^{60}Co is overestimated), because a fraction of the external dose at the present time is due to other photon-emitting radionuclides in the waste.

If the dose rate from ^{60}Co at the present time is assumed to be 200 mrem/hr, the dose rate at 100 years after disposal would be reduced by a factor of 1.95×10^{-6}, based on the known half-life of 5.27 years. Thus, a conservative, upper-bound estimate of the external dose rate from ^{60}Co at 100 years after disposal would be (200 mrem/hr) × (1.95×10^{-6}) = 3.9×10^{-4} mrem/hr.

In order to estimate the contributions to external dose from ^{94}Nb at 100 years after disposal, it is assumed that the activities of ^{60}Co and ^{94}Nb in disposed waste are in the same proportions as the total activities of these radionuclides in all casks given in Table 2.4.2. That is, the ratio of the activities of ^{60}Co and ^{94}Nb cask are assumed to be (2.5×10^{5}):(16) at the time of disposal. Then, taking into account the decay of ^{60}Co after 100 years (^{94}Nb experiences negligible decay over this time), the activity ratio at 100 years would be (0.48):(16). It is then assumed that each radionuclide is a point source, in which case the dose rate per unit activity of each radionuclide calculated by Unger and Trubey (1982) can be used. The values in units of mrem/hr per Ci are 1,400 for ^{60}Co and 980 for ^{94}Nb. It is further assumed that the only shielding between the source and receptor locations is the 4 cm minimum carbon steel container thickness and the transmission curves for iron in Figure 13 of NCRP (1976) are used to estimate the reduction in dose rate due to shielding. The latter assumption provides conservative overestimates of dose rate (i.e., underestimates of shielding) for ^{94}Nb, because the actual amount of shielding will be greater than 4 cm and the transmission of photons from ^{94}Nb relative to the transmission of photons from ^{60}Co decreases as the amount of shielding increases. The transmission (shielding) factor for ^{60}Co for 4 cm of iron is obtained directly from Figure 13 of NCRP (1976) as 0.4, respectively, and the value for ^{94}Nb is estimated by interpolation, based on the known photon spectrum for this radionuclide, as 0.32. With the information on dose rate per unit activity and transmission through 4 cm of iron given above, a conservative, upper-bound estimate of the external dose rate from ^{94}Nb can be obtained from the previous upper-bound estimate of the dose rate from ^{60}Co of 3.9×10^{-4} mrem/hr as (3.9×10^{-4}) × (16/0.48) × (980/1,400) × (0.32/0.4) = 7.3 x 10^{-3} mrem/hr.

Thus, a conservative, upper-bound estimate of external dose rate next to a waste cask at 100 years after disposal is 7.7×10^{-3} mrem/hr (3.9×10^{-4} plus 7.3 x 10^{-3} mrem/hr), which is due mostly to ^{94}Nb. This estimate is conservative because (1) the dose rate from ^{60}Co alone is likely overestimated, based on the acceptance criterion on external dose rate for the waste casks of 200 mrem/hr and measured dose rates for casks at the present time of less than 15 mrem/hr, and (2) the transmission of photons from ^{94}Nb through all shielding materials relative to the transmission for ^{60}Co has been overestimated. The analysis would provide an underestimate of the dose rate only if the activity of ^{94}Nb in the waste relative to the activity of ^{60}Co has been greatly underestimated.

The conservative, upper-bound estimate of external dose rate obtained above corresponds to a dose from continuous exposure of 67 mrem per year. If an inadvertent intruder were located next to an "average" waste cask continuously throughout the year, with no additional shielding assumed, the conservative, upper-bound estimate of the external dose rate indicates that the dose received from external exposure would be considerably less than the dose limit of 100 mrem per year in the assessment requirement for protection of inadvertent intruders (Section 1.3). Thus, although the

foregoing analysis is not intended to provide an accurate estimate of external dose at future times, the analysis clearly shows that the external dose to an inadvertent intruder due to exposure to intact waste casks at the end of institutional control should not be of concern in regard to meeting the performance measures for disposal of the Naval Reactor wastes if the dose rate at the surface of the casks is less than 50 mrem/hr. External dose to an inadvertent intruder could be of concern only if the container thickness fails to provide adequate shielding at the start of institutional control, or if the inventory of ^{94}Nb in the waste casks relative to the inventory of ^{60}Co were much larger than reported in Table 2.4-2.

6.0 PERFORMANCE EVALUATION

The purpose of this reanalysis of the performance of the Naval Reactor Waste Disposal Pad is to evaluate the at-grade facility in accordance with requirements set forth in USDOE Order 435.1 for maintaining PAs for LLWFs located at USDOE field sites. The latest revision of the E-Area LLWF PA included an evaluation of the Naval Reactor Waste Disposal Pad placed at a depth in soil such that the bottom of the pad was approximately 7.6 m above the water table. The present analysis considers that an at-grade placement will locate the bottom of the pad at least 18.3 m above the water table. In this section of the reanalysis, comparison of the PA results (i.e., inventory limits) to the performance objectives and assessment requirements is provided, and the utility of the results in developing operational limits is discussed.

6.1 Comparison of Results to Performance Objectives and Assessment Requirements

The performance objectives and assessment requirements of USDOE Order 435.1 for low-level waste disposal are listed in Section 1.3. In essence, these objectives and requirements result in performance measures for members of the public and inadvertent intruders that are not to be exceeded through consideration of credible pathways.

For the assessment of impacts to water resources, it has been determined that compliance with current, not proposed, USEPA standards is required because of the interpretation of CERCLA regulations by the State of South Carolina. However, if and when USEPA changes those standards, the inventory limits presented in this report must be recalculated.

In order to evaluate the performance of the E-Area LLWF with respect to protection of the public from releases to water, soil, plants and animals and with respect to protection of water resources, groundwater concentrations were estimated, and compared with the more restrictive of either allowable groundwater concentrations based on a 25 mrem/yr dose (the performance objective for protection of off-site members of the public from radionuclides released to any media but the atmosphere) or MCLs (the performance measure for protection of water resources). Based on this comparison, inventory limits for the Naval Reactor Waste Disposal Pad were developed (Section 4.1). Inventory limits resulting from atmospheric releases of radionuclides were also evaluated (Section 4.2), based on a comparison of estimated doses with the EDE limit of 10 mrem/yr (Section 1.3). For inadvertent intruders, the only credible scenario in the 10,000 years after disposal is the resident scenario, in which an individual builds a home on top of the Waste Disposal Pad, but does not otherwise intrude into the facility (Section 5.2.2). Dose from the agriculture scenario is not deemed credible until long after 10,000 years (Section 5.2.1). The discussion in Section 5.2.2 demonstrates that only ^{94}Nb is of potential concern in the resident scenario, but that even under very conservative conditions, the dose from this scenario is not likely to exceed the 100-mrem/yr limit. Thus, inventory limits were not explicitly calculated based on intruder scenarios. It should be noted, however, that the protection of the intruder may limit the allowable inventory of ^{94}Nb in this facility if the external dose rate at the surface of the casks exceeds 200 mrem/hr at any time in the period of assessment.

In this section, these calculated inventory limits are compared to the projected inventories for the Naval Reactor Waste Disposal Pad (Table 2.4-2), in order to evaluate whether the units will meet the performance measures from the standpoint of protection of the public, and assessing impacts on groundwater and inadvertent intruders. The result of this comparison is presented in Table 6.1-1. The inventory limits in the second column of this table refer to the lowest limit calculated

Table 6.1-1. Inventory Limits for Naval Reactor Waste Disposal Pad; Limiting Pathway; Comparison to Projected Inventory

Radionuclide [a]	Inventory Limit [b] Ci	Limiting Pathway [c]	Projected Inventory Ci/pad	Ratio of Inventory Limit to Projected Inventory [d]
^{3}H	2.9×10^{5}	air	3.4×10^{2}	8.5×10^{2}
^{14}C	7.7×10^{2}	gw	3.4×10^{2}	2.3
^{59}Ni	1.2×10^{9}	gw	3.9×10^{3}	3.1×10^{5}
^{79}Se	6.8×10^{2}	gw	3.1×10^{-3}	2.2×10^{5}
$^{90}Sr+d$	2.2×10^{14}	gw	1.3×10^{1}	1.7×10^{13}
^{93}Mo	5.2×10^{3}	gw	3.6	1.4×10^{3}
$^{93}Zr+d$	8.8×10^{12}	gw	1.9×10^{4}	4.6×10^{8}
^{94}Nb	8.7×10^{3}	gw[e]	1.6×10^{1}	5.4×10^{2}
^{99}Tc	8.8×10^{1}	gw	3.6×10^{-1}	2.4×10^{2}
$^{126}Sn+d$	3.2×10^{2}	gw	1.9×10^{-3}	1.7×10^{5}
^{129}I	9.0×10^{-4}	gw	9.4×10^{-6}	9.6×10^{1}
^{135}Cs	2.8×10^{7}	gw	8.7×10^{-5}	3.2×10^{11}
$^{232}Th+d$	$>1 \times 10^{20}$	gw	1.7×10^{-10}	$>1 \times 10^{20}$
$^{232}U+d$	2.5×10^{12}	gw	2.8×10^{-6}	8.9×10^{17}
$^{234}U+d$	1.7×10^{3}	gw	7.0×10^{-4}	2.4×10^{6}
$^{235}U+d$	8.1×10^{2}	gw	5.2×10^{-7}	1.6×10^{9}
^{236}U	1.8×10^{3}	gw	1.1×10^{-3}	1.6×10^{6}
$^{238}U+d$	1.2×10^{2}	gw	5.8×10^{-5}	2.1×10^{6}
$^{237}Np+d$	1.3×10^{1}	gw	1.0×10^{-5}	1.3×10^{6}
$^{238}Pu+d$	2.4×10^{7}	gw	6.7×10^{-1}	3.6×10^{7}
$^{239}Pu+d$	5.9×10^{2}	gw	3.1×10^{-1}	1.9×10^{3}
$^{240}Pu+d$	1.3×10^{3}	gw	2.8×10^{-1}	4.6×10^{3}
$^{241}Pu+d$	9.3×10^{6}	gw	8.5×10^{1}	1.1×10^{5}
$^{242}Pu+d$	4.6×10^{2}	gw	1.0×10^{-3}	4.6×10^{5}
$^{244}Pu+d$	4.5×10^{2}	gw	6.9×10^{-11}	6.5×10^{12}
$^{241}Am+d$	6.6×10^{4}	gw	8.8×10^{-1}	7.5×10^{4}
$^{243}Am+d$	7.5×10^{2}	gw	6.0×10^{-3}	1.3×10^{5}
$^{244}Cm+d$	7.1×10^{4}	gw	4.8×10^{-1}	1.5×10^{5}
$^{245}Cm+d$	5.7×10^{3}	gw	2.6×10^{-5}	2.2×10^{8}
^{246}Cm	$>1 \times 10^{20}$	gw	9.8×10^{-6}	$>1 \times 10^{20}$
$^{247}Cm+d$	2.7×10^{3}	gw	2.0×10^{-11}	1.4×10^{14}
$^{248}Cm+d$	1.7×10^{6}	gw	4.7×10^{-11}	3.6×10^{16}
$^{249}Cf+d$	1.4×10^{5}	gw	3.1×10^{-10}	4.5×10^{14}

[a] "+d" indicates potentially-significant short- and long-lived daughters are accounted for in the limit.
[b] Inventory limit based on consideration of peak groundwater concentration outside 100-m buffer zone around disposal units, air pathway (Section 4) and inadvertent intruder doses (Section 5).
[c] Refers to intruder scenario, air pathway, or groundwater ingestion pathway ("gw"); whichever is more restrictive with respect to developing inventory limits.
[d] A ratio that is one or greater indicates the projected inventory is less than the estimated inventory limit and the performance objectives and assessment requirements are met.
[e] The resident scenario may limit the inventory if the ^{94}Nb: ^{60}Co projected inventory is underestimated, or if the surface dose rate of the casks significantly exceeds 200 mrem/hr.

after consideration of all intruder scenarios, groundwater pathways, and air pathways for up to 10,000 years after closure of the facility. The limiting pathway is identified in the third column, which refers to the intruder scenario or groundwater or air pathway that provides this lowest limit for each radionuclide. The fourth column lists the projected inventory of each radionuclide, and is the sum of

the "crud" and "activation" inventories in Table 2.4-2. Finally, the fifth column in these tables represents the value by which the performance of the disposal unit is evaluated in this PA; the ratio of the inventory limit (i.e., how much activity can be disposed without violating requirements of the performance objectives and assessment requirements) to the projected inventory. When the values in the fifth column are greater than one, disposal of the projected inventory is not expected to violate the requirements of the performance objectives or assessment requirements.

For all radionuclides, the calculated inventory limit is greater than the projected inventory for the Naval Reactor Waste Disposal Pad for all exposure scenarios considered credible in the 10,000 years after disposal. Thus, it is believed that this facility can dispose of the forecasted waste receipts. For most radionuclides, the ratio of inventory limit to projected inventory is much greater than 100.

6.2 Critical Radionuclides and Assumptions

From Table 6.1-1, it is apparent that the critical radionuclides in this assessment of performance of the Naval Reactor Pad are ^{14}C, ^{129}I, ^{99}Tc and ^{94}Nb. The inventory limits for these critical radionuclides are based on estimated groundwater concentrations. All other radionuclides have inventory limits that are more than two orders of magnitude greater that the projected inventory of the radionuclide. These radionuclides have very long to moderately long half-lives; 5730 years for ^{14}C, 15,700,000 years for ^{129}I, 213,000 years for ^{99}Tc and 20,300 years for ^{94}Nb.

A comprehensive list of important parameters and assumptions used in developing inventory limits provided in Sect. 6.1 is given in Table 6.2-1. The parameters identified are grouped according to the endpoint of the calculation using the parameter; whether it is for all-pathways or water resource protection analysis, or for the intruder exposure assessment. Sensitivity analyses of many of these parameters with respect to calculated inventory limits have been conducted in the past, and are reported in Section 5.4 of the E-Area LLWF PA (McDowell-Boyer et al. 2000).

A review of the list of parameters in light of the critical radionuclides in this analysis suggest the following critical assumptions. First, critical assumptions and parameters affecting the all-pathways and water resource protection analysis are likely to be hydraulic conductivities and moisture characteristic curves in the waste and vadose zone, Kds, and the corrosion rate of activated metals. Infiltration rates and failure history for the moisture barrier and waste containers are less likely to be important because the critical radionuclides are sufficiently long-lived. Changes in infiltration will occur long before these radionuclides are depleted by radioactive decay. Thus, the peak groundwater concentration will remain relatively unaffected.

Critical assumptions and parameters affecting the intruder analysis are the corrosion rate of disposal canisters and activated metal waste, Kds, waste container design with respect to wall thickness, and the maximum exposure rate at the surface of the disposal container. Assumed corrosion rates and waste container design are critical to the conclusion drawn that the agriculture scenario is not credible before 10,000 years. Changes in values of these critical parameters that render this assumption invalid would necessitate a reevaluation of the intruder dose, especially with respect to the ^{94}Nb exposure in the agriculture scenario. The maximum external dose rate at the surface of the disposal container is critical to establishing whether the inventory limit of ^{94}Nb from the resident scenario

Table 6.2-1. Important Parameters and Assumptions Used in Developing Inventory Limits

Parameter Description	Parameter Value Assumed	Endpoint of Calculation Using Parameter	Comments
Factors affecting water flow through waste:			
Infiltration rate at surface	40 cm/yr	All-pathways and water resource protection analysis	On the upper end of a range of annual average estimates (22 to 45 cm/yr); extremes of 10 cm/yr and 81 cm/yr have been measured
Failure history for moisture barrier	Assumed to fail at 750 years, corresponding to the time it is estimated that the 75% of the weld sealing the disposal containers to corrode.	All-pathways and water resource protection analysis	After institutional control period, moisture barrier is conservatively assumed to be non-functional as soon as containers fail; large uncertainty
Failure history for waste form or engineered barrier	Disposal container failure (i.e., becomes pervious to water) occurs at 750 years based on corrosion rate calculations	All-pathways and water resource protection analysis	Moderate uncertainty due to unknown rate of corrosion of weld material – chose that for carbon steel
Hydraulic conductivities and moisture characteristic curves for all materials in waste zone and vadose zone	Based on both field- and laboratory-derived values, with emphasis on more recently-derived values that reflect better data acquisition technologies	All-pathways and water resource protection analysis	Chose average values for all media based largely on laboratory-derived values

Table 6.2-1. Important Parameters and Assumptions Used in Developing Inventory Limits (cont.)

Parameter Description	Parameter Value Assumed	Endpoint of Calculation Using Parameter	Comments
Factors affecting release of radionuclides from waste:			
Kds	Site-specific when available; most from Sheppard and Thibault when site-specific values not available	All-pathways and water resource protection analysis	Chose average values for sand, which are generally lower than for other media, and thus likely overestimate desorption from waste; large uncertainty
Apparent diffusivities	Value of 5×10^{-6} cm^2/s assumed	All-pathways and water resource protection analysis	Ranges from 10^{-8} to 10^{-5} cm^2/s for various materials; likely contributes minor uncertainty when only interested in peak concentration
Corrosion rate of activated metals	2×10^{-6} to 2×10^{-5} inch/yr for Zircaloy and Inconel, respectively	All-pathways and water resource protection analysis	Conservative estimates, based on specified waste composition; especially for Zircaloy which is reported to be orders of magnitude below that of Inconel; large uncertainty

Table 6.2-1. Important Parameters and Assumptions Used in Developing Inventory Limits (cont.)

Parameter Description	Parameter Value Assumed	Endpoint of Calculation Using Parameter	Comments
Factors affecting transport and dilution of plume in the saturated zone:			
Simulation grid block size for saturated zone	200' by 200' grid size	All-pathways and water resource protection analysis	Chosen to minimize artificial dilution (a close match to the size of the base of the disposal units)
Hydraulic conductivities of aquifers and confining units	Based on both field- and laboratory-derived values, with emphasis on more recently-derived values that reflect better data acquisition technologies	All-pathways and water resource protection analysis	Chose average values for all media based on both field- and laboratory-derived values; individual measurements can vary over several orders of magnitude
Hydrodynamic dispersion	0	All-pathways and water resource protection analysis	Chosen to minimize dispersive effects, and thus conservatively concentrate plume; large uncertainty
Factors affecting intruder dose calculations:			
Air intake (breathing rate)	8,000 m^3/year	Agriculture Scenario	Standard value

Table 6.2-1. Important Parameters and Assumptions Used in Developing Inventory Limits (cont.)

Parameter Description	Parameter Value Assumed	Endpoint of Calculation Using Parameter	Comments
Factors affecting intruder dose calculations:			
Air intake (breathing rate)	8,000 m^3/year	Agriculture Scenario	Standard value
Atmospheric mass loading of contaminated surface soil - working in garden residing in home	10^{-7} kg/m^3 10^{-8} kg/m^3	Agriculture Scenario	Value based on site-specific and nonurban U.S. data
Consumption of contaminated soil	0.037 kg/year	Agriculture Scenario	EPA value - may vary by at least an order of magnitude
Consumption of contaminated vegetables	90 kg (fresh weight) per year	Agriculture Scenario	Site-specific and assumes ½ veg from garden
Corrosion rate of disposal canisters and activated metal waste	4×10^{-3} cm/yr for disposal container; 2×10^{-6} to 2×10^{-5} in/yr for Zircaloy and Inconel, respectively	Agriculture Scenario	Conservatively used corrosion rate of carbon steel for corrosion-resistant waste containers; Inconel and Zircaloy corrosion rates conservative
Density of soil when waste is mixed in	1,400 kg/m^3	Agriculture and Resident Scenarios	Reasonable, and in agreement with site-specific values
Dilution factor for mixing of exhumed waste with native soil in vegetable garden	0.2	Agriculture Scenario	May vary by about 1 order of magnitude; no data to support, but consistent with other intruder analyses

Table 6.2-1. Important Parameters and Assumptions Used in Developing Inventory Limits (cont.)

Parameter Description	Parameter Value Assumed	Endpoint of Calculation Using Parameter	Comments
Exposure times - working in garden residing in home	1% per year 50% per year	Agriculture and Resident Scenarios	Reasonable, but could vary for individuals
Fraction remaining in waste	Varies according to groundwater analysis	Agriculture and Resident Scenarios	May not be conservative for intruders due to Kd or waste corrosion rate selection
Projected inventory of radionuclides; particularly ^{60}Co and ^{94}Nb	Specified by KAPL	Agriculture and Resident Scenarios	Uncertainty unknown, but could be significant
Plant-to-soil concentration factors	Element-specific	Agriculture Scenario	May be uncertain by as much as one-to-three orders of magnitude
Shielding factor for external exposure during indoor residence	0.7	Agriculture and Resident Scenarios	May be uncertain by about a factor of two
Thickness of top and sides of waste disposal container	38 cm (top), 35 cm (sides)	Agriculture Scenario	Based on specified design; critical to whether agriculture scenario is credible within 10,000 years
Maximum external dose rate at surface of disposal container	<200 mrem/hr	Resident Scenario	May be up to 13 times too high, but high value leads to overestimate of ^{94}Nb inventory limit

approaches the projected inventory. If the external dose rate at the surface of any one cask significantly exceeds 200 mrem/hr within the assessment period, there is a possibility the intruder dose limits will be exceeded. The value of 200 mrem/hr is believed to be conservative on the basis of information provided. Furthermore, the ratio of ^{94}Nb to ^{60}Co in the projected inventory is critical to the conclusion that the doses from the resident scenario are likely to be much less than 100 mrem/yr.

6.3 Use of Performance Assessment Results

The inventory limits calculated in this analysis are implemented through a set of WAC and managed through the SRS computerized Waste Information Tracking System (WITS). The operating limits for the E-Area LLWF, as documented in the SRS WAC Manual (WSRC 2001), are derived from safety documentation and a PA. The WAC Manual is a compilation of the radionuclide limits from a Safety Analysis Report (SAR), criticality limits, 100 nCi/g transuranic concentration limit, and performance-based inventory limits, as calculated here. Each of these limits is converted into a hypothetical container limit. For each radionuclide, the most restrictive limit is then implemented as a WAC container limit for the waste generators.

As packages are received for emplacement in the various disposal facilities of the E-Area LLWF, their package contents will be entered in WITS. Before emplacement of each package, WITS will compare the package contents with the 100 nCi/g transuranic limits, and calculate the cell inventory (to ensure compliance with the cell criticality limits), and the total inventory (to ensure compliance with the PA-based limits). The SAR and PA-based limits are tracked as a sum-of-fractions of the individual radionuclide limits. For the PA-based limits, the total inventory for each radionuclide is divided by its corresponding limit. The sum of these fractions will be maintained less than one to ensure compliance with the limits. A similar procedure will be followed to ensure compliance with the SAR limits.

Requirements for pad disposal of Naval Reactor activated-metal components have been developed. They are:

1. Welds of material and thickness such that no less than 1 cm of the material remains after 500 years. A 1.25 inch weld on the representative carbon steel container meets this minimum requirement.

2. Casks and wasteforms of material and thickness such that complete corrosion will not occur until after 9,500 years.

3. External radiation dose at the exposed surfaces of the disposal cask after placement on the pad is less than 200 mrem/hr.

4. The curie ratio of ^{60}Co to ^{94}Nb is not greater than 16,000 (2.5×10^5 to 16).

Surface contaminated auxiliary equipment such as pumps and closure heads may be disposed on the NR Pad without the above container requirements.

THIS PAGE INTENTIONALLY LEFT BLANK

7.0 QUALITY ASSURANCE

The report underwent a design check (Cook, 2002b). The design check was intended to conform to WPT guidelines (WSRC, 2002).

THIS PAGE INTENTIONALLY LEFT BLANK

8.0 REFERENCES

40 CFR Part 141. *Code of Federal Regulations, Title 40, Protection of Environment,* Part 141, "National Primary Drinking Water Regulations."

49 CFR Part 173. *Code of Federal Regulations, Title 49, Transportation, Part 173*, "Shippers - General Requirements for Shipments and Packagings."

40 CFR Part 300. *Code of Federal Regulations, Title 40, Protection of Environment,* Part 300. "National Oil and Hazardous Substances Pollution Contingency Plan."

ACRI. 1998. *PORFLOW: A Model for Fluid Flow, Heat and Mass Transport in Multifluid, Multiphase Fractured or Porous Media, Users Manual, Version 4.00.7.* Analytic and Computational Research, Inc., Los Angeles, California.

Anderson, M. 2001. "On Contact Dose Rates for CB/TS Shipping/Storage/Disposal Casks Modeled in the Special Performance Assessment for the Naval Reactors Pad", Memorandum OEH-61310-MJA-01-0XX, Waste Engineering & Services, KAPL, Inc., November 9, 2001.

Cook, J. R. and E. L. Wilhite. 1998. *Radionuclide Screening and Preliminary Scoping Study for EAC Disposal of APT and TEF Wastes.* WSRC-RP-098-00084. Westinghouse Savannah River Company, Savannah River Site, Aiken, South Carolina.

Cook, J. R., E. L. Wilhite, K. E. Young. 2000. *Closure Plan for the E-Area Low-Level Waste Facility.* WSRC-RP-2000-00425. Westinghouse Savannah River Company, Savannah River Site, Aiken, South Carolina.

Cook, James R. 2002a. *Evaluation of Proposed New LLW Disposal Activity: Clarification of Disposal Requirements for Naval Reactor Components*, WSRC-RP-2002-00561.

Cook, James R. 2002b. *Design Check Package for Special Analysis: Naval Reactor Waste Disposal Pad*, WSRC-RP-2002-00441.

Dunn, K. A. 2002. *Corrosion of HY-80 Steel in SRS Soils*, SRT-MTS-2002-20056, Revision 1.

Freeze, R. A. and J. A. Cherry. 1979. *Groundwater*. Prentice-Hall, Inc., New Jersey.

Gough, A. R. 2002. *Request for a Legal Opinion: Use of EPA's Proposed MCLs*, letter to W. T. Goldston, March 13, 2002.

Hamby, D. M. 1994. *MAXINE: An Improved Methodology for Estimating Maximum Individual Dose from Chronic Atmospheric Radioactive Releases.* WSRC-TR-94-053. Westinghouse Savannah River Company, Savannah River Site, Aiken, South Carolina.

Hamby, D. M. 1995. *Verification of the MAXIGASP and POPGASP Computer Codes for Environmental Dose Assessment.* WSRC-RP-94-522. Westinghouse Savannah River Company, Savannah River Site, Aiken, South Carolina.

Hamm, L. L., S. E. Aleman, G. P. Flach, and W. F. Jones. 1996. *FACT (Version 1.0): Subsurface Flow and Contaminant Transport Documentation and User's Guide (U).* WSRC-TR-95-0223.

Savannah River Technology Center, Westinghouse Savannah River Company, Savannah River Site, Aiken, South Carolina.

Hanford. 1993. *Corrosion Behavior of HY-80 Steel, Type 304 Stainless Steel, and Inconel Alloy 500 at 218-E-12B Burial Ground, Hanford, WA.* Hanford report no. TR-2001-SHR, Hanford, Washington.

Hoeffner, S. L. 1984. *Additional Laboratory Studies on Radionuclide Sorption at the SRP Burial Ground.* DPST-84-797. E. I. du Pont de Nemours and Company, Savannah River Laboratory, Aiken, South Carolina.

Hoeffner, S. L. 1985. *Radionuclide Sorption on Savannah River Plant Burial Ground Soil: A Summary and Interpretation of Laboratory Data.* DP-1702. E. I. du Pont de Nemours and Company, Savannah River Laboratory, Aiken, South Carolina.

ICRP. 1959. *Recommendation of the International Commission on Radiological Protection.* International Commission on Radiological Protection. Publication No. 2. Pergamon Press, New York.

ICRP. 1979. *Limits on Intakes of Radionuclides by Workers.* International Commission on Radiological Protection. Publication 30, Part 1. *Ann. ICRP 2*. No.3/4. Pergamon Press, Oxford, U.K.

Kocher, D. C. 1981. *Radioactive Decay Data Tables.* DOE/TIC-11026. Technical Information Center, U.S. Department of Energy, Oak Ridge, Tennessee.

McDowell-Boyer, L., A. D. Yu, J. R. Cook, D. C. Kocher, E. L. Wilhite, H. Holmes-Burns, K. E. Young. 2000. *Radiological Performance Assessment for the E-Area Low-Level Waste Facility.* WSRC-RP-94-218, Revision 1. Westinghouse Savannah River Company, Savannah River Site, Aiken, South Carolina.

McIntyre, P. F. 1988. *Sorption Properties of Carbon-14 on Savannah River Plant Soil.* DPST-88-900. E. I. du Pont de Nemours and Company, Savannah River Laboratory, Aiken, South Carolina.

NCRP. 1976. *Structural Shielding Design and Evaluation for Medical Use of X Rays and Gamma Rays of Energies up to 10 MeV.* NCRP Report No. 49. National Council on Radiation Protection and Measurements, Bethesda, Maryland.

NCRP. 1996. *Screening Models for Releases of Radionuclides to Atmosphere, Surface, Water, and Ground, Vol. I and II.* NCRP Report No. 123. National Council on Radiation Protection and Measurements, Bethesda, Maryland.

NRC. 1997. *Calculation of Annual Doses to Man from Routine Release of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I.* Washington, D. C.

Oblath, S. B. 1982. *Migration of TcO_4- in SRP Soils.* DPST-82-815. E. I. DuPont de Nemours and Company, Savannah River Laboratory, Aiken, South Carolina.

Rosenberger, K. H. 1997. *Health Physics Technology Evaluation of Dose Rates Associated with Naval Reactor Waste Containers.* ESH-HPT-97-0058. Westinghouse Savannah River Company, Savannah River Site, Aiken, South Carolina.

Sheppard, M. I. and D. H. Thibault. 1990. Default Soil Solid/Liquid Partition Coefficients, Kds, for Four Major Soil Types: A Compendium. *Health Physics*, 59: 471-482.

Simpkins, A. A. 1998. *Modeling of Releases from the New Burial Ground, E Area.* SRT-EST-98-288. Westinghouse Savannah River Company, Savannah River Site, Aiken, South Carolina.

Simpkins, A. A. and W. H. Carlton. 1998. *Modeling of Releases from the New Burial Ground, E-Area - MEI at 100 m.* SRT-EST-98-333. Westinghouse Savannah River Company, Savannah River Site, Aiken, South Carolina.

Sullivan, T. M., C. R. Kempf, C. J. Suen, and S. M. Mughabghab. 1988. *Low-Level Radioactive Waste Source Term Model Development and Testing.* NUREG/CR-5204, BNL-NUREG-52160. Brookhaven National Laboratory, Upton, NY.

Ticknor, K. V. and B. Ruegger. 1989. *A Guide to the NEA's Sorption Data Base.* Version 2.0, 19 p.

Unger, L. M. and D. K. Trubey. 1982. *Specific Gamma-Ray Dose Constants for Nuclides Important to Dosimetry and Radiological Assessment.* ORNL/TSIC-45/R1. Oak Ridge National Laboratory, Oak Ridge, Tennessee.

U.S. Department of Commerce. 1963. *Maximum Permissible Body Burdens and Maximum Permissible Concentrations of Radionuclides in Air and Water for Occupational Exposure.* National Bureau of Standards Handbook 69. NCRP Report 22.

USDOE. 1988. *Internal Dose Conversion Factors for Calculation of Dose to the Public.* DOE/EH-0071. U.S. Department of Energy, Washington, D.C.

USDOE 1996. *Interim Format and Content Guide and Standard Review Plan for U.S. Department of Energy Low-Level Waste Disposal Facility Performance Assessments.* October 1996. U.S. Department of Energy, Washington, D.C.

USDOE. 1999. *Radioactive Waste Management.* Order 435.1. U.S. Department of Energy, Washington, D.C.

USEPA. 2000. 40 CFR Parts 9, 141, and 142 – National Primary Drinking Water Regulations; Radionuclides; Final Rule. *Federal Register, 65*:76707.

USEPA. 2001. *Implementation Guidance for Radionuclides*. Draft Report EPA 816-D-00-002, August 2001.

WSRC. 2001. *WSRC 1S Savannah River Site Waste Acceptance Criteria Manual*, Procedure WAC-OG 1, Low-Level Radioactive Waste Acceptance Criteria for Naval Reactor Generators, Rev. 2, September 2001.

WSRC. 2002. *Technical Report Design Check Guidelines*, WSRC-IM-2002-00011.

Yu, A. D., C. A. Langton, and M. G. Serrato. 1993. *Physical Property Measurement Program.* WSRC-RP-93-849. Westinghouse Savannah River Company, Savannah River Site, Aiken, South Carolina.

APPENDIX A

PORFLOW INPUT FILE FOR

UNSATURATED-ZONE FLOW MODELING

THIS PAGE INTENTIONALLY LEFT BLANK

Appendix A

PORFLOW Input File for Unsaturated-Zone Flow Modeling

```
*****    \NR01\SAFE\UNSAT\FLOW\RUN.DAT     *****

/

/---------------------------------------------------------------------

/
/     NAVAL REACTOR COMPONENTS PERFORMANCE ASSESSMENT

/       FLOW FIELD FOR THE 60-FT VADOSE ZONE SCENARIO

/
/     A. D. YU
/     1/22/01
/

/---------------------------------------------------------------------

/

TITLE ***** NR 2001 PA REVISION - 60-FT VADOSE ZONE *****
/
GRID  37  BY  47
/
COOR X Y 'GRID.POR'
/
/NATIVE SOIL
MATErial type  1 from      1    1   to   37   47
/
/BACKFILL
MATErial type  2 from     1   30   to   37   47
/
/WASTE
MATErial type  3 from     1   30   to   26   40
/
/=====================================================================
/
FOR 1   native soil
MATErial DENSity 2.65
MATErial POROsity = .42  .42  .42
TRANsport  for C Kd= 0. D= 158  al= 0.  at= 0.
HYDR  ss = 1.e-3  kx = 315.4  ky=315.4  kz=315.4

MULT  TABL  25  /NATIVE
   4.000006E-01   3.999999E+11
   4.006000E-01   3.999999E+06
   4.060000E-01   8.617713E+04
   4.120000E-01   2.714367E+04
   4.300000E-01   5.893200E+03
   4.450000E-01   2.996996E+03
   4.600000E-01   1.854134E+03
   4.750000E-01   1.276873E+03
   4.900000E-01   9.408308E+02
   5.050000E-01   7.261895E+02
   5.200000E-01   5.797906E+02
   5.500000E-01   3.968953E+02
   5.800000E-01   2.899679E+02
   6.100000E-01   2.212632E+02
   6.400000E-01   1.740319E+02
   7.000000E-01   1.140935E+02
   7.600000E-01   7.789433E+01
   8.200000E-01   5.341967E+01
   8.800000E-01   3.514850E+01
   9.400000E-01   1.979191E+01
```

```
   9.580000E-01   1.523891E+01
   9.760000E-01   1.035160E+01
   9.910000E-01   5.443573E+00
   9.994000E-01   9.857911E-01
   1.000000E+00   0.000000E+00
/
MULT  COND  TABL  25  /NATIVE
   4.000006E-01   0.000000E+00
   4.006000E-01   3.101070E-04
   4.060000E-01   3.110756E-03
   4.120000E-01   6.243268E-03
   4.300000E-01   1.577517E-02
   4.450000E-01   2.387837E-02
   4.600000E-01   3.213413E-02
   4.750000E-01   4.054967E-02
   4.900000E-01   4.913280E-02
   5.050000E-01   5.789193E-02
   5.200000E-01   6.683617E-02
   5.500000E-01   8.532041E-02
   5.800000E-01   1.046757E-01
   6.100000E-01   1.250099E-01
   6.400000E-01   1.464541E-01
   7.000000E-01   1.933582E-01
   7.600000E-01   2.472711E-01
   8.200000E-01   3.114942E-01
   8.800000E-01   3.928174E-01
   9.400000E-01   5.102211E-01
   9.580000E-01   5.614891E-01
   9.760000E-01   6.313292E-01
   9.910000E-01   7.279879E-01
   9.994000E-01   8.825107E-01
   1.000000E+00   1.000000E+00
/
FOR 2 TO 3
MATErial DENSity 2.65
MATErial POROsity = .42  .42  .42
TRANsport  for C Kd= 0. D= 158  al= 0.  at= 0.
HYDR  ss = 1.e-3  kx = 315.4  ky=315.4  kz=315.4

MULT  TABL  25  /NATIVE
   4.000006E-01   3.999999E+11
   4.006000E-01   3.999999E+06
   4.060000E-01   8.617713E+04
   4.120000E-01   2.714367E+04
   4.300000E-01   5.893200E+03
   4.450000E-01   2.996996E+03
   4.600000E-01   1.854134E+03
   4.750000E-01   1.276873E+03
   4.900000E-01   9.408308E+02
   5.050000E-01   7.261895E+02
   5.200000E-01   5.797906E+02
   5.500000E-01   3.968953E+02
   5.800000E-01   2.899679E+02
   6.100000E-01   2.212632E+02
   6.400000E-01   1.740319E+02
   7.000000E-01   1.140935E+02
   7.600000E-01   7.789433E+01
   8.200000E-01   5.341967E+01
   8.800000E-01   3.514850E+01
   9.400000E-01   1.979191E+01
   9.580000E-01   1.523891E+01
   9.760000E-01   1.035160E+01
   9.910000E-01   5.443573E+00
   9.994000E-01   9.857911E-01
   1.000000E+00   0.000000E+00
/
MULT  COND  TABL  25  /NATIVE
   4.000006E-01   0.000000E+00
   4.006000E-01   3.101070E-04
   4.060000E-01   3.110756E-03
   4.120000E-01   6.243268E-03
```

```
   4.300000E-01   1.577517E-02
   4.450000E-01   2.387837E-02
   4.600000E-01   3.213413E-02
   4.750000E-01   4.054967E-02
   4.900000E-01   4.913280E-02
   5.050000E-01   5.789193E-02
   5.200000E-01   6.683617E-02
   5.500000E-01   8.532041E-02
   5.800000E-01   1.046757E-01
   6.100000E-01   1.250099E-01
   6.400000E-01   1.464541E-01
   7.000000E-01   1.933582E-01
   7.600000E-01   2.472711E-01
   8.200000E-01   3.114942E-01
   8.800000E-01   3.928174E-01
   9.400000E-01   5.102211E-01
   9.580000E-01   5.614891E-01
   9.760000E-01   6.313292E-01
   9.910000E-01   7.279879E-01
   9.994000E-01   8.825107E-01
   1.000000E+00   1.000000E+00
/
/=====================================================================
/
DATUm = 0. 0.
GRAVity components are: 0.0, -1.
/
BOUNdary P at X- FLUX = 0.0

BOUNdary P at X+ FLUX = 0.0

BOUNdary P at Y- VALU = 0.0

BOUNdary P at Y+ FLUX = 40.

/
LOCAte subregion (  1,  1) to ( 37,  47)
SET SELEct S = 0.65
LOCAte subregion (  1,  1) to ( 37,   3)
SET SELEct S = 1.00
LOCAte subregion (  1,  4) to ( 37,   6)
SET SELEct S = 0.90
LOCAte subregion (  1,  7) to ( 37,   9)
SET SELEct S = 0.80
LOCAte subregion (  1, 10) to ( 37,  12)
SET SELEct S = 0.70
/

PROPerty for P is HARMonic

MATRix for P NSPC CONJ CHOL SYMM
/

DIAG TIME P U V S at (10,30) every 100 steps

OUTPut every 900000 steps

/
TIME = 0. years

CONVergence for FLOW 1.e-3 max iterations = 100
CONVergence for P    1.e-5 max iterations = 300
SOLVe P 2000 dt  1.e-5 inc 1.001 max 0.1

SAVE U V S in 'FlowUns.ARC' NOW
/
END
```

THIS PAGE INTENTIONALLY LEFT BLANK

APPENDIX B

PORFLOW INPUT FILE FOR

UNSATURATED-ZONE ^{14}C TRANSPORT MODELING

THIS PAGE INTENTIONALLY LEFT BLANK

Appendix B

PORFLOW Input File for Unsaturated-Zone ^{14}C Transport Modeling

```
**COMPONENT = C-14
/
TITLE 2001 NR PA Unsaturated-Zone Activation
GRID 37 by 47
/
COOR X Y 'GRID.POR'
/
READ 1 'FlowUns.ARC' STARt
/
/NATIVE SOIL
MATErial type  1 from      1    1   to   37   47
/
/BACKFILL
MATErial type  2 from     1   30   to   37   47
/
/WASTE
MATErial type  3 from     1   30   to   26   40
/
FOR 1  Native Soil
MATErial  DENSity 2.65
MATErial  POROsity = .42  .42  .42
TRAN for C  Kd= 2.00E+00 diff= 1.58E+02 al= 0 at= 0
/
FOR 2  Backfill
MATErial  DENSity 2.65
MATErial  POROsity = .42  .42  .42
TRAN for C  Kd= 2.00E+00 diff= 1.58E+02 al= 0 at= 0
/
FOR 3  waste
MATErial  DENSity 2.65
MATErial  POROsity = .42  .42  .42
TRAN for C  Kd= 2.00E+00 diff= 1.58E+02 al= 0 at= 0
/
ALLO C5
ALLO C6
DECA HALF LIFE for C  5.73E+03 year
/
LOCA ( 1, 1) to (37,47) ID=DOMA
LOCA ( 1,30) to (26,40) ID=WAST
/
BOUN C  X- FLUX= 0.
BOUN C  X+ FLUX= 0.
BOUN C  Y- VALU= 0.
BOUN C  Y+ FLUX= 0.
/
SOUR C ID=WAST VOLU TABL   42 sets
      7.500E+02      1.208E-05
      9.756E+02      1.176E-05
      1.201E+03      1.144E-05
      1.427E+03      1.113E-05
      1.652E+03      1.083E-05
      1.878E+03      1.054E-05
      2.104E+03      1.026E-05
      2.329E+03      9.980E-06
      2.555E+03      9.712E-06
      2.780E+03      9.450E-06
      3.006E+03      9.196E-06
      3.232E+03      8.948E-06
      3.457E+03      8.707E-06
      3.683E+03      8.473E-06
      3.909E+03      8.245E-06
      4.134E+03      8.023E-06
      4.360E+03      7.807E-06
      4.585E+03      7.597E-06
      4.811E+03      5.517E-06
      5.037E+03      5.369E-06
```

```
      5.262E+03      5.224E-06
      5.488E+03      5.084E-06
      5.713E+03      4.947E-06
      5.939E+03      4.814E-06
      6.165E+03      4.684E-06
      6.390E+03      4.558E-06
      6.616E+03      4.435E-06
      6.841E+03      4.316E-06
      7.067E+03      4.200E-06
      7.293E+03      4.086E-06
      7.518E+03      3.976E-06
      7.744E+03      3.869E-06
      7.970E+03      3.765E-06
      8.195E+03      3.664E-06
      8.421E+03      3.565E-06
      8.646E+03      3.469E-06
      8.872E+03      3.376E-06
      9.098E+03      3.285E-06
      9.323E+03      3.197E-06
      9.549E+03      3.110E-06
      9.774E+03      3.027E-06
      1.000E+04      2.945E-06
/
PROPerty for C C2 C3 C4 C5 is HARMonic
MATRix in X and Y for C C2 C3 C4 C5 in 3 sweep using ADI
/
DIAG TIME S C C2 C3 at ( 12,2) every  1000 steps
OUTPut every 900000 steps
/
TIME = 750. years
CONVergence for C REFE GLOBal 1.e-3, max iterations = 30
DISAble FLOW
/
FLUX C  ID=DOMA 'RUN.FLX' TIME 5.0E+00 yr
FLUX C  ID=WAST 'RUN.FLX' TIME 5.0E+00 yr
SOLV C     9250 dt 1.0E-04 inc 1.001 max 5.00E-01
SAVE U V S C in 'END.ARC' NOW
/
END
```

APPENDIX C

PORFLOW INPUT FILE FOR THE "CRUD"

UNSATURATED-ZONE NO_3 TRANSPORT MODELING

THIS PAGE INTENTIONALLY LEFT BLANK

Appendix C

PORFLOW Input File for the "Crud" Unsaturated-Zone NO_3 Transport Modeling

```
**COMPONENT = NO3 (Conservative tracer)
/
TITLE 2001 NR PA Unsaturated-Zone CRUD
GRID 37 by 47
/
COOR X Y 'GRID.POR'
/
READ 1 'FlowUns.ARC' STARt
/
/NATIVE SOIL
MATErial type  1 from      1    1   to   37   47
/
/BACKFILL
MATErial type  2 from     1   30   to   37   47
/
/WASTE
MATErial type  3 from     1   30   to   26   40
/
/=====================================================================
/
FOR 1  Native Soil
MATErial  DENSity 2.65
MATErial  POROsity = .42  .42  .42
TRAN for C  Kd= 0.00E+00 diff= 1.58E+02 al= 0 at= 0
/
FOR 2  Backfill
MATErial  DENSity 2.65
MATErial  POROsity = .42  .42  .42
TRAN for C  Kd= 0.00E+00 diff= 1.58E+02 al= 0 at= 0
/
FOR 3  waste
MATErial  DENSity 2.65
MATErial  POROsity = .42  .42  .42
TRAN for C  Kd= 0.00E+00 diff= 1.58E+02 al= 0 at= 0
/
/=====================================================================
/
ALLO C5
ALLO C6
DECA HALF LIFE for C  1.00E+30 year
/
LOCA ( 1, 1) to (37,47) ID=DOMA
LOCA ( 1,30) to (26,40) ID=WAST
/
BOUN C  X- FLUX= 0.
BOUN C  X+ FLUX= 0.
BOUN C  Y- VALU= 0.
BOUN C  Y+ FLUX= 0.
/
/
SET INVE C   1.0000E+06 UNIF  ID=WAST
/
PROPerty for C C2 C3 C4 C5 is HARMonic
MATRix in X and Y for C C2 C3 C4 C5 in 3 sweep using ADI
/
DIAG TIME S C C2 C3 at ( 12,2) every  100 steps
OUTPut every 900000 steps
/
TIME = 750. years
CONVergence for C REFE GLOBal 1.e-3, max iterations = 30
DISAble FLOW
/
FLUX C  ID=DOMA 'RUN.FLX' TIME 5.00E-02 yr
FLUX C  ID=WAST 'RUN.FLX' TIME 5.00E-02 yr
SOLV C       50 dt 1.00E-04 inc 1.001 max 5.00E-03
```

```
SAVE U V S C in 'END.ARC' NOW
/
END
```

APPENDIX D

PORFLOW INPUT FILE FOR THE "ACTIVATION"

UNSATURATED-ZONE NO_3 TRANSPORT MODELING

THIS PAGE INTENTIONALLY LEFT BLANK

Appendix D

PORFLOW Input File for the "Activation" Unsaturated-Zone NO_3 Transport Modeling

```
**COMPONENT = NO3   (conservative tracer)
/
TITLE 2001 NR PA Unsaturated-Zone ACTI
GRID 37 by 47
/
COOR X Y 'GRID.POR'
/
READ 1 'FlowUns.ARC' STARt
/
/NATIVE SOIL
MATErial type  1 from      1    1   to   37   47
/
/BACKFILL
MATErial type  2 from     1   30   to   37   47
/
/WASTE
MATErial type  3 from     1   30   to   26   40
/
/=====================================================================
/
FOR 1  Native Soil
MATErial  DENSity 2.65
MATErial  POROsity = .42  .42  .42
TRAN for C  Kd= 0.00E+00 diff= 1.58E+02 al= 0 at= 0
/
FOR 2  Backfill
MATErial  DENSity 2.65
MATErial  POROsity = .42  .42  .42
TRAN for C  Kd= 0.00E+00 diff= 1.58E+02 al= 0 at= 0
/
FOR 3  waste
MATErial  DENSity 2.65
MATErial  POROsity = .42  .42  .42
TRAN for C  Kd= 0.00E+00 diff= 1.58E+02 al= 0 at= 0
/
/=====================================================================
/
ALLO C5
ALLO C6
DECA HALF LIFE for C  1.00E+30 year
/
LOCA ( 1, 1) to (37,47) ID=DOMA
LOCA ( 1,30) to (26,40) ID=WAST
/
BOUN C  X- FLUX= 0.
BOUN C  X+ FLUX= 0.
BOUN C  Y- VALU= 0.
BOUN C  Y+ FLUX= 0.
/
/
SOUR C  ID=WAST VOLU TABL   51 sets
  7.5000E+02  8.5428E-04
  7.7100E+02  8.5428E-04
  7.9200E+02  8.5428E-04
  8.1300E+02  8.5428E-04
  8.3400E+02  8.5428E-04
  8.5500E+02  8.5428E-04
  8.7600E+02  8.5428E-04
  8.9700E+02  8.5428E-04
  9.1800E+02  8.5428E-04
  9.3900E+02  8.5428E-04
  9.6000E+02  8.5428E-04
  9.8100E+02  8.5428E-04
  1.0020E+03  8.5428E-04
```

```
  1.0230E+03  8.5428E-04
  1.0440E+03  8.5428E-04
  1.0650E+03  8.5428E-04
  1.0860E+03  8.5428E-04
  1.1070E+03  8.5428E-04
  1.1280E+03  8.5428E-04
  1.1490E+03  8.5428E-04
  1.1700E+03  8.5428E-04
  1.1910E+03  8.5428E-04
  1.2120E+03  8.5428E-04
  1.2330E+03  8.5428E-04
  1.2540E+03  8.5428E-04
  1.2750E+03  8.5428E-04
  1.2960E+03  8.5428E-04
  1.3170E+03  8.5428E-04
  1.3380E+03  8.5428E-04
  1.3590E+03  8.5428E-04
  1.3800E+03  8.5428E-04
  1.4010E+03  8.5428E-04
  1.4220E+03  8.5428E-04
  1.4430E+03  8.5428E-04
  1.4640E+03  8.5428E-04
  1.4850E+03  8.5428E-04
  1.5060E+03  8.5428E-04
  1.5270E+03  8.5428E-04
  1.5480E+03  8.5428E-04
  1.5690E+03  8.5428E-04
  1.5900E+03  8.5428E-04
  1.6110E+03  8.5428E-04
  1.6320E+03  8.5428E-04
  1.6530E+03  8.5428E-04
  1.6740E+03  8.5428E-04
  1.6950E+03  8.5428E-04
  1.7160E+03  8.5428E-04
  1.7499E+03  8.5428E-04
  1.7501E+03  0.0000E+00
  1.7790E+03  0.0000E+00
  1.8000E+03  0.0000E+00
/
PROPerty for C C2 C3 C4 C5 is HARMonic
MATRix in X and Y for C C2 C3 C4 C5 in 3 sweep using ADI
/
DIAG TIME S C C2 C3 at ( 12,2) every  100 steps
OUTPut every 900000 steps
/
TIME = 750. years
CONVergence for C REFE GLOBal 1.e-3, max iterations = 30
DISAble FLOW
/
FLUX C  ID=DOMA 'RUN.FLX' TIME 1.05E+00 yr
FLUX C  ID=WAST 'RUN.FLX' TIME 1.05E+00 yr
SOLV C     1050 dt 1.00E-04 inc 1.001 max 1.05E-01
SAVE U V S C in 'END.ARC' NOW
/
END
```

APPENDIX E

PORFLOW INPUT FILE FOR THE "CRUD"

SATURATED-ZONE NO_3 TRANSPORT MODELING

THIS PAGE INTENTIONALLY LEFT BLANK

Appendix E

PORFLOW Input File for the "Crud" Saturated-Zone NO_3 Transport Modeling

```
**COMPONENT = NO3  (conservative tracer)
/
TITLE NAVAL REACTOR COMPONENTS SATURATED ZONE MODELING
GRID 27 by 27 by 14
/
INCL '\NR01\GSA\2001\GridNR01\Coord\Coord.out'
READ 1 '\NR01\GSA\2001\GridNR01\Archive\Flow.ARC'  start
MATErial type 1 from (1,1,1) to (27,27,14)
INCL '\NR01\GSA\2001\GridNR01\KdZone\KdZone.out'
/
/ Material Type 3 for 2001 Naval Source Nodes
MATErial type 3 from  11  12   5  to  11  12   5
MATErial type 3 from  11  13   5  to  11  13   5
MATErial type 3 from  12  13   5  to  12  13   5
/
/FOR 1  thru  10
/units for D is in ft^2/yr, rock density in g/cm^3 and Kd in cm^3/g
/We use rho [=] gm/cc so that Kd [=] cc/g
/D=5.E-6 cm^2/sec = 5.E-6*365*24*3600/30.48^2 = 0.170 ft^2/yr
/al and at [=] ft
/We set al = at = 0. because we have large numerical dispersion
/and this is conservative.
/kx, ky, and kz (ft/yr) are not needed because we disable flow
/
FOR 1  thru 3 Native Soil - Aquifer
MATErial  DENSity 2.65
MATErial  POROsity = .25  .25  .25
TRAN for C  Kd= 0.00E+00 diff= 1.70E-01 al= 0 at= 0
/
FOR 2  Clay
MATErial  DENSity 2.65
MATErial  POROsity = .25  .25  .25
TRAN for C  Kd= 0.00E+00 diff= 8.50E-02 al= 0 at= 0
/
ALLO C5
ALLO C6
DECA HALF LIFE for C  1.00E+30 year
/
LOCA ( 1, 1, 1) to (27,27,14) ID=DOMA
LOCA material type 3 as subregion ID=NAVY
/
BOUN C  X- VALU= 0.
BOUN C  X+ VALU= 0.
BOUN C  Y- VALU= 0.
BOUN C  Y+ VALU= 0.
BOUN C  Z- VALU= 0.
BOUN C  Z+ VALU= 0.
/
INCL 'SOURCE.DAT'
/
SET S 1.0 ID=DOMA
/
PROPerty for C is HARMonic
MATRix in X and Y and Z for C in 3 sweeps using ADI
/
DIAG TIME S C C2 C3 at (12,15,5) every  100 steps
OUTPut every 900000 steps
/
HISTory at
     (11,15,6) (11,15, 5) (11,15, 4) (11,15, 3) (11,15, 2)
     (12,15,6) (12,15, 5) (12,15, 4) (12,15, 3) (12,15, 2)
     (13,15,6) (13,15, 5) (13,15, 4) (13,15, 3) (13,15, 2)
HISTory for C 'RUN.HIS' TABLE at
      TIME =    0.03
```

```
/
TIME     752
CONVergence for C REFE GLOBal 1.e-3, max iterations = 1
DISAble FLOW
/
FLUX C  ID=DOMA 'RUN.FLX' TIME 2.50E-02 yr
SOLV C    25.0000 dt 2.00E-05 inc 1.010 max 2.000E-04
SAVE C in 'END.ARC' NOW
/
END
```