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SPECIAL ANALYSIS:

REEVALUATION OF THE INADVERTENT INTRUDER, GROUNDWATER, AIR, and RADON ANALYSES FOR THE SALTSTONE DISPOSAL FACILITY

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TABLE OF CONTENTS

LIST OF FIGURES

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LIST OF TABLES

LIST OF ACRONYMS AND ABBREVIATIONS

ACRONYMS

ARAR	Applicable or Relevant and Appropriate Requirements
CERCLA	Comprehensive Environmental Response, Compensation, and Liability Act
CFR	Code of Federal Regulations
DAS	Disposal Authorization Statement
DWPF	Defense Waste Processing Facility
EDE	effective dose equivalent
ELLWF	E-Area Low-Level Waste Facility
ETF	Effluent Treatment Facility
HLW	high-level waste
ICRP	International Commission on Radiological Protection
IL	Intermediate Level
LAW	Low-Activity Waste
LCS	low curie salt
MCL	Maximum Contaminant Level
MMES	Martin Marietta Energy Systems
NBS	National Bureau of Standards
NCRP	National Council on Radiation Protection and Measurements
NPL	National Priorities List
ORNL	Oak Ridge National Laboratory
PA	Performance Assessment
QA	Quality Assurance
RPA	Radiological Performance Assessment
SDCF	scenario dose conversion factor
SDF	Saltstone Disposal Facility
SPF	Saltstone Production Facility
SRS	Savannah River Site
USDA	United States Department of Agriculture
USDOE	United States Department of Energy
USEPA	United States Environmental Protection Agency
USNRC	United States Nuclear Regulatory Commission
WAC	Waste Acceptance Criteria
WSRC	Westinghouse Savannah River Company

ABBREVIATIONS

Bq	becquerel
cc	cubic centimeter
Ci	curie
cm	centimeter
g	gram
gal	gallon
h	hour
kg	kilogram
km	kilometer
L	liter
m	meter
μCi	microcurie
mg	milligram
min	minute
mm	millimeter
mrem	millirem
mSv	millisievert
nCi	nanocurie
Pa	Pascal
pCi	picocurie
s	second
Sect.	Section
Sv	sievert
wt	weight
y	year

PREFACE

This document was prepared jointly by the authors listed on the title page. However, not all authors contributed to each section, and the contributorship is described below. James R. Cook performed the groundwater, air, and radon analyses, authored Sections 5, 6, and 7, and co-authored Sections 1, 2, and 3. David C. Kocher performed the intruder analysis, and authored Section 4 and the information in Appendix A. Laura McDowell-Boyer performed QA checks on the intruder analysis (results presented in Appendix B) and pertinent tables of Section 8, and co-authored Sections 1, 2, and 3. Elmer L. Wilhite authored Section 8 and served as technical editor for the entire document.

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1. EXECUTIVE SUMMARY

This Special Analysis updates the inadvertent intruder analysis conducted in 1992 in support of the SDF RPA, extends the groundwater analysis to consider additional radionuclides, and provides an assessment of the air and radon emanation pathways. The results of the RPA were originally published in the WSRC report (WSRC-RP-92-1360) entitled *Radiological Performance Assessment for the Z-Area Saltstone Disposal Facility* (MMES et al., 1992). The present reevaluation considers new requirements and guidance of the USDOE Order 435.1 (USDOE, 1999), expands the list of radionuclides considered, incorporates an increase in design thickness of the roof on a disposal vault, and produces results in terms of interim limits on radionuclide-specific concentration and inventory rather than dose resulting from a projected inventory. The limits derived herein will be updated when the Saltstone PA is revised (currently planned for fiscal years 2003/2004).

The SDF is located within a 650,000 m^2 area of SRS designated as Z Area. The SDF together with the SPF are part of an integrated waste treatment and disposal system at the SRS. Saltstone is a solid waste form that is the product of chemical reactions between a salt solution and a blend of cementitious materials (slag, flyash, and cement). Based on the present projected site layout of the SDF, up to 730-million L (192 million gal) of wastewater can be treated for subsequent disposal as saltstone. The SPF and SDF are regulated by the State of South Carolina, USDOE Orders, and other Federal regulations that are applicable to disposal of solid waste.

As part of the RPA process, USDOE Order 435.1 requires an assessment of the dose to a potential member of the general public to limit doses from all pathways to no more than 25 mrem in a year and, from the air pathway alone, to no more than 10 mrem in a year. The Order also requires an assessment of radon release to ensure that the radon flux does not exceed 20 $pCi/m^2/s$. Additionally, for purposes of establishing limits on concentrations of radionuclides for disposal, the Order requires that an assessment be made of impacts to hypothetical persons assumed to inadvertently intrude into the low-level waste disposal facility and an assessment of the impacts to water resources. For the intruder analysis, the pertinent performance measure specifies that dose to such hypothetical individuals may not exceed 100 mrem EDE per year for chronic exposure, and may not exceed 500 (EDE) mrem from a single event. To meet the assessment requirement addressing impact on water resources in the Order, SRS uses the Safe Drinking Water Act Maximum Contaminant Levels (USEPA, 2000) as the pertinent performance measure.

To limit the number of radionuclides for which analyses are needed, the half-lives of radionuclides and physical processes by which low-level waste destined for the SDF is generated were considered. Such considerations led to selection of 75 radionuclides for analysis. Potentially significant contributions by radioactive decay products of these 75 radionuclides were also assessed.

Two time frames for the analyses are considered in this Special Analysis. The USDOE Order 435.1 specifies a time frame of 1,000 years after facility closure for establishing limits on allowable disposals. Here, both the 1,000-y time frame and a longer time frame of 10,000

years after facility closure are also considered, to be consistent with both the USDOE Order and the Disposal Authorization Statement (DAS) for SRS (Fiori and Frei, 1999).

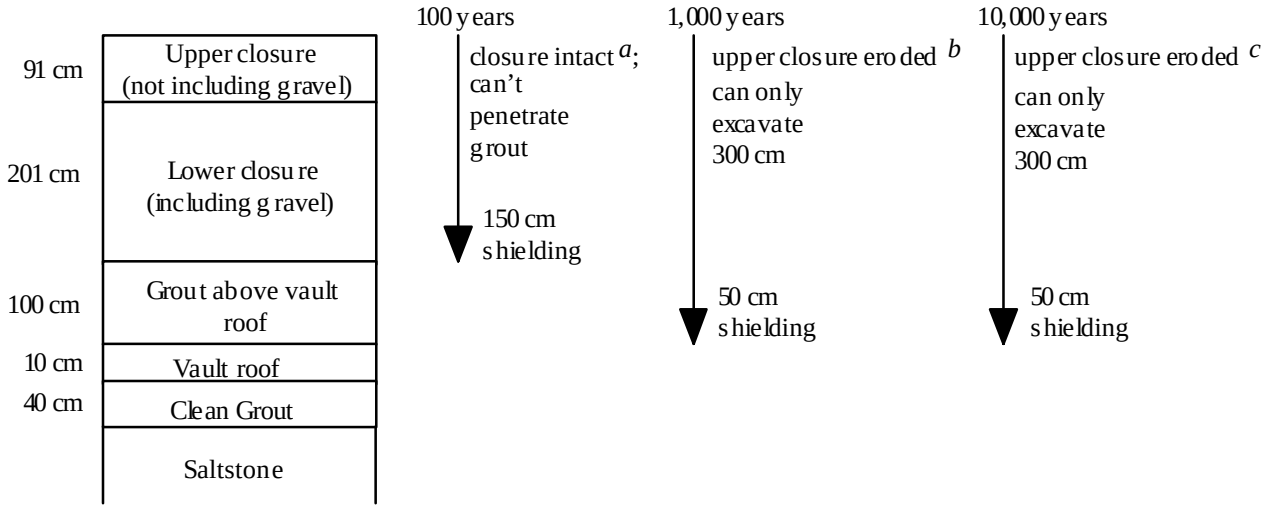
In the intruder analysis, the only credible scenario within 10,000 years is the resident scenario, based on the current design of the SDF. The 0.4 m of grout directly above the saltstone, 0.1-m concrete roof over the vaults, and 1 m of grout on top of the roof combine to provide at least 0.5-m of shielding up to 10,000 years, assuring that excavation into the waste during this time period is not a credible occurrence (Fig. 1-1). The resident scenario is evaluated at 100, 1,000, and 10,000 years after disposal. In the resident scenario, the intruder is assumed to excavate no more than 3 meters in building a home. Evaluation of the scenario at 100 years, when the engineered barriers (i.e., the grout above the saltstone, the vault roof, and the grout above the roof) are assumed to be intact, resulting in the intruder's home being constructed on top of the uppermost layer of grout, is used to determine limits on allowable disposals of shorter-lived photon-emitting radionuclides in the waste. Evaluation of the resident scenario at 1,000 and 10,000 years, when the engineered barriers are assumed to have failed (i.e., have lost their physical integrity) and are no longer a deterrent to intrusion, resulting in a lesser thickness of shielding above the waste, is used to determine limits on allowable disposals of longer-lived photon-emitting radionuclides. The thickness of uncontaminated material above the waste is the same at these two later times because the upper 0.9 m of the closure has eroded (Fig. 1-1) and the depth of the intruder's excavation is limited to 3 m. The resident scenario at 1,000 years may be important for radionuclides having longer-lived photon-emitting decay products. The resident scenario at 10,000 years is important only when a longer-lived radionuclide has long-lived photon-emitting decay products whose activities increase with time beyond 1,000 years.

For the groundwater, air, and radon emanation pathways, results from the previous SDF PA and applicable portions of the E-Area LLWF PA were used to derive limits on allowable disposals based on analyses for time frames of 1,000 years and 10,000 years after facility closure. For the groundwater pathway, it was necessary to extend the previous analysis in the SDF PA to radionuclides not previously considered, using the PATHRAE code.

The results of this Special Analysis indicate that, for the 10,000-year time frame, 41 radionuclides, of the 75 selected, require limits on disposal. Of the 41 radionuclides for which disposal limits were derived, 34 are limited by the intruder analysis, four by the groundwater pathway analysis, two by the air pathway analysis, and one by the radon emanation analysis. The radionuclide disposal limits were compared with the currently estimated radionuclide concentrations in low curie salt. The greatest fraction of a limit is 0.038 for ^{126}Sn and the total sum-of-fractions of all the limits is 0.084. This provides assurance that low curie salt can be disposed in the saltstone disposal facility without exceeding any of the USDOE performance objectives.

For the 1,000-year time frame, 37 of the 75 radionuclides would require disposal limits. Of these, 35 would be limited by the intruder analysis, none by the groundwater analysis, two by the air pathway analysis, and none by the radon emanation analysis. The greatest fraction of a limit would remain 0.038 for ^{126}Sn and the total sum-of-fractions would decrease to 0.048.

The 10,000-year time frame limits should be used to develop WAC for the SDF.



[a] At 100 years after closure, there has been no erosion and the grout and vault roof have not deteriorated so that they effectively prevent excavation. Therefore, the intruder constructs his residence atop the grout above the vault roof, resulting in a total of 150 cm of shielding between the residence and the saltstone.

[b] At 1,000 years after closure, erosion has removed the upper 91 cm of the closure. However, the gravel, which is the uppermost portion of the lower closure, prevents further erosion. The grout and vault roof have deteriorated to soil equivalent material so that they no longer can prevent excavation. Since the intruder's excavation is limited to 300 cm, the residence is constructed on top of the vault roof, resulting in a total of 50 cm of shielding between the residence and the saltstone.

[c] At 10,000 years after closure, erosion has not penetrated further than at 1,000 years (i.e., 91 cm), because of the gravel layer. Since the intruder's excavation is limited to 300 cm, the residence is constructed on top of the vault roof, resulting in a total of 50 cm of shielding between the residence and the saltstone.

Fig. 1-1. Resident Scenario Conceptual Model

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2. INTRODUCTION

The present study is a reevaluation of the inadvertent intruder analysis and an extension of the groundwater pathway analysis conducted for the RPA of the SDF located within Z Area at SRS. This study also provides an evaluation of the air and radon emanation pathways. The original RPA for this facility, prepared in accordance with the requirements of Chapter III of USDOE Order 5820.2A, was issued in December 1992 (MMES et al., 1992) and received conditional USDOE approval in February 1998. The report herein is supplemental to this earlier document.

The purpose of this reevaluation and extension is to incorporate new requirements and guidance in Chapter IV of USDOE Order 435.1, as well as update the analyses to reflect any changes in methodology and data that are deemed more appropriate at this time. In particular, the interpretation of time of compliance has been reevaluated, the list of radionuclides considered has been greatly expanded, the performance measure for groundwater protection has been revised, disposal limits on average concentrations and inventories rather than estimated doses are calculated, the design thickness of the roof on a disposal vault has been increased, and some updated dose factors are being used.

To understand the context of the present Special Analysis, information pertinent to the performance assessment in general, and more specifically to the SDF, is briefly reviewed in Sect. 2.1 and 2.2 below. Descriptions of the performance criteria and associated points of compliance are presented in Sect. 2.3. Interim (i.e., until the RPA is revised, which is expected in fiscal years 2003/2004) disposal limits for individual radionuclides are developed based on the analyses conducted and the performance criteria. The interim limits are compared with the currently expected radionuclide concentrations in low curie salt solution feed to saltstone. Throughout this report, there are references to the original RPA by section to facilitate locating pertinent information in the reference document.

2.1 APPROACH TO PERFORMANCE ASSESSMENT

The original Z-Area SDF RPA was developed using USDOE requirements and guidance for performance assessments specified in Chapter III of USDOE Order 5820.2A (USDOE, 1988). In 1999, USDOE issued Order 435.1 (USDOE, 1999a), replacing Order 5820.2A, which provides an updated set of requirements and guidance for performance assessments, which are specified in Chapter IV of the later Order. The present study was conducted according to the requirements and guidance of this most recent Order.

The results of this Special Analysis are presented in terms of limits on average concentration and inventory of individual radionuclides with respect to inadvertent intruders, and the groundwater, air, and radon emanation pathways. For inadvertent intruders, the inventory limit is determined by comparing calculated annual doses per unit activity concentration of each radionuclide considered in the wasteform with the dose limits specified in the USDOE Order as performance measures for these hypothetical individuals. For the groundwater pathway, inventory limits are derived by comparing calculated groundwater concentrations at a designated point of compliance with the performance measures for both the all-pathways objective and the assessment requirement addressing impacts on water resources. For the air

pathway, calculated doses are compared with the performance measure specified in the USDOE Order. Finally, for the radon emanation pathway, a limit on ^{234}U inventory is determined by comparing estimated emanation rates of radon with the USDOE performance measure for that objective. The level of technical detail presented in this report is sufficient to allow a reviewer to reproduce the results of the calculations.

2.2 GENERAL BACKGROUND ON THE SALTSTONE DISPOSAL FACILITY

The SDF is located within a 650,000 m^2 area of the SRS known as Z Area. The Z Area lies on a local topographic high, approximately 91 m above sea level. The SPF and SDF at Z Area are part of an integrated waste treatment and disposal system at SRS. The SPF and SDF are regulated by the State of South Carolina, USDOE Orders, and other Federal regulations that are applicable to disposal of solid waste.

Saltstone is a solid waste form that is the product of chemical reactions between a salt solution and a blend of cementitious materials (slag, flyash, and cement). A slurry of the components is pumped into vaults located in the SDF, where the saltstone grout solidifies into a monolithic, nonhazardous solid low-level wasteform. Based on the projected vault and site layout of the SDF in the original RPA (MMES et al., 1992), up to 15 vaults will be constructed for saltstone disposal. This capacity of the SDF will enable up to 730-million L (192 million gal) of wastewater to be treated for subsequent disposal as saltstone. Approximately 25 years at the design basis production rate for the SPF would be needed to reach this disposal capacity.

Once the capacity of this facility is reached, or the wastewater supply has been exhausted, the SDF will be closed. The present closure concept includes two moisture barriers consisting of clay/gravel drainage systems, along with backfill layers and a shallow-rooted bamboo vegetative cover.

2.3 PERFORMANCE CRITERIA

The specific performance criteria for solid waste disposal in Z Area are contained in USDOE Order 435.1 (USDOE, 1999a):

2.3.1 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 of after September 26, 1988:

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

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

- 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 of near-surface, an assessment of impacts to water resources and to hypothetical persons assumed to inadvertently intrude into the low-level waste disposal facility. Table 2-1 lays out the performance measures and the associated points of compliance.

USDOE Order 435.1 states that "The performance assessment shall include calculations for a 1,000-y 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 criteria. The performance criteria, including the inadvertent intruder and groundwater analysis requirements, are applied for 10,000 years after disposal. The longer time frame was selected to be consistent with the SRS DAS (Fiori and Frei, 1999).

2.3.2 Intruder Analysis

USDOE Order 435.1 provides a performance measure pertinent to impacts to hypothetical persons who are assumed to inadvertently intrude into the Z-Area SDF which specifies 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 y 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 reasonable and credible pathways. However, consumption of groundwater and crop irrigation are exposure pathways that are excluded from the intruder analysis (USDOE, 1996); impacts of groundwater contamination are evaluated separately in the original SDF RPA (MMES et al., 1992) and in this study.

2.3.3 Groundwater Analysis

USDOE Order 435.1 requires an analysis of groundwater concentrations of radionuclides leached from the waste disposal facility in order to address both the all-pathways performance objective and the water resources impact assessment requirement (Table 2-1). 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, including potential ingestion of groundwater. The point of compliance is the point of highest calculated dose beyond a 100-meter buffer zone surrounding the waste.

Table 2-1. Performance objectives, assessment requirements, and points of compliance

Component	Performance Objective	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

For the water resources 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, at SRS 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 contamination of groundwater at SRS is regulated under 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).

The Primary Drinking Water Standards for radionuclides, promulgated on December 7, 2000, are used in this Special Analysis (USEPA, 2000). The current 4 mrem/y standard for beta and/or photon emitters in drinking water requires that MCLs be developed based on internal dosimetry data from National Bureau of Standards (NBS) Handbook 69 (U.S. Department of Commerce, 1963) and specified MCLs for ^{3}H and ^{90}Sr. A listing of the resulting MCLs is available in the Implementation Guidance for Radionuclides (USEPA, 2001). There are several radionuclides in the present analysis for which MCLs are not available in this listing. For the radionuclides important to the groundwater analysis in this study (^{79}Se and ^{126}Sn), an MCL is derived assuming a limit of 4 mrem/y EDE and internal dosimetry based on ICRP Publication 30 (1979). This method is consistent with that used in the approved PA for E-Area (McDowell-Boyer et al., 2000).

2.3.4 Air Analysis

The all-pathways performance objective of USDOE Order 435.1 includes all modes of exposure, including the air pathway, but excluding exposures to radon and short-lived progeny. In addition to this objective, calculated dose via the air pathway is not to exceed 10 mrem/y total EDE, again excluding dose from radon and short-lived progeny (Table 2-1). Again, the point of compliance is the point of highest calculated dose beyond a 100-meter buffer zone surrounding the waste.

2.3.5 Radon Emanation Analysis

Radon is addressed separately in a performance objective under USDOE Order 435.1, 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 SDF 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-meter buffer zone surrounding the disposed waste) should be applied (USDOE, 1996).

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3. DISPOSAL FACILITY CHARACTERISTICS

As noted in the previous section, the SDF is located within the SRS in an area designated as Z Area. Before discussing characteristics particular to the Z-Area site and SDF facility (Sect. 3.1 through 3.3), regional characteristics of the SRS are briefly reviewed here. A more in-depth treatment of the regional geography, demography, meteorology, seismicity, hydrogeology, surface water hydrology, soils, and ecology is provided in Sect. 2.1 of the original PA (MMES et al., 1992).

The SRS occupies about 780 km^2 in Aiken, Barnwell, and Allendale counties on the Upper Atlantic Coastal Plain of southwestern South Carolina (Fig. 3-1). The elevation of the SRS ranges from 24 m above sea level at the Savannah River to about 122 m above sea level in the upper northwest portion of the site. The Pleistocene Coastal terraces and the Aiken Plateau form two distinct physiographic subregions at SRS (WSRC, 1992). The Pleistocene Coastal terraces are below 82 m in elevation with the lowest terrace constituting the present flood plain of the Savannah River and the higher terraces characterized by gently rolling topography. The relatively flat Aiken Plateau occurs above 82 m.

The Aiken Plateau is dissected by numerous streams. Because of the large number of tributaries to small streams on the SRS site, no location on the site is far from a flowing stream, most of which drain to the Savannah River. The Savannah River bounds the SRS for 28 km on the southwest.

The dominant vegetation on the SRS is forest with types ranging from scrub oak communities on the driest areas to bald cypress and black gum in the swamps. Pine forests cover more area than any other forest type. Land utilization presently is about 56% in pine forests, 35% in hardwoods, 7% in SRS facilities and open fields, and 2% in water (WSRC, 1992).

Most of the soils at the SRS are sandy over a loamy or clayey subsoil. The distribution of soil types is very much influenced by the creeks on the site with colluvial deposits on hilltops and hillsides giving way to alluvium in valley bottoms (Dennehy et al., 1989). Weathering effects are evident. Average soil erosion rates for the area surrounding the SRS, much of which is cropland, range from 1.5 to 2.0 kg m^{-2} y^{-1} (U.S. Department of Agriculture, 1985). Employing the Universal Soil Loss Equation to predict erosion at the SRS under different vegetative conditions, Horton and Wilhite (1978) estimate that the presence of natural successional forests would reduce erosion by a factor of 400 to 500 over cropland erosion.

Except for three roadways and a railway that are near the edge of SRS, public access to SRS is restricted to guided tours, controlled deer hunts, and authorized environmental studies. Fig. 3-2 shows the major areas at SRS and their location within the site boundary. The major production areas located at the site include: Raw Materials (M Area), Separations (F and H Areas), Waste Management Operations (E, F, and H Areas), and Defense Waste Processing (S and Z Areas) (WSRC, 1992). Administrative and support services, the Savannah River Technology Center, and the Savannah River Ecology Laboratory are located in A Area. Additional administrative and support services are located in B and C Areas.

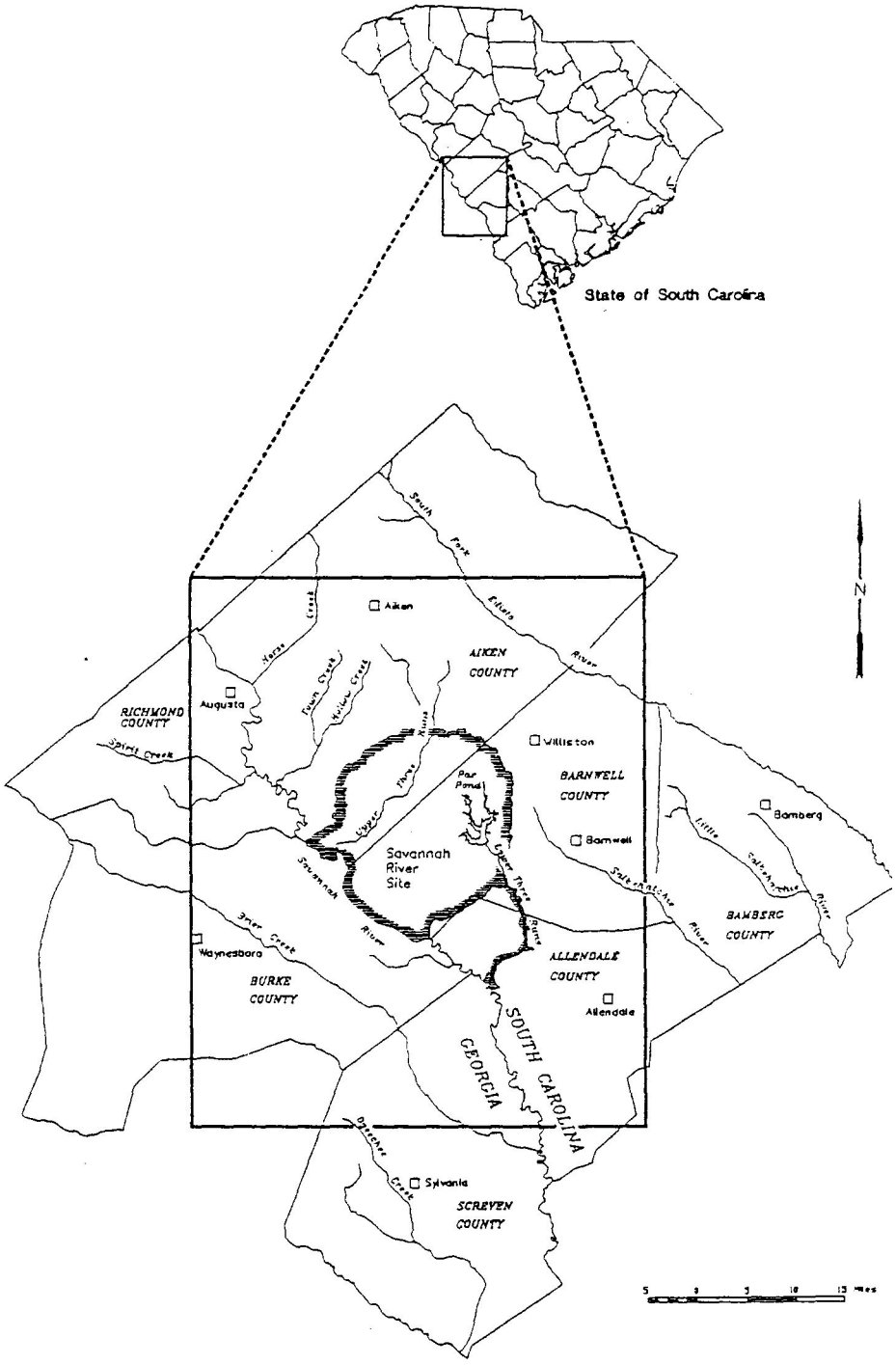

Fig. 3-1. SRS Regional Location Map

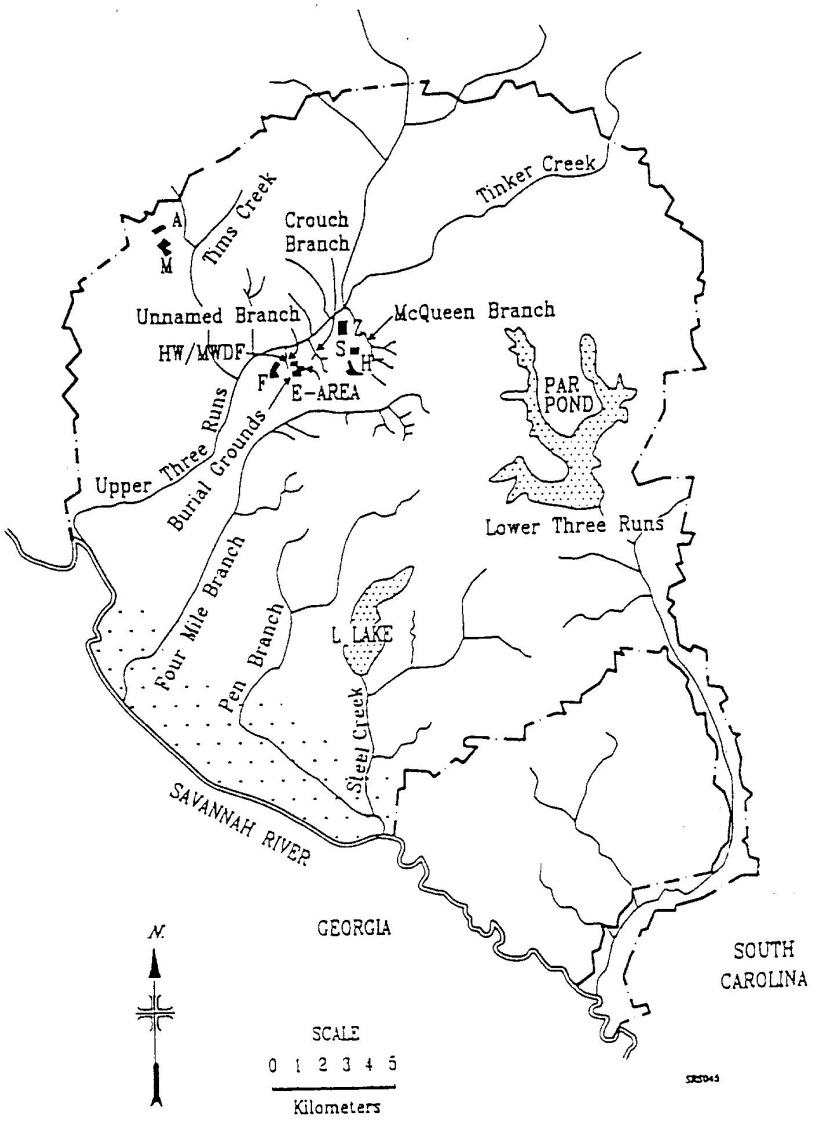

Fig. 3-2. Facility Location Map of SRS Showing Surface Drainage

3.1 SITE CHARACTERISTICS

Z Area was chosen for the SDF site based on considerations of depth to the water table, distance to surface water and the public, available surface area, surface topography, and its proximity to the wastewater generation site. Z Area at SRS, where the SDF is located, consists of approximately 650,000 m^2 and is situated about 2 km northeast of the SRS DWPF at S Area (Fig. 3-2).

The Aiken Plateau is dissected by numerous streams near Z Area that greatly influence the local groundwater system (Dennehy et al., 1989). The Z Area lies on a local topographic high, at approximately 91 m above sea level. Z Area is bounded by McQueen Branch in the northeast and Upper Three Runs in the northwest. The local relief is about 50 m. McQueen Branch is a tributary of Upper Three Runs. Upper Three Runs drains into the Savannah River, some 15 km southwest of Z Area. Upper Three Runs lies about 1.2 km from the northwest corner of Z Area. The northeast corner of Z Area is located only about 150 m from McQueen Branch. McQueen Branch and Crouch Branch are incised into the topographic high, southeast and southwest of Z Area, such that their headwaters come within about 1 km of each other at approximately 1.4 km south of Z Area (Dennehy et al., 1989). The elevations of both tributaries range from about 46 m to 76 m. Presently, open fields characterize Z Area.

Except in the vicinity of the creeks, the water table occurs in what is called the "Upland Unit" of the southwestern South Carolina Coastal Plain. The depth to the water table in a normal precipitation year, in the Z-Area vicinity, ranges from 8 to 18 m (Dennehy et al., 1989). Under Z Area only, the minimum depth to the water table from the ground surface in any given year is estimated to be 13 m on the basis of water table fluctuations from several years' data (Cook, 1983). This minimum depth corresponds to a year in which the highest recorded precipitation of 188 cm occurred near SRS, and thus, corresponds to the historic high water table. The direction of flow is affected by the creeks and is generally in a northern direction at Z Area (Dennehy et al., 1989). The horizontal gradient ranges from 0.002 in the southern part of Z Area to 0.05 at the northeastern hill slope. An in-depth discussion of the hydrogeology of Z Area is provided in Sect. 2.2 of the original PA (MMES et al., 1992).

The watershed of Upper Three Runs drains about 500 km^2 of the Upper Coastal Plain northeast of the Savannah River. Significant tributaries to this creek are Tinker Creek, which is a headwaters branch that comes in north of Z Area, and Tims Branch, which connects up south of Z Area (Fig. 3-2). There are no lakes or flow control structures on Upper Three Runs or its tributaries. The stream channel has a low gradient and is meandering. Its floodplain ranges in width from 0.4 to 1.6 km and is heavily forested with hardwoods.

Two smaller tributaries of Upper Three Runs, McQueen Branch and Crouch Branch are located north and south, respectively, of Z Area. Both tributaries receive runoff from Z Area. McQueen Branch has a drainage area of about 11 km^2 and Crouch Branch has a drainage area of about 2.8 km^2.

Currently, groundwater in the upper four stratigraphic units is not pumped from Z Area (MMES et al., 1992, Sect. 2.2.4). Water from the creeks local to Z Area is not currently used for human consumption.

3.2 PRINCIPAL FACILITY DESIGN FEATURES

The SDF is permitted as a landfill for the disposal of solid industrial waste by the state of South Carolina. As presently planned, the facility will contain several large concrete vaults divided into cells. Each of the cells will be filled with solid saltstone. The saltstone itself provides primary containment of the waste, and the walls, floor, and roof of the vaults provide secondary containment.

Approximately 3 to 4.5 m of overburden have been removed to prepare and level the site for vault construction. All vaults will be built at or slightly below the grade level that exists after the overburden and leveling operations are complete. The bottom of the saltstone monoliths will be at least 8 m above the historic high water table beneath the Z-Area site, thus, avoiding disposal of waste in a zone of water table fluctuation. Run-on and runoff controls are installed to minimize site erosion during the operational period.

In the proposed disposal site layout, up to 15 concrete vaults will be constructed for saltstone disposal (Fig. 3-3). Fourteen of these vaults will each have dimensions of approximately 60-m wide by 180-m long by 7.6-m high. The other vault (Vault 1) is approximately 30-m wide by 180-m long by 7.6-m high. Based on current vault designs, each of the 14 larger vaults will be divided into 12 cells that are approximately 30-m wide by 30-m long by 7.6-m high. Vault 1 is divided into six cells with the same cell dimensions as the larger vaults. Operationally, the cells of these vaults will be filled to a height of about 7.3 m with saltstone, and then a layer of uncontaminated grout approximately 0.4-m thick will be poured to fill the space between the saltstone and the vault roof. The permanent roof is currently designed with a specified minimum thickness of 0.75 m and a minimum slope of 2 cm/m. Additional details of the vault designs are provided in Sect. 2.5 of the original PA (MMES et al., 1992).

In terms of capacity, the disposal site is best described in terms of the number of vault cells used to receive waste. The proposed layout will thus contain 174 vault cells distributed over the 15 vaults that can receive saltstone grout. Each cell is sized to handle the volume of saltstone that would be produced from the treatment of approximately 4.2-million L (1.1-million gal) of wastewater. Active disposal operations in Z Area are projected to continue for about 25 y before the permitted disposal capacity is reached.

Except for erosion control purposes, backfilling around the vaults will not be done prior to filling the vaults with saltstone. Final back-filling to cover vaults will be deferred until several or all of the vaults have been built and filled. This approach of delaying backfilling until near the end of the operational period allows the vaults to be visually monitored for several years before closure operations begin. This approach also would enable the use of improved closure technology that may be developed during the operational period at the SDF.

Closure operations will begin near the end of the active disposal period in the SDF, i.e., after most or all of the vaults have been constructed and filled (Cook et al., 2000). Backfill of native soil will be placed around the vaults. The vaults will be covered with a clay/gravel drainage system comprised of 0.5 m of clay with an overlying 0.15-m layer of gravel. The clay/gravel drainage system is intended to prevent the buildup of perched water above the vaults. Above the clay/gravel drainage system, a geotextile fabric to maintain layer separation from overlying

backfill and a minimum of 0.3 m of backfill will be placed. Above this layer of backfill, a laterally extensive moisture barrier will be installed. This upper moisture barrier will consist of 0.76 m of clay and an overlying layer of 0.3 m of gravel. A geotextile fabric will also be placed on this upper gravel layer, and a second backfill layer, approximately 0.76-m thick, will be placed over the moisture barrier. Finally, a 0.15 m layer of topsoil will be placed on the top layer of backfill to complete closure of the SDF. This sequence of layers will provide a minimum of 2.92 m of cover for each vault.

Final closure of the SDF will be accomplished by constructing a drainage system and revegetating the site. The drainage system will consist of a system of rip-rap lined ditches that intercept the gravel layer of the moisture barrier. These ditches will divert surface runoff and water intercepted by the moisture barrier away from the disposal site. The drainage ditches will be constructed between rows of vaults and around the perimeter of the SDF.

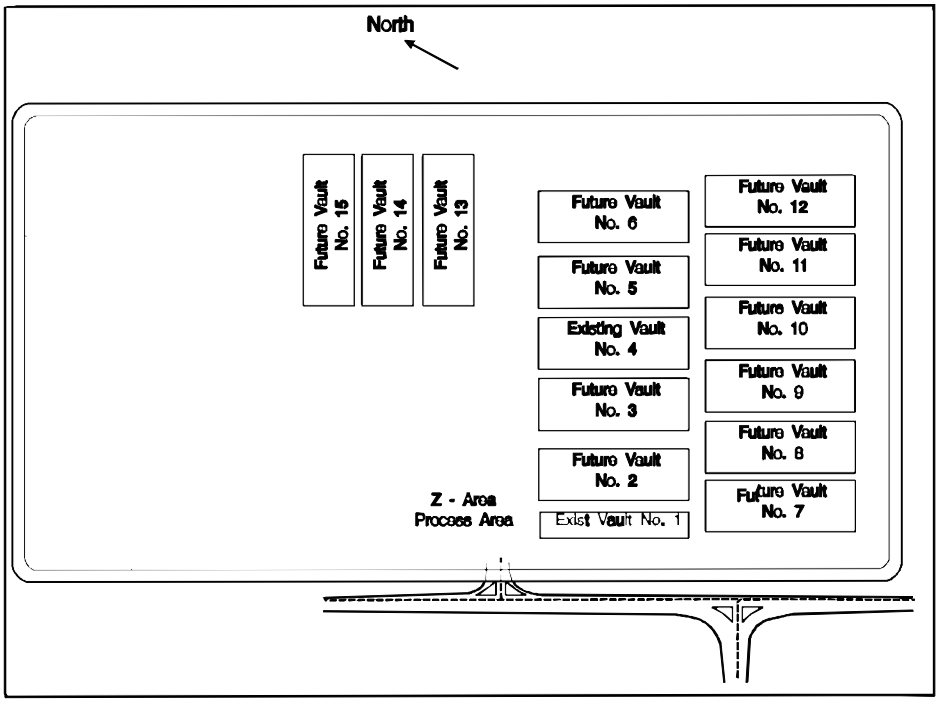

Fig. 3-3. Projected Layout of Z-Area Saltstone Vaults

The topsoil will be revegetated with bamboo. A study conducted by the USDA Soil Conservation Service (Cook and Salvo, 1992) has shown that two species of bamboo (Phyllostachys bissetii and Phyllostachys rubromarginata) will quickly establish a dense

ground cover which will prevent the growth of pine trees, the most deeply rooted naturally occurring plant type at SRS. Bamboo is a shallow-rooted climax species which evapotranspirates year-round in the SRS climate removing a large amount of moisture from the soil and decreasing the infiltration into the underlying disposal system.

3.3 WASTE CHARACTERISTICS

As presently planned, contaminated wastewater from two sources will be sent to Z Area for treatment and disposal. The wastewater sent to Z Area contains principally soluble solids and very low levels of most radioactive contaminants. Soluble incidental waste from the HLW tanks at the SRS is a major source of wastewater sent to Z Area. A second wastewater stream also containing principally soluble solids and very low levels of radioactive contaminants is generated in the F/H Area ETF where condensate from evaporators in the Separations Facilities and the HLW Tank Farm is sent for treatment. Miscellaneous wastewater streams containing low levels of radioactive contaminants from other sources on the site are also treated in the ETF.

As noted earlier, saltstone is produced from a mixture of salt solution and a dry blend of cementitious materials (slag, fly ash, and cement), and an acceptable waste form can be produced over a range of these individual components. Solid saltstone is a complex mixture of insoluble solids, soluble solids, and water. As the saltstone grout is prepared and cured, several chemical reactions occur between the components of the dry blend and contaminants in the salt solution. Several wastewater contaminants are converted to insoluble species or incorporated into the cement matrix, effectively retarding their release from the saltstone waste form.

Development of this waste form and its physical and chemical properties are described in Sect. 2.4.1 of the original PA (MMES et al., 1992). Briefly, between 1979 and 1987, a formulation for saltstone was developed that rendered the final wasteform product that is resistant to leaching of contaminants present in the porous matrix and is classified as nonhazardous solid waste as defined by USEPA protocol (USEPA, 2002).

The average projected composition of the saltstone that will be sent to the SDF for disposal is 47 wt% salt solution, 25 wt% slag, 25 wt% fly ash, and 3 wt% cement. When first prepared, the saltstone grout is readily pumped from the SPF to a cell in a disposal vault. After setting, the saltstone is self-supporting with a 28-day compressive strength in excess of 1.45×10^6 Pa. The specific gravity of the solidified saltstone ranges from 1.6 to 1.8, and bulk density is estimated at 1.7×10^3 kg/m^3.

The initial incidental wastewater that will be sent to Z Area is called Low Curie Salt (LCS). This wastewater is produced from selected HLW salt tanks that are expected to be low in ^{137}Cs. The supernate in these tanks, which contains the bulk of the cesium, will be drained and pumped to another tank. The resulting salt cake will be dissolved and transferred to HLW tank 50, from which it will be sent to Z Area. The currently estimated radionuclide composition of LCS is presented in Table 3-1. Radionuclide limits derived in this study are compared with this radionuclide composition in Sect. 8.

Table 3-1. Estimated average radionuclide concentrations in low curie salt solution feed to saltstone[a]

Nuclide	Low Curie Salt Solution, Ci/L	Nuclide	Low Curie Salt Solution, Ci/L
H-3	0.00E+00	Sm-151	0.00E+00
C-14	4.46E-10	Eu-154	1.00E-04
Co-60	4.08E-05	Eu-155	0.00E+00
Ni-59	2.57E-07	Th-232	2.88E-10
Ni-63	3.98E-10	U-232	6.23E-11
Se-79	1.51E-07	U-233	1.72E-08
Sr-90	8.94E-03	U-234	6.34E-09
Y-90	8.94E-03	U-235	2.00E-10
Tc-99	2.57E-06	U-236	9.63E-10
Ru-106	9.50E-07	U-238	4.78E-09
Rh-106	9.50E-07	Np-237	8.92E-09
Sn-126[b]	7.50E-07	Pu-238	2.18E-04
Sb-125	2.43E-05	Pu-239	3.32E-06
Sb-126	2.02E-07	Pu-240	1.55E-06
Te-125m	0.00E+00	Pu-241	1.06E-04
I-129	2.37E-11	Pu-242	3.60E-09
Cs-134	1.06E-06	Am-241	2.22E-05
Cs-135	1.81E-09	Am-242m	2.11E-08
Cs-137	2.26E-02	Cm-242	2.20E-05
Ba-137m	2.14E-02	Cm-243	0.00E+00
Ce-144	4.90E-07	Cm-244	2.20E-05
Pr-144	4.90E-07	Cm-245	1.64E-09
Pm-147	5.13E-04		

[a] Values from Drumm (2002), Appendix D, Average Feed, 30% Interstitial, 300 mg/L sludge.

[b] Value from Reboul (2002).

4. ANALYSIS OF INADVERTENT INTRUSION

This section presents an assessment of potential radiation doses to a hypothetical inadvertent intruder onto the site of the Saltstone Disposal Facility (SDF) at the Savannah River Site (SRS). Results of the dose assessment are used to derive a set of limits on allowable average concentrations and total inventories of radionuclides in waste at the time of disposal.

Doses to a hypothetical inadvertent intruder are estimated based on assumptions about credible exposure scenarios at different times after disposal and their associated exposure pathways. The scenarios for inadvertent intrusion at different times are based on an assumed design and performance of the cover system above a disposal vault. Results of the dose assessment for the assumed scenarios are expressed in terms of annual effective dose equivalents (EDE) per unit concentration of radionuclides in a disposal vault; these doses per unit concentration are referred to as scenario dose conversion factors (SDCFs). Limits on allowable concentrations and inventories of radionuclides at the time of disposal then are calculated based on the SDCFs for each radionuclide of concern, a specified performance measure for exposure of inadvertent intruders, assumptions about the time of occurrence of the assumed scenarios, and assumptions about the degradation of the cover system above a vault over time.

The specified performance measures for inadvertent intruders (USDOE, 1999a) include (1) an annual effective dose equivalent of 100 mrem (1 mSv) for scenarios involving chronic exposure and (2) an effective dose equivalent of 500 mrem (5 mSv) for scenarios involving a single acute exposure (see Sect. 2.3.2). In both performance measures for inadvertent intruders, potential doses due to inhalation of radon and its short-lived decay products are excluded (USDOE, 1999a). The relevant scenarios for inadvertent intrusion involve exposure to residual solidified waste in a disposal facility, and scenarios that involve exposure to contaminated groundwater or surface water on the disposal site are excluded (USDOE, 1996). The scenarios for inadvertent intrusion assumed in this analysis involve chronic exposure.

For the purpose of establishing limits on allowable disposals of radionuclides in a near-surface facility, a time frame for assessments of inadvertent intrusion of 1,000 years after facility closure is specified (USDOE, 1999a), and the assessments also should assume that active institutional control will be maintained over a disposal site for at least 100 years (USDOE, 1999a). In this analysis, limits on allowable disposals of radionuclides in the SDF are calculated based on a longer time frame of 10,000 years for assessments of inadvertent intrusion, to be consistent with the SRS DAS (Fiori and Frei, 1999), as well as the time frame of 1,000 years specified by USDOE (1999a).

The following section identifies the radionuclides that are included in the dose analysis for inadvertent intruders. Sect. 4.2 describes the scenarios for inadvertent intrusion at the SDF that are assumed in the present analysis. The scenarios assumed in this analysis, as well as the design of the cover system above a disposal vault, differ from the scenarios assumed in the previous analysis (MMES et al., 1992). The scenarios assumed in the previous analysis and the results of the previous analysis are summarized in Appendix A. The rationale for the changes in the assumed scenarios and design of the cover system is also discussed in

Appendix A. Sect. 4.3 presents the dose analysis for the assumed intrusion scenarios and the calculated SDCFs for each radionuclide and scenario. Finally, Sect. 4.4 presents the calculated limits on allowable concentrations and inventories of radionuclides for disposal in the SDF based on the results of the dose assessment for inadvertent intruders, the assumed times of occurrence of the exposure scenarios, and the assumed conditions of exposure at those times.

4.1 RADIONUCLIDES CONSIDERED IN DOSE ANALYSIS

Low-level radioactive waste that may be sent to the SDF contains many radionuclides. However, the number of radionuclides that need to be included in a dose analysis for inadvertent intruders can be reduced substantially based on considerations of radionuclide half-lives and the processes by which low-level waste at the SDF is generated.

Since institutional control will be maintained for at least 100 years after closure of the SDF (USDOE, 1999a), radionuclides with a half-life less than about 5 years can be excluded from the analysis, unless the radionuclide has a decay product with a half-life greater than about 5 years, because these shorter-lived radionuclides would decay to innocuous levels during the institutional control period regardless of their inventories in waste at the time of disposal. Selection of longer-lived radionuclides for inclusion in the dose analysis for inadvertent intruders was based on the following considerations.

In a recent report, the National Council on Radiation Protection and Measurements (NCRP) developed screening levels for radionuclides in contaminated surface soils based on the results of dose assessments for assumed exposure scenarios and an assumed dose of concern (NCRP, 1999). More than 200 radionuclides with a half-life greater than 30 days were considered, without regard for how they are produced or whether they could be important in contaminated soils. When radionuclides with a half-life less than about 5 years that do not have decay products with a half-life greater than about 5 years are eliminated, based on the assumed period of institutional control at the SDF, 99 radionuclides remain. Of these, the 60 radionuclides listed in Table 4-1 were selected for inclusion in the dose analysis for inadvertent intruders. This list includes all potentially important fission and activation products and all actinide and transuranic radionuclides that could occur in significant amounts in operations of nuclear reactors. The inclusion of Cm-242, Bk-249, and Cf-252, which have a half-life substantially less than 5 years, is based on their decay to longer-lived Pu-238, Cf-249, and Cm-248, respectively.

Table 4-1. Radionuclides considered in dose analysis for inadvertent intruders

Radionuclide[a]	Half-life[b]	Radionuclide[a]	Half-life[b]
H-3	12.33 y	I-129	1.57×10^7 y
Be-10	1.51×10^6 y	Cs-135	2.3×10^6 y
C-14	5.73×10^3 y	Cs-137	30.07 y
Al-26	7.17×10^5 y	Ba-137m (0.946)	2.552 m
Co-60	5.27 y	Sm-151	90 y
Ni-59	7.6×10^4 y	Eu-152	13.516 y
Ni-63	100.1 y	Eu-154	8.592 y
Se-79	1.1×10^6 y	Eu-155	4.761 y
Sr-90	28.79 y	Pb-210	22.3 y
Y-90 (1.0)	64.0 h	Po-210	138.376 d
Zr-93	1.53×10^6 y	Ra-226	1.6×10^3 y
Nb-93m (1.0)[c]		Rn-222 (1.0)	3.8235 d
Nb-93m	16.13 y	Pb-214 (1.0)	26.8 m
Nb-94	2.03×10^4 y	Bi-214 (1.0)	19.9 m
Tc-99	2.11×10^5 y	Pb-210 (1.0)[c]	
Pd-107	6.5×10^6 y		
Cd-113m	14.1 y		
Sn-121m	55 y		
Sn-126	1.0×10^5 y		
Sb-126m (1.0)	19.15 m		
Sb-126 (0.14)	12.46 d		

Table is continued on following page.

Table 4-1. (continued)

Radionuclide[a]	Half-life[b]	Radionuclide[a]	Half-life[b]
Ra-228	5.75 y	Th-232	1.405×10^{10} y
Ac-228 (1.0)	6.15 h	Ra-228 (1.0)[c]	
Th-228 (1.0)	1.9116 y	Pa-231	3.276×10^{4} y
Ra-224 (1.0)	3.66 d	Ac-227 (1.0)[c]	
Rn-220 (1.0)	55.6 s	U-232[d]	68.9 y
Pb-212 (1.0)	10.643 h	U-233	1.592×10^{5} y
Bi-212 (1.0)	60.55 m	U-234	2.455×10^{5} y
Tl-208 (0.3594)	3.053 m	U-235	7.038×10^{8} y
Ac-227	21.773 y	Th-231 (1.0)	25.52 h
Th-227 (0.9862)	18.72 d	U-236	2.342×10^{7} y
Ra-223 (1.0)	11.435 d	U-238	4.468×10^{9} y
Pb-211 (1.0)	36.1 m	Th-234 (1.0)	24.10 d
Bi-211 (1.0)	2.14 m	Pa-234m (1.0)	1.17 m
Tl-207 (0.9972)	4.77 m	Pa-234 (0.0016)	6.70 h
Th-229	7.34×10^{3} y	Np-237	2.144×10^{6} y
Ra-225 (1.0)	14.9 d	Pa-233 (1.0)	26.967 d
Ac-225 (1.0)	10.0 d	Pu-238	87.7 y
Fr-221 (1.0)	4.9 m	Pu-239	2.411×10^{4} y
Bi-213 (1.0)	45.59 m	Pu-240	6.564×10^{3} y
Tl-209 (0.0209)	2.161 m	Pu-241	14.29 y
Th-230	7.54×10^{4} y	Pu-242	3.773×10^{5} y

Table is continued on following page.

Table 4-1. (continued)

Radionuclide[a]	Half-life[b]	Radionuclide[a]	Half-life[b]
Pu-244	8.00×10^7 y	Cm-244	18.11 y
Np-240m (0.9988)	7.22 m	Cm-245	8.5×10^3 y
Am-241	432.2 y	Cm-246	4.76×10^3 y
Am-242m	141 y	Cm-247	1.56×10^7 y
Am-242 (0.9954)	16.02 h	Pu-243 (1.0)	4.956 h
Cm-242 (0.823)	162.8 d	Cm-248	3.48×10^5 y
Np-238 (0.0046)	2.117 d	Bk-249[e]	330 d
Pu-238 (0.828)[c]		Cf-249	351 y
Am-243	7.37×10^3 y	Cf-250	13.08 y
Np-239 (1.0)	2.3565 d	Cf-251	900 y
Cm-242[e]	162.8 d	Cf-252[e]	2.645 y
Cm-243	28.5 y		

[a] Indented entries are radiologically significant shorter-lived decay products of parent radionuclide listed. For each decay product, branching fraction in decay of parent radionuclide (Tuli, 2000) is given in parentheses.

[b] Values from Tuli (2000). Units are y = years, d = days, h = hours, m = minutes, s = seconds.

[c] Decay product is listed separately when it is sufficiently long-lived that its occurrence in disposed waste could result from processes other than decay of its longer-lived parent.

[d] Shorter-lived decay products Th-228, Ra-224, Rn-220, Pb-212, Bi-212, and Tl-208 are listed following entry for Ra-228.

[e] Radionuclide is included only because it has longer-lived decay products (see Table 4-2).

Many radionuclides listed in Table 4-1 have shorter-lived decay products that also are listed in the table. All such decay products are taken into account in the dose analysis for inadvertent intruders based on an assumption of activity equilibrium with the parent radionuclide. All radionuclides listed in Table 4-1 beginning with Pb-210 also are members of a long decay chain of alpha-emitting actinide and transuranic radionuclides. The radionuclides in these decay chains are listed in Table 4-2. Buildup of radioactive decay products in disposed waste over time, including decay products that are longer-lived than their parent radionuclide (e.g., Am-241 produced in decay of Pu-241) as well as decay products that are shorter-lived than their parent (e.g., Ra-226 produced in decay of Th-230), is taken into account in the dose analysis for inadvertent intruders. The importance of a decay product depends on its half-life, the radiological properties of the parent and decay product, and the time frame for the analysis. The half-life of the parent also is important when the decay product is longer-lived.

The remaining 39 radionuclides with a half-life greater than 5 years considered by the NCRP (1999) were excluded from the dose analysis for inadvertent intruders based on the following considerations. First, many of these radionuclides are not fission products and, thus, would not be present in wastes generated at the SRS, or they are not important activation products and, thus, could not be present in more than trace amounts. These radionuclides include the following:

> Si-32, Cl-36, K-40, Ca-41, Ti-44, Mn-53, Fe-60, Mo-93, Tc-97, Tc-98, Ag-108m, Te-123, Ba-133, La-137, La-138, Pm-145, Sm-146, Eu-150, Gd-148, Gd-152, Tb-157, Tb-158, Ho-166m, Lu-176, Hf-178m, Hf-182, Ta-180m, Re-187, Os-194, and Pt-193.

Some of these radionuclides also can be excluded based on their very long half-life (i.e., very low activity per unit mass). The activity of the longest-lived radionuclides in waste would always be orders of magnitude less than the activity of such potentially important long-lived fission products as Tc-99, Sn-126, and I-129. These radionuclides include Te-123 ($>6 \times 10^{14}$ y), La-138 (1.05×10^{11} y), Gd-152 (1.08×10^{14} y), Ta-180m ($>1.2 \times 10^{15}$ y), and Re-187 (4.35×10^{10} y). Long-lived K-40 (1.277×10^{9} y) could occur in low-level waste at the SRS, but only as a consequence of its occurrence in natural materials. Incidental levels of naturally occurring radionuclides that are not enhanced by activities at the SRS are a part of natural background and are not considered to be subject to requirements on disposal of radioactive waste.

Second, a few fission products, including Rb-87, Cd-113, In-115, and Sm-147, can be excluded on the basis of their long half-life, which ranges from about 5×10^{10} y to nearly 10^{16} y. The activities of these radionuclides in waste would always be several orders of magnitude less than the activities of other important fission products with shorter half-lives, and previous assessments have indicated that disposal limits for these radionuclides based on analyses of scenarios for inadvertent intrusion should exceed their specific activities (ORNL, 1997).

Table 4-2. Principal members of decay chains of actinide and transuranic radionuclides[a]

Neptunium series	Uranium series	Actinium series	Thorium series
Bk-249	Cf-250	Cf-251	Cf-252[i]
Cf-249	Cm-246[b]	Cm-247[g]	Cm-248[j]
Cm-245	Cm-242[c]	Cm-243[h]	Cm-244[k]
Pu-241	Am-242m[d]	Am-243	Pu-244
Am-241	Pu-242[e]	Pu-239	Pu-240
Np-237	Pu-238[f]	U-235	U-236
U-233	U-238	Pa-231	U-232[m]
Th-229	U-234	Ac-227	Th-232
	Th-230		Ra-228
	Ra-226		
	Pb-210		

[a] Only radionuclides listed in Table 4-1 are included. Except as noted, entry immediately below a given radionuclide is its decay product.

[b] Decay product is Pu-242.

[c] Decay product is Pu-238; radionuclide is produced in decay of Am-242m.

[d] Decay products are Pu-242 and Pu-238 with branching fractions of 0.172 and 0.828, respectively (Tuli, 2000); radionuclide is not produced by decay of any other member of uranium series listed.

[e] Decay product is U-238.

[f] Decay product is U-234.

[g] Decay product is Am-243.

[h] Decay products are Am-243 and Pu-239 with branching fractions of 0.0029 and 0.9971, respectively (Tuli, 2000); radionuclide is not produced by decay of any other member of actinium series listed.

[i] Branching fraction in decay to Cm-248 is 0.9691 (Tuli, 2000); remainder of decays are by spontaneous fission.

[j] Radionuclide decays to Pu-244.

[k] Radionuclide decays to Pu-240 and is not produced by decay of any other member of thorium series listed.

[m] Radionuclide decays to Th-228, which is shorter-lived decay product of Ra-228 (see Table 4-1), and is not produced by decay of any other member of thorium series listed.

Third, a few radionuclides can be excluded because they are not fission products or important activation products and, furthermore, do not occur in the long decay chains of actinide and transuranic radionuclides. These radionuclides include Pb-202, Pb-205, Bi-207, and Bi-210m.

Finally, Np-236 (1.54×10^5 y) can be excluded because it is produced in much smaller amounts in nuclear reactors than Np-237.

None of the radionuclides excluded from the present analysis have been reported to occur in significant amounts in low-level waste (either commercial or USDOE). This information provides support for neglecting these radionuclides in the analysis. If the excluded radionuclides occur in waste generated at the SRS, their activities would be inconsequential compared with the activities of other radionuclides that are considered in the analysis.

4.2 SCENARIOS FOR EXPOSURE OF INADVERTENT INTRUDERS

This section discusses the exposure scenarios and associated exposure pathways that are assumed in the dose analysis for inadvertent intruders at the SDF. The discussion is divided into two parts. The present design of the cover system on each disposal vault, updated from that in the existing PA (MMES et al., 1992), is described in Sect. 4.2.1. Sect. 4.2.2 discusses the assumed exposure scenarios for inadvertent intruders based on the new design of the cover system. The exposure scenarios that were assumed in the existing PA for the SDF (MMES et al., 1992) and the results of the analysis are summarized in Appendix A. Appendix A also presents a reevaluation of the results of the previous dose analysis for inadvertent intruders taking into account, first, changes in estimates of the inventories of important radionuclides in waste intended for disposal in the SDF and, second, certain assumptions used in the previous analysis that are not justified on technical grounds.

4.2.1 Change in Design of Cover System for Disposal Vault

Based on the reevaluation of the previous dose analysis for inadvertent intruders described in Appendix A, the design of the cover system above a disposal vault documented previously (Cook et al., 2000) has been modified to include an additional layer of grout above the reinforced concrete roof on a vault. The design thickness of the additional grout layer is 1 m. No other changes in the documented design of the cover system have been made. With this addition, the design of the cover system, including all layers between the ground surface and the buried waste (saltstone), is as summarized in Table 4-3.

Table 4-3. Design thicknesses of different layers of material in cover system on a disposal vault assumed in present analysis[a]

Material	Thickness (m)	Comment
Ground surface		Surface will be revegetated to enhance evapotranspiration and reduce erosion
Cover above engineered barriers		Total thickness of cover is 2.9 m
Layer of topsoil	0.15	
Layer of backfill	0.76	
Layer of gravel	0.3	
Layer of clay	0.76	
Layer of backfill	0.3	
Layer of gravel	0.15	
Layer of clay	0.5	
Engineered barriers above waste		Total thickness of barriers to deter excavation to depth of saltstone is 1.5 m
Layer of grout	1	
Reinforced concrete roof on disposal vault	0.1	
Layer of grout above saltstone in disposal vault	0.4	
Saltstone		

[a] Specifications for all components of cover system except layer of grout immediately above reinforced concrete roof are given by Cook et al. (2000). Geotextile membranes above two gravel layers are not included.

4.2.2 Selection of Credible Exposure Scenarios for Inadvertent Intruders

4.2.2.1 Credibility of Agriculture Scenario. As described in Appendix A and documented in the existing PA (MMES et al., 1992), the exposure scenario for inadvertent intruders referred to as the agriculture scenario resulted in the highest estimates of dose when the design of the cover system above a disposal vault documented previously (Cook et al., 2000) was assumed. The key assumption in the agriculture scenario is that an inadvertent intruder excavates into saltstone in digging a foundation for a home at the location of a disposal vault. A reevaluation of the dose analysis for the agriculture scenario described in Appendix A indicates that doses to inadvertent intruders would exceed the applicable performance measure of 100 mrem per year if the scenario were a credible occurrence. Therefore, an essential function of the redesigned cover system is to preclude the occurrence of the agriculture scenario during the 10,000-year time frame of concern to this analysis. That is, the additional 1-m thick layer of grout is intended to help ensure that excavation into saltstone is not a credible occurrence within 10,000 years. An assumption that the agriculture scenario is not credible during this time frame is based on arguments about the long-term performance of the cover system that are summarized in Table 4-4 and described in the following paragraphs.

First, consider the top layers of topsoil and backfill in the cover above the engineered barriers (see Table 4-3). These layers, which have a total thickness of 0.9 m, will erode over time. Average soil erosion rates in cropland areas near the SRS are about 1.5-2.0 kg/m^2 per year (U.S. Department of Agriculture, 1985). Thus, if an average density of soil of 1,400 kg/m^3 is assumed (Baes and Sharp, 1983), the soil erosion rate on cultivated lands is about 1 mm/y, or 1 m per 1,000 years. At this erosion rate, and assuming that the site would not be used for agricultural purposes until after the 100-year period of institutional control, the top 0.9-m thick layer of cover material would be removed by about 1,000 years. This estimate should be conservative, given that the presence of natural successional forests at the site would reduce the soil erosion rate by a factor of 400 to 500 compared with the erosion rate on cultivated lands (Horton and Wilhite, 1978). Use of the site for agricultural purposes should be discouraged because, first, a stand of persistent, shallow-rooted bamboo will be planted at the site to reduce erosion and enhance evapotranspiration and, second, the top of the cover system will be several meters above the elevation of the surrounding terrain (Cook et al., 2000). At the lower erosion rate that applies to undisturbed land, less than 5 cm of the cover should erode within 10,000 years.

For purposes of this analysis, the erosion rate of the top layers of the cover system is assumed to be 1 m per 1,000 years. Thus, the top 0.9-m layer of cover material is assumed to be removed at 1,000 years. The likelihood that the erosion rate will be substantially lower, and that little of the top layers of the cover system will be removed by erosion within 10,000 years, provides an added margin of safety.

Second, consider the topmost 0.3-m thick layer of gravel. In this analysis, it is assumed that this gravel layer would effectively prevent further erosion once the top 0.9-m thick layer of cover material is removed. This assumption is based on the likelihood that materials comprising the gravel layer will be too large to be transported by overland flow during extreme rain events and will be highly leach resistant (insoluble). Given the assumption that

Table 4-4. Assumptions about long-term performance of cover system on a disposal vault and saltstone used in present analysis[a]

Component of disposal system	Assumed performance
Topmost layers of topsoil and backfill	Entire 0.9-m thickness of material is removed by erosion by 1,000 years after disposal[b]
Topmost layer of gravel	0.3-m thick gravel layer provides barrier to further erosion of cover system
Layer of grout above reinforced concrete roof on disposal vault	1-m thick grout layer loses its physical integrity as barrier to excavation at same time as reinforced concrete roof[c]
Reinforced concrete roof on disposal vault	Roof loses its physical integrity as barrier to excavation at 1,000 years after disposal[d]
Layer of grout above saltstone in disposal vault	Weathers to soil-like material at rate of 0.1 m per 1,000 years; entire thickness of 0.4 m is weathered within 10,000 years[e]
Saltstone monolith	Weathers to soil-like material at rate of 0.1 m per 1,000 years after entire thickness of overlying grout layer has weathered[e]

[a] Bases for assumptions are described in Sect. 4.2.2.1.

[b] Assumption should be conservative; if disposal site is not used for agricultural purposes, less than 5 cm should be removed by erosion within 10,000 years.

[c] No additional credit is taken for performance of 1-m thick grout layer beyond that assumed for reinforced concrete roof.

[d] Assumption may be conservative; analysis of degradation indicates that roof may maintain its physical integrity for as long as 10,000 years (MMES et al., 1992).

[e] Assumed weathering rate is based on data on weathering rate of carbonate rock (limestone).

the topmost gravel layer is an erosion barrier, there is no need of assumptions about the performance of the rest of the cover above the engineered barriers, since these layers do not provide a barrier to excavation.

Third, consider the 0.1-m thick reinforced concrete roof on a disposal vault and the new 1-m thick layer of grout on top of the roof. Based on an analysis of the rate of degradation of the reinforced concrete roof in the existing PA (MMES et al., 1992), the roof is assumed to maintain its physical integrity and, thus, provide a barrier to excavation for 1,000 years after disposal. This assumption should be conservative because the previous analysis indicated that the roof could maintain its physical integrity for perhaps as long as 10,000 years. For purposes of this analysis, the additional 1-m thick layer of grout on top of the roof also is assumed to fail completely at 1,000 years. That is, it is assumed that this grout layer does not provide an additional barrier to excavation beyond that provided by the reinforced concrete roof. This assumption also should be conservative. It is invoked because the long-term performance of the reinforced concrete roof and overlying grout layer has not been analyzed, and it thus is difficult to justify taking substantial credit for the performance of the grout layer.

Finally, consider the layer of grout above saltstone in a disposal vault and the saltstone monolith itself. Based on the assumption used in the previous analysis (MMES et al., 1992) and discussed in Appendix A that these materials will weather to soil-like material at a rate of 0.1 m per 1,000 years, the entire 0.4-m thick layer of grout and at least 0.5 m of saltstone will weather to soil-like material within 10,000 years. Again, weathering of saltstone is assumed not to begin until the entire layer of overlying grout has weathered.

Based on the assumption about the time at which the reinforced concrete roof and 1-m thick layer of grout above the roof would fail and no longer provide a barrier to excavation (1,000 years) and the assumption about the weathering rate of the layer of grout above saltstone in a disposal vault and the saltstone monolith itself, the agriculture scenario would be a credible occurrence within the 10,000-year time frame of concern if the depth of an excavation in digging a foundation for a home could extend into saltstone during that time. However, based on the assumption that the topmost layer of gravel in the cover above the engineered barriers would provide a barrier to further erosion of the cover, the depth of material between the ground surface and saltstone would be at least 3.5 m at any time (see Table 4-3). A standard assumption developed by the USNRC for use in analyses of inadvertent intrusion is that a typical maximum depth of an excavation in digging a foundation for a home is 3 m (Oztunali and Roles, 1986). Thus, the depth of an excavation would not reach the depth of saltstone within 10,000 years, and the agriculture scenario is not a credible occurrence during that time frame.

These arguments may be summarized as follows. The addition of a 1-m thick layer of grout above the reinforced concrete roof on a disposal vault serves to preclude the agriculture scenario as a credible occurrence within 10,000 years by increasing the depth of saltstone below ground to greater than a typical maximum depth of an excavation in digging a foundation for a home. The conclusion that the agriculture scenario is not credible does not rely on an assumption that the engineered barriers, including the 1-m thick layer of grout, would provide a barrier to excavation into saltstone for 10,000 years.

The conclusion that the depth of an excavation would not reach the depth of saltstone within 10,000 years relies essentially on an assumption that the topmost layer of gravel in the cover system will provide an effective barrier to erosion. However, some of the other assumptions about the performance of the cover system include margins of safety that should help ensure that the agriculture scenario would not be a credible occurrence within 10,000 years. First, the disposal site may not be used for agricultural purposes after loss of institutional control, and very little of the top layers of topsoil and backfill would be expected to erode over 10,000 years if the site remains largely undisturbed. Second, the reinforced concrete roof may maintain its physical integrity for substantially longer than 1,000 years. Finally, the new 1-m thick layer of grout above the vault roof also may maintain its physical integrity for longer than 1,000 years.

4.2.2.2 Definition of Credible Resident Scenarios. As described in Appendix A and documented in the existing PA (MMES et al., 1992), a second exposure scenario for inadvertent intruders referred to as the resident scenario was included in the intruder dose analysis. As in the agriculture scenario, the resident scenario assumes that an intruder excavates a foundation for a home on top of a disposal vault. However, the resident scenario assumes that excavation into saltstone is precluded, either because the intruder encounters an intact engineered barrier (e.g., vault roof) that cannot be readily penetrated by the types of excavation equipment normally used in the vicinity of the SRS, or because the depth of buried waste (saltstone) is greater than a typical maximum depth of an excavation in digging a foundation for a home (i.e., 3 m). The resident scenario then occurs after the home is constructed, and the only relevant pathway is external exposure to photon-emitting radionuclides in the waste while residing in the home on top of shielded waste. The presence of uncontaminated material above the waste would preclude inhalation or ingestion exposure. Based on the conclusion discussed in the previous section that the agriculture scenario involving excavation into saltstone is not a credible occurrence within the 10,000-year time frame of concern to the intruder dose analysis, the only credible scenarios during this time frame are resident scenarios at different times after disposal and involving different thicknesses of shielding above the waste.

The resident scenario is a credible occurrence at any time after institutional control over the site is assumed to be relinquished at 100 years after disposal. However, the external dose in the resident scenario can increase over time due to a decrease in the thickness of shielding between saltstone and the depth of an excavation. The thickness of shielding can decrease as the cover material above the engineered barriers erodes and the engineered barriers above the waste lose their physical integrity and no longer deter excavation. Thus, the resident scenario needs to be evaluated at a number of times between 100 and 10,000 years after disposal. The assumed times of occurrence of the resident scenario and the thickness of shielding at these times are summarized in Table 4-5 and described in the following paragraphs.

Table 4-5. Summary of resident scenarios for exposure of inadvertent intruders evaluated in present analysis[a]

Time of occurrence of resident scenario	Assumptions	Importance of scenario
100 years	Intruder's home is constructed on top of intact engineered barriers Thickness of shielding provided by engineered barriers is 1.5 m	Determines limits on allowable disposals only for shorter-lived photon-emitting radionuclides (half-life on the order of 100 years or less)[b]
1,000 years	Top 0.9 m of cover material has eroded and 1-m thick layer of grout above reinforced concrete roof has lost its physical integrity as barrier to excavation Intruder's home is constructed at depth of 3 m, i.e., at depth of top of concrete roof Thickness of shielding provided by concrete roof and layer of grout above saltstone is 0.5 m	Determines limits on allowable disposals of longer-lived photon-emitting radionuclides that do not have important decay products whose activities increase with time beyond 1,000 years[b,c]
10,000 years	Topmost gravel layer in cover system at depth of 0.9 m provides barrier to further erosion beyond 1,000 years Intruder's home is constructed at same depth as at 1,000 years Thickness of shielding is same as at 1,000 years	Determines limits on allowable disposals of longer-lived photon-emitting radionuclides with important decay products whose activities increase with time beyond 1,000 years[b]

[a] Scenarios are discussed in Sect. 4.2.2.2.

[b] Limits on allowable disposals of shorter-lived radionuclides may be based on resident scenario at 1,000 or 10,000 years if radionuclide has longer-lived photon-emitting decay products.

[c] Resident scenario at 1,000 years determines disposal limits for all longer-lived radionuclides if time frame of 1,000 years for intruder dose analysis (USDOE, 1999a) is used.

At 100 years after disposal, which is the earliest time the resident scenario can occur, all engineered barriers above the waste are assumed to be intact. An assumption that the barriers to excavation will not degrade by a significant amount during the 100-year period of institutional control is reasonable when surveillance and maintenance of the cover system presumably will be performed during that time. An inadvertent intruder then is assumed to excavate to the depth of the intact engineered barriers (2.9 m), since this depth is less than a typical maximum depth of an excavation in digging a foundation for a home of 3 m noted previously, and the assumed thickness of shielding is 1.5 m (see Table 4-3).

The next time the resident scenario is evaluated is at 1,000 years. The assumptions about the resident scenario at this time are based on the assumed performance of the cover system and engineered barriers discussed in the previous section. The top 0.9-m thick layer of cover material is assumed to have eroded down to the topmost layer of gravel, and the layer of grout above the reinforced concrete roof is assumed to have lost its physical integrity as a barrier to excavation. Based on the assumed erosion of the cover and failure of the grout layer at 1,000 years, a 3-m deep excavation would reach only to the top of the reinforced concrete roof and, thus, the thickness of shielding at that time is 0.5 m (see Table 4-3). Although the concrete roof also is assumed to fail at 1,000 years and some weathering of the layer of grout beneath the roof could occur within this time, assumptions about the performance of these barriers over 1,000 years are not needed. The assumed thickness of shielding at 1,000 years may be conservative, since erosion of the top layer of cover material may be much less than assumed and some of the 1-m thick layer of grout may still be intact.

The final time the resident scenario is evaluated is at 10,000 years. All engineered barriers above the waste are assumed to have failed (i.e., lost their physical integrity) by this time. However, since the topmost gravel layer in the cover system is assumed to provide a barrier to erosion, a 3-m deep excavation would reach only to the depth of the top of the concrete roof. Thus, as in the resident scenario at 1,000 years, the thickness of shielding at 10,000 years is 0.5 m; the shielding is the same regardless of whether the engineered barriers are intact or fully degraded. It should be noted that the resident scenario involving external exposure on top of uncovered waste, which was included in the previous analysis (MMES et al., 1992), is no longer a credible occurrence within 10,000 years, as is also the case with the agriculture scenario discussed in the previous section.

Each of the resident scenarios at different times after disposal is important in determining limits on allowable disposals of photon-emitting radionuclides. The resident scenario at 100 years is used to determine disposal limits for shorter-lived radionuclides with a half-life on the order of 100 years or less (e.g., Cs-137). These radionuclides are unimportant at 1,000 years and beyond, due to the rapid depletion of their inventories by radioactive decay. Longer-lived radionuclides are unimportant in the analysis of the resident scenario at 100 years, because any depletion by decay at 1,000 years is more than compensated by the much higher external dose per unit concentration at that time, due to the considerable reduction in the thickness of shielding. However, the resident scenario at 1,000 or 10,000 years can be important for shorter-lived radionuclides that have longer-lived photon-emitting decay products.

The resident scenario at 1,000 years is used to determine disposal limits for longer-lived radionuclides that do not have photon-emitting decay products whose activities increase with time beyond 1,000 years (e.g., Sn-126). Disposal limits for longer-lived radionuclides of this type based on the resident scenario at 10,000 years would not exceed the limits based on the scenario at 1,000 years, since the same thickness of shielding is assumed at the two times and some depletion by radioactive decay would occur after 1,000 years. The resident scenario at 1,000 years also would determine disposal limits for all other longer-lived radionuclides if the time frame of 1,000 years for intruder dose analyses specified by USDOE (1999a) is used.

The resident scenario at 10,000 years is used to determine disposal limits for longer-lived radionuclides that have photon-emitting decay products whose activities increase with time beyond 1,000 years (e.g., U-233). In all such cases, the external dose at 10,000 years would be greater than at 1,000 years, since the same thickness of shielding is assumed at the two times. However, disposal limits would not be based on the resident scenario at 10,000 years if a time frame of 1,000 years for an intruder dose analysis (USDOE, 1999a) is used.

The approach of evaluating the resident scenario at discrete, well separated times can be justified on the following grounds. First, nearly all the radionuclides of concern (see Table 4-1) can be grouped into two categories according to their half-life: (1) radionuclides with a half-life less than about 100 years, which should decay to innocuous levels during the lifetime of engineered barriers and for which an analysis at 100 years thus should give the maximum credible dose; and (2) radionuclides with a half-life of a few thousand years or more, which do not decay appreciably during the lifetime of engineered barriers and for which an analysis at times after the engineered barriers are assumed to have failed would give the maximum credible dose. There are very few radionuclides with intermediate half-lives. The only important radionuclides in saltstone with intermediate half-lives are Am-241 and Am-242m (see Sect. 3.3). However, based on the data and analysis presented in Sects. 4.3 and 4.4, the external dose per unit concentration of Am-241 and Am-242m and limits on allowable disposals of these radionuclides are determined by the increases in activities of their longer-lived photon-emitting decay products over 10,000 years. Thus, given the thicknesses of shielding at various times assumed in the resident scenarios, Am-241 and Am-242m are similar to longer-lived radionuclides.

Second, the failure of engineered barriers and consequent reductions in the thickness of shielding above the waste over time have not been modeled or described in any detail. Rather, for the one engineered barrier of importance to the resident scenario—namely, the new 1-m thick layer of grout above the reinforced concrete roof—a simple, step-function model of failure over time is assumed. That is, the barrier essentially is assumed to fail (i.e., lose its physical integrity) instantaneously at 1,000 years. An effort to model the slow degradation of the grout layer over time and the resulting slow decrease in the amount of shielding that the remaining intact material would provide has not been undertaken and is far beyond the scope of this analysis. Given the absence of a detailed model of degradation of the grout layer, an attempt to evaluate the resident scenario essentially continuously over time would not necessarily give better results than the approach taken in this analysis of evaluating the resident scenario at discrete, well separated times.

4.2.2.3 Summary of Credible Scenarios. The discussions on the development of credible scenarios for exposure of inadvertent intruders at the SDF may be summarized as follows.

First, based on the addition of a 1-m thick layer of grout above the reinforced concrete roof in the cover system for a disposal vault, the agriculture scenario evaluated in the existing PA (MMES et al., 1992) is not considered to be a credible occurrence within 10,000 years. The essential function of the additional layer of grout is to increase the depth of saltstone below ground such that a 3-m deep excavation at any time within 10,000 years would not reach the depth of saltstone; the grout layer is not assumed to provide a barrier to excavation for 10,000 years. The conclusion that the agriculture scenario is not credible also depends on an assumption that the topmost layer of gravel in the cover material provides a barrier to erosion. However, there should be substantial margins of safety in the assumptions about the rate of erosion of the cover and the long-term performance of the engineered barriers, and these should help ensure that the agriculture scenario is not credible over the time period of concern.

Second, given that the agriculture scenario is not credible, the only credible scenario within 10,000 years is the resident scenario. This scenario assumes that an intruder constructs a home on top of a disposal vault and receives an external exposure during indoor residence. In contrast to the agriculture scenario, the resident scenario occurs even when excavation into the waste is not a credible occurrence. However, the magnitude of the dose depends on the thickness of shielding between the waste and the depth of an excavation.

The resident scenario is evaluated at 100, 1,000, and 10,000 years after disposal. The assumed thickness of shielding at 100 years is substantially higher than at the later times, due to the addition of the 1-m thick layer of grout above the reinforced concrete roof. The resident scenario at 100 years determines limits on allowable disposals of photon-emitting radionuclides with relatively short half-lives on the order of 100 years or less, except the resident scenario at longer times may be important when the radionuclide has longer-lived photon-emitting decay products. Based on assumptions that the grout layer above the roof would lose its physical integrity at 1,000 years and that the topmost layer of gravel in the cover material provides a barrier to erosion, the assumed thickness of shielding is the same at 1,000 and 10,000 years. The resident scenario at these times determines limits on allowable disposals of longer-lived photon-emitting radionuclides, with the scenario at 10,000 years used to determine limits for radionuclides with photon-emitting decay products whose activities increase with time beyond 1,000 years.

4.3 DOSE ANALYSIS FOR RESIDENT SCENARIO

In the resident scenario, the only exposure pathway of concern is external exposure during indoor residence in a home located on top of shielded waste in a disposal facility. Thus, the dose in this scenario depends on the external dose rate per unit activity concentration of each photon-emitting radionuclide in the waste. In this report, these factors are referred to as dose coefficients (ICRP, 1996; Eckerman and Ryman, 1993), rather than "dose conversion factors" as used, for example, in existing PAs at the SRS (MMES et al., 1992; McDowell-Boyer et al., 2000).

4.3.1 Dose Coefficients for External Exposure

Dose coefficients for external exposure depend on the distribution of radionuclides in the source region (saltstone), the amount of self-shielding provided by materials in the source region, and the shielding provided by uncontaminated materials between the source region and the assumed location of an exposed individual above ground. The source region is assumed to be infinitely thick and infinite in lateral extent. The assumption of an infinitely thick source region is appropriate when the depth of saltstone in a disposal vault is several meters (see Sect. 3.2) and a few meters of saltstone provides essentially complete shielding from all sources at greater depth (Kocher and Sjoreen, 1985). Based on an analysis described in Sect. A.2.3 and the dimensions of a disposal vault (see Sect. 3.2), an assumption that the source region is infinite in lateral extent results in negligible error in estimated doses.

During indoor residence, some shielding is provided by the walls and floor of the home. This shielding is not included in the external dose coefficients used in this analysis, but is treated as a separate parameter in the model to estimate external dose.

As described in Sect. 4.2.2.2 and summarized in Table 4-5, thicknesses of shielding above the waste of 0.5 and 1.5 m are assumed in evaluating the resident scenario at different times after disposal. External dose coefficients for these thicknesses of shielding are given in Table 4-6; the dose coefficients are based on an assumption that saltstone and grout are similar to soil in their shielding properties. This table also gives external dose coefficients that apply to exposure on top of uncovered (unshielded) waste. These dose coefficients are not used in the dose analysis for the resident scenario but are included in the table to show the large decreases in external dose that result from the substantial thicknesses of shielding assumed in the analysis.

The external dose coefficients in the case of no shielding in Table 4-6 are obtained from current Federal guidance (Eckerman and Ryman, 1993). The dose coefficients for the two thicknesses of shielding assumed in the resident scenario are based on calculations for monoenergetic sources by Kocher and Sjoreen (1985), since values are not given in Federal guidance. In the case of no shielding, dose coefficients obtained from the calculations of Kocher and Sjoreen agree with the Federal guidance within a few tens of percent and generally are higher. When there is no entry for a given radionuclide, either the radionuclide itself is not a photon emitter (e.g., Cs-137) or the energies of emitted photons are sufficiently low that the resulting external dose from shielded waste is inconsequential. When a radionuclide has shorter-lived photon-emitting decay products, the dose coefficient for each decay product takes into account the branching fraction in decay of its parent radionuclide given in Table 4-1; i.e., a parent and its shorter-lived decay products are assumed to be in activity equilibrium.

Table 4-6. External dose coefficients for radionuclides uniformly distributed in infinite thickness of soil-like material and different thicknesses of shielding between source and receptor locations[a]

	Dose coefficient (rem/y per μCi/m^3)[c]		
Nuclide[b]	No shielding[d]	0.5-m shielding[e]	1.5-m shielding[e]
Al-26	1.1×10^{-2}	2.3×10^{-4}	2.9×10^{-7}
Co-60	1.0×10^{-2}	1.7×10^{-4}	6.8×10^{-8}
Nb-94	6.1×10^{-3}	4.5×10^{-5}	3.6×10^{-9}
Sn-121m	1.2×10^{-6}	—	—
Sn-126	9.2×10^{-5}	—	—
Sb-126m	5.8×10^{-3}	2.9×10^{-5}	1.1×10^{-9}
Sb-126	1.5×10^{-3}	8.2×10^{-6}	4.6×10^{-10}
I-129	8.1×10^{-6}	—	—
Cs-137	—	—	—
Ba-137m	2.1×10^{-3}	1.1×10^{-5}	3.6×10^{-10}
Eu-152	4.4×10^{-3}	5.4×10^{-5}	1.9×10^{-8}
Eu-154	4.8×10^{-3}	6.0×10^{-5}	1.6×10^{-8}
Eu-155	1.1×10^{-4}	5.7×10^{-10}	9.9×10^{-20}
Ra-226	—	—	—
Pb-214	8.4×10^{-4}	—	—
Bi-214	6.1×10^{-3}	1.2×10^{-4}	1.2×10^{-7}

Table is continued on following page.

Table 4-6. (continued)

Nuclide[b]	Dose coefficient (rem/y per μCi/m^3)[c] No shielding[d]	0.5-m shielding[e]	1.5-m shielding[e]
Ra-228	—	—	—
Ac-228	3.7×10^{-3}	4.1×10^{-5}	1.5×10^{-8}
Pb-212	4.4×10^{-4}	—	—
Bi-212	7.3×10^{-4}	9.3×10^{-6}	5.4×10^{-9}
Tl-208	5.2×10^{-3}	1.8×10^{-4}	6.8×10^{-7}
Ac-227	—	—	—
Th-227	3.2×10^{-4}	1.8×10^{-7}	1.6×10^{-13}
Ra-223	3.8×10^{-4}	2.4×10^{-7}	3.8×10^{-13}
Pb-211	1.9×10^{-4}	1.1×10^{-6}	6.7×10^{-11}
Bi-211	1.6×10^{-4}	2.3×10^{-7}	5.1×10^{-13}
Tl-207	1.2×10^{-5}	7.8×10^{-8}	8.6×10^{-12}
Th-229	2.0×10^{-4}	9.7×10^{-9}	—
Ra-225	6.9×10^{-6}	—	—
Ac-225	4.0×10^{-5}	6.2×10^{-9}	—
Fr-221	9.6×10^{-5}	1.8×10^{-8}	—
Bi-213	4.8×10^{-4}	1.4×10^{-6}	5.0×10^{-11}
Tl-209	1.7×10^{-4}	3.3×10^{-6}	2.8×10^{-9}
Th-232	—	—	—
Ra-228[f]			

Table is continued on following page.

Table 4-6. (continued)

Nuclide[b]	Dose coefficient (rem/y per μCi/m^3)[c]		
	No shielding[d]	0.5-m shielding[e]	1.5-m shielding[e]
Pa-231	1.2×10^{-4}	7.9×10^{-8}	8.2×10^{-14}
Ac-227[f]			
U-232[g]	—	—	—
U-235	4.5×10^{-4}	4.6×10^{-8}	9.4×10^{-16}
Th-231	2.3×10^{-5}	—	—
U-238	—	—	—
Th-234	1.5×10^{-5}	—	—
Pa-234m	5.6×10^{-5}	4.4×10^{-7}	6.4×10^{-11}
Pa-234	1.2×10^{-5}	1.2×10^{-7}	3.5×10^{-11}
Np-237	4.9×10^{-5}	—	—
Pa-233	6.4×10^{-4}	6.5×10^{-7}	9.8×10^{-13}
Pu-244	—	—	—
Np-240m	1.3×10^{-3}	9.4×10^{-6}	2.3×10^{-9}
Am-241	2.7×10^{-5}	4.2×10^{-14}	—
Am-242m	1.1×10^{-6}	—	—
Am-242	3.1×10^{-5}	3.3×10^{-10}	—
Np-238	9.9×10^{-6}	1.2×10^{-7}	2.0×10^{-11}
Am-243	8.9×10^{-5}	—	—
Np-239	4.7×10^{-4}	1.5×10^{-7}	1.0×10^{-13}
Cm-243	3.6×10^{-4}	1.0×10^{-7}	3.4×10^{-14}
Cm-245	2.1×10^{-4}	6.0×10^{-9}	5.7×10^{-17}

Table is continued on following page.

Table 4-6. (continued)

Nuclide[b]	Dose coefficient (rem/y per μCi/m^3)[c]		
	No shielding[d]	0.5-m shielding[e]	1.5-m shielding[e]
Cm-247	1.1×10^{-3}	1.9×10^{-6}	6.8×10^{-12}
Pu-243	5.0×10^{-5}	—	—
Cf-249	1.2×10^{-3}	1.8×10^{-6}	5.0×10^{-12}
Cf-251	3.3×10^{-4}	3.9×10^{-8}	6.8×10^{-15}

[a] Values give external effective dose-equivalent rates per unit activity concentration in soil at height of 1 m above ground.

[b] Indented entries are radiologically significant shorter-lived decay products of parent radionuclide listed; all such shorter-lived decay products are assumed to be in activity equilibrium with parent radionuclide, and dose coefficient for each decay product takes into account branching fraction in decay of the parent given in Table 4-1.

[c] When no value is given, radionuclide is not a photon emitter or emits only lower-energy photons, and external dose is negligible.

[d] Values from Eckerman and Ryman (1993); dose coefficients apply to external exposure during indoor residence on top of unshielded waste, except values do not take into account additional shielding provided by walls and floor of house. Dose coefficients are not used in dose analysis for resident scenario, but are included to indicate large decreases in external dose rates that result from substantial thicknesses of shielding assumed in analysis.

[e] Values are based on calculations of absorbed dose in air at 1 m above ground from monoenergetic sources in soil by Kocher and Sjoreen (1985), energies and intensities of photons in decay of each radionuclide from Kocher (1981), and assumed factor of 0.8 to convert absorbed dose in air to effective dose equivalent (ICRP, 1996); dose coefficients apply to external exposure during indoor residence on top of shielded waste, except values do not take into account additional shielding provided by walls and floor of house.

[f] Dose coefficient for decay product is listed separately.

[g] Dose coefficients for shorter-lived decay products Pb-212, Bi-212, and Tl-208 are listed following entry for Ra-228.

4.3.2 Scenario Dose Conversion Factors

In the resident scenario, the annual effective dose equivalent (rem per year) from external exposure (e) to radionuclide i in the waste is given by

$$H_{i,e} = C_{i,\mathrm{V}} \times U \times D_{i,e} \times S\,, \tag{4.1}$$

where

$C_{i,\mathrm{V}}$	=	concentration of radionuclide i in disposal vault (μCi/m^3),
U	=	fraction of the year spent residing in the home on top of unshielded waste in disposal vault,
$D_{i,e}$	=	dose coefficient for external exposure to radionuclide i in waste in disposal vault (rem/y per μCi/m^3),
S	=	shielding factor during indoor residence.

The shielding factor (S) takes into account the reduction in external dose due to the presence of the walls and floor of the home.

In implementing the model in eq. (4.1), the fraction of the year that an inadvertent intruder spends residing in the home is assumed to be 0.5 (Oztunali et al., 1981). The exposure time indoors could be as much as a factor of two higher, although such an extreme value is highly unlikely. The assumed exposure time indoors is intended to be representative of an average individual residing on the disposal site. The shielding factor during indoor residence is assumed to be 0.7 for all photon-emitting radionuclides (USNRC, 1977). The shielding factor of 0.7 during indoor residence assumes that the flooring used in the home is a thin layer of wood or other material that provides only a small amount of shielding from sources beneath the home. Assumptions about how a home is constructed clearly are subjective, and many homes are constructed on top of a concrete foundation of substantial thickness. However, use of a minimal thickness of foundation or flooring material is not unknown and, thus, is assumed to provide a reasonably conservative basis for an intruder dose analysis. Dose coefficients for external exposure for the assumed thicknesses of shielding are given in Table 4-6.

Based on the model and parameter values described above, the estimates of annual effective dose equivalents per unit concentration of radionuclides in a disposal vault for the thicknesses of shielding assumed in the resident scenarios are given in Table 4-7. These results are the scenario dose conversion factors (SDCFs) used in this analysis. The SDCFs are used in the following section to derive limits on allowable disposals of radionuclides.

4.4 DERIVATION OF LIMITS ON ALLOWABLE DISPOSALS

In this section, the results of the dose assessment for inadvertent intruders based on the assumed resident scenarios, as summarized in Table 4-7, are used to determine limits on allowable disposals of radionuclides in the SDF. These limits are expressed in terms of average concentrations and total inventories of radionuclides in a single disposal vault.

Table 4-7. Annual effective dose equivalents to inadvertent intruders from external exposure while residing in home on top of shielded waste in resident scenario per unit concentration of radionuclides in a disposal vault[a]

	Annual dose (rem/y per μCi/m^3)[c]	
Nuclide[b]	0.5-m shielding	1.5-m shielding
Al-26	8.1×10^{-5}	1.0×10^{-7}
Co-60	6.0×10^{-5}	2.4×10^{-8}
Nb-94	1.6×10^{-5}	1.3×10^{-9}
Sn-121m	—	—
Sn-126 + d	1.3×10^{-5}	5.6×10^{-10}
I-129	—	—
Cs-137 + d	3.9×10^{-6}	1.3×10^{-10}
Eu-152	1.9×10^{-5}	6.7×10^{-9}
Eu-154	2.1×10^{-5}	5.6×10^{-9}
Eu-155	2.0×10^{-10}	3.5×10^{-20}
Ra-226 + d	4.2×10^{-5}	4.2×10^{-8}
Ra-228 + d	8.1×10^{-5}	2.5×10^{-7}
Ac-227 + d	6.3×10^{-7}	2.7×10^{-11}
Th-229 + d	1.6×10^{-6}	1.0×10^{-9}
Th-232[d]	—	—
Pa-231[e]	2.8×10^{-8}	2.9×10^{-14}
U-232 + d	6.7×10^{-5}	2.4×10^{-7}
U-235 + d	1.6×10^{-8}	3.3×10^{-16}
U-238 + d	2.0×10^{-7}	3.5×10^{-11}

Table is continued on following page.

Table 4-7. (continued)

Nuclide[b]	Annual dose (rem/y per μCi/m^3)[c] 0.5-m shielding	1.5-m shielding
Np-237 + d	2.3×10^{-7}	3.4×10^{-13}
Pu-244 + d	3.3×10^{-6}	8.1×10^{-10}
Am-241	1.5×10^{-14}	—
Am-242m + d	4.2×10^{-8}	7.0×10^{-12}
Am-243 + d	5.3×10^{-8}	3.5×10^{-14}
Cm-243	3.5×10^{-8}	1.2×10^{-14}
Cm-245	2.1×10^{-9}	2.0×10^{-17}
Cm-247 + d	6.7×10^{-7}	2.4×10^{-12}
Cf-249	6.3×10^{-7}	1.8×10^{-12}
Cf-251	1.4×10^{-8}	2.4×10^{-15}

[a] Values are annual effective dose equivalents per unit concentration for indicated thicknesses of shielding provided by uncontaminated materials above waste in a vault.

[b] "+ d" with some entries denotes that shorter-lived decay products are assumed to be in activity equilibrium with parent radionuclide; see Table 4-1 for listing of decay products and branching fractions.

[c] When no value is given, radionuclide emits only lower-energy photons and external dose is negligible.

[d] Contributions to doses from Ra-228 decay product are not included.

[e] Contributions to doses from Ac-227 decay product are not included.

The approach taken in this report of using assumed scenarios for inadvertent intrusion to derive limits on allowable disposals of radionuclides in waste differs from the approach taken in the existing PA (MMES et al., 1992). In the previous analysis, assumed inventories of radionuclides in waste intended for disposal in the SDF were used to estimate doses to inadvertent intruders based on assumed exposure scenarios.

This section is organized as follows. Sects. 4.4.1-4.4.3 present the general approach used to determine limits on allowable disposals of radionuclides in the SDF, taking into account the results of the dose assessment for the resident scenario in Table 4-7, the performance measure for chronic exposure of inadvertent intruders of 100 mrem per year (USDOE, 1999a), and other factors of importance in estimating doses to inadvertent intruders including radioactive decay and the layout of disposal vaults in the SDF. The limits on allowable disposals are derived and discussed in Sects. 4.4.4, 4.4.5, and 4.4.6. Issues of sensitivity and uncertainty in the model for assessing external dose to inadvertent intruders are discussed in Sects. 4.4.7 and 4.4.8.

4.4.1 General Approach to Determining Limits on Allowable Disposals

In any exposure scenario for inadvertent intruders, the annual effective dose equivalent, H, from exposure to a particular radionuclide at the time t after disposal when the scenario is assumed to occur can be expressed as

$$H(t) = \overline{C}_V(t) \times \text{SDCF} , \tag{4.2}$$

where $\overline{C}_V$ is the average concentration of the radionuclide in a disposal vault at time t and SDCF is the scenario dose conversion factor. The use of average concentrations of radionuclides in a disposal vault, rather than the maximum concentrations at any location in a vault, is appropriate when an inadvertent intruder would access a vault at random locations.

The dose in eq. (4.2) is expressed in terms of the average concentration of a radionuclide at the time an exposure scenario is assumed to occur. For purposes of establishing waste acceptance criteria, the desired quantity is the average concentration in waste (saltstone) at the time of disposal. These concentrations are related by two factors. The first is radioactive decay, which changes the inventories of radionuclides between the time of disposal and the time an exposure scenario is assumed to occur. The second is a so-called waste dilution factor, which takes into account that, even in the absence of radioactive decay, average concentrations of radionuclides in materials to which an intruder would be exposed would be less than the average concentrations in disposed waste, due to the layout of the disposal vaults and the presence of uncontaminated materials in a disposal vault and its cover. Thus, the average concentration of a radionuclide in saltstone at the time of disposal, denoted by $\overline{C}_W$, can be represented by

$$\overline{C}_W = \overline{C}_V(t)/[G \times f_D(t)] , \tag{4.3}$$

where G denotes the waste dilution factor and f_D denotes the change in inventory with time due to radioactive decay.

By combining eqs. (4.2) and (4.3), the annual effective dose equivalent at the assumed time of occurrence of an exposure scenario, *t*, can be expressed in terms of the average concentration of a radionuclide in saltstone at the time of disposal by

$$H(t) = \overline{C}_W \times \text{SDCF} \times G \times f_D(t) \; . \tag{4.4}$$

Since the performance measure for chronic exposure of inadvertent intruders is an annual effective dose equivalent of 0.1 rem (USDOE, 1999a), the limit on allowable average concentration of a radionuclide in saltstone at the time of disposal can be represented by

$$\overline{C}_W = (0.1 \text{ rem/y})/[\text{SDCF} \times G \times f_D(t)] \; , \tag{4.5}$$

where SDCF is in units of rem/y per μCi/m^3 and $\overline{C}_W$ is in units of μCi/m^3. The waste dilution factor, *G*, that applies to the resident scenario and the radioactive decay factor, f_D, are described in the following sections.

4.4.2 Waste Dilution Factor

The waste dilution factor, *G*, in eq. (4.5) takes into account that the average concentrations of radionuclides to which an intruder would be exposed generally would be less than the average concentrations in disposed waste (saltstone), even in the absence of radioactive decay.

The waste dilution factor that applies to the resident scenario is based on consideration that an inadvertent intruder would access disposal vaults at random locations in the absence of knowledge of prior disposal activities at the site (USDOE, 1999a). Thus, on average, when an inadvertent intruder excavates at the disposal site in digging a foundation for a home, as assumed in the resident scenario, some fraction of the excavated area would be located in the region between adjacent disposal vaults where only uncontaminated materials are found. The dose from external exposure in the resident scenario then would be reduced by a factor equal to the fraction of the excavated area that is occupied by a disposal vault.

The waste dilution factor can be estimated based on the dimensions of an individual vault and the spacing between vaults. All but one of the vaults will be 60 m wide and 180 m long, and the dimensions of Vault 1 are 30 m × 180 m (see Sect. 3.2). Based on the planned layout of vaults shown in Fig. 3-3, the spacing between adjacent vaults in a row is 35 m and the spacing between Vaults 1-6 and Vaults 7-12 is 60 m. From these dimensions, about 60% of the area inside an envelope encompassing Vaults 1-12 is occupied by vaults; i.e., the waste dilution factor is 0.6. This estimate ignores the thickness of the exterior walls of a vault and the presence of interior walls that define disposal cells within a vault. However, the area occupied by these walls is insignificant. The estimated waste dilution factor also ignores Vaults 13-15, which will be located some distance from Vaults 1-12. Including these vaults would lower the waste dilution factor, but not by a large amount. In effect, the estimated waste dilution factor of 0.6 assumes that an excavation will occur in the region occupied by Vaults 1-12. These vaults occupy most of the land area that will be dedicated to waste disposal at the site.

In summary, the waste dilution factor, *G*, that applies to the resident scenario is 0.6. This factor takes into account that when an inadvertent intruder excavates at random locations at the site of disposal vaults, 60% of land area excavated will be occupied by vaults and the remaining 40% will be occupied by uncontaminated materials between vaults.

4.4.3 Radioactive Decay Factor

The SDCFs in Table 4-7 do not take into account radioactive decay between the time of disposal and the time that a resident scenario is assumed to occur. For radionuclides that do not have radiologically significant decay products or radionuclides that have only shorter-lived decay products that were taken into account in calculating the SDCFs (e.g., Sn-126 and Cs-137), the radioactive decay factor in eq. (4.5) is given by

$$f_D(t) = \exp(-0.693t/T_{½}) , \tag{4.6}$$

where $T_{½}$ is the radionuclide half-life and *t* again is the time after disposal when an exposure scenario is assumed to occur.

When significant buildup of longer-lived decay products would occur prior the assumed time of occurrence of an exposure scenario, the limit on allowable average concentration of the parent radionuclide at the time of disposal is obtained by replacing the term SDCF × $f_D(t)$ in eq. (4.5) by a sum of products of the SDCFs and decay factors for all radiologically significant members of the decay chain, including the parent, given by

$$\Sigma\,[\mathrm{SDCF}_i \times f_{D,i}(t)] , \; i = \text{radionuclide index} , \tag{4.7}$$

where the decay factor $f_{D,i}(t)$ is the activity of the *i*th radionuclide in the decay chain at time *t* relative to the initial activity of the parent radionuclide (i = 1), which is calculated using the Bateman equations for radioactive decay (Evans, 1955). Long-term buildup of important decay products can occur either when the decay products are shorter-lived than the parent (e.g., Th-230, Ra-226, and Pb-210 produced in decay of U-234) or when the decay products are longer-lived than the parent (e.g., Am-241, Np-237, and Th-229 produced in decay of Pu-241). When a decay product is much longer-lived than its parent, the maximum activity of the decay product essentially is given by the initial activity of the parent multiplied by the ratio of the half-life of the parent to the half-life of the decay product.

4.4.4 Limits on Allowable Disposals Based on Resident Scenario at Different Times

In this analysis, limits on allowable disposals of radionuclides in the SDF are calculated based on the resident scenario at 100, 1,000, and 10,000 years after disposal (see Table 4-5). At each time, limits on allowable average concentrations of the radionuclides of concern are calculated using eq. (4.5). The SDCFs are obtained from Table 4-7, the waste dilution factor, *G*, is 0.6, and the radioactive decay factor, f_D, at the assumed time of occurrence of the scenario, *t*, is applied as described in Sect. 4.4.3 and eqs. (4.6) and (4.7).

Each disposal vault is treated as a separate disposal unit for the purpose of determining limits on allowable inventories of radionuclides in saltstone. In all vaults except one, each of the 12 cells is 30 m × 30 m and will be filled with saltstone to a height of 7.3 m (see Sect. 3.2). Thus, the expected volume of saltstone in each of these vaults is 7.9×10^4 m^3. In the one remaining vault (Vault 1 in Fig. 3-3), the dimensions of each cell are the same, but there are only 6 cells, and the expected volume of saltstone thus is 3.9×10^4 m^3. If the volume of saltstone in a vault is denoted by V_W, the allowable inventory of a radionuclide per vault, I_W, is given by

$$I_W(\text{Ci}) = \overline{C_w}\,(\mu\text{Ci/m}^3) \times (10^{-6}\ \text{Ci/}\mu\text{Ci}) \times V_W(\text{m}^3)\ . \qquad (4.8)$$

In Vaults 2-15, the allowable inventory in curies (Ci) is $0.079 \times \overline{C_w}\,(\mu\text{Ci/m}^3)$, and the allowable inventory in Vault 1 is half this amount.

The limits on allowable average concentrations and inventories of radionuclides in saltstone at the time of disposal, as obtained from the results of the dose assessment for the resident scenario at 100, 1,000, and 10,000 years, are given in Tables 4-8, 4-9, and 4-10, respectively. The radionuclides included in the analysis at each time are selected as described in Sect. 4.2.2.2 and summarized in Table 4-5. The following points about these results should be noted.

First, limits on allowable disposals of Th-232 and Pa-231 are calculated by assuming that their shorter-lived decay products Ra-228 and Ac-227, respectively, are in activity equilibrium with the parent radionuclide. Therefore, the SDCFs for these radionuclides are the sum of the values for the parent and its decay product.

Table 4-8. Limits on allowable average concentrations and inventories of radionuclides per vault in SDF based on resident scenario for inadvertent intruders at 100 years after disposal[a]

Nuclide	SDCF × G (rem/y per μCi/m^3)[b]	f_D[c]	$\overline{C}_W$ (μCi/m^3)[d]	Inventory (Ci per vault)[e]
Co-60	1.4×10^{-8}	1.9×10^{-6}	3.8×10^{12}	3.0×10^{11}
Cs-137	7.8×10^{-11}	1.0×10^{-1}	1.3×10^{10}	1.0×10^{9}
Eu-152	4.0×10^{-9}	5.9×10^{-3}	4.2×10^{9}	3.3×10^{8}
Eu-154	3.4×10^{-9}	3.1×10^{-4}	9.5×10^{10}	7.5×10^{9}
Eu-155	2.1×10^{-20}	4.8×10^{-7}	No limit[f]	No limit
Ra-228	1.5×10^{-7}	5.8×10^{-6}	1.1×10^{11}	9.1×10^{9}
Ac-227	1.6×10^{-11}	4.1×10^{-2}	1.5×10^{11}	1.2×10^{10}
U-232	1.4×10^{-7}	3.7×10^{-1}	1.9×10^{6}	1.5×10^{5}
Am-242m	4.2×10^{-12}	6.1×10^{-1}	3.9×10^{10}	3.1×10^{9}
Cm-243	7.2×10^{-15}	8.8×10^{-2}	1.6×10^{14}	1.2×10^{13}

[a] Limits on allowable disposals are calculated using eqs. (4.5) and (4.8). Disposal limits are calculated only for photon-emitting radionuclides with a half-life on the order of 100 years or less (see Sect. 4.2.2.2 and Table 4-5).

[b] Scenario dose conversion factors (SDCFs) are values for 1.5 m of shielding given in Table 4-7, and waste dilution factor (G) is 0.6 (see Sect. 4.4.2).

[c] Radioactive decay factor giving radionuclide inventory at time resident scenario is assumed to occur (100 years) relative to inventory at time of disposal.

[d] Limits on allowable average concentrations of radionuclides in saltstone at time of disposal.

[e] Limits on allowable inventories of radionuclides in saltstone at time of disposal in Vaults 2-15 (see Fig. 3-3); limits on allowable inventories in Vault 1 are half the given values.

[f] Calculated concentration limit exceeds specific activity of isotope.

Table 4-9. Limits on allowable average concentrations and inventories of radionuclides per vault in SDF based on resident scenario for inadvertent intruders at 1,000 years after disposal[a]

Nuclide[b]	SDCF × G (rem/y per μCi/m^3)[c]	f_D[d]	$\overline{C}_W$ (μCi/m^3)[e]	Inventory (Ci per vault)[f]
Al-26	4.9×10^{-5}	1.0	2.0×10^{3}	1.6×10^{2}
Nb-94	9.6×10^{-6}	9.7×10^{-1}	1.1×10^{4}	8.5×10^{2}
Sn-126	7.8×10^{-6}	9.9×10^{-1}	1.3×10^{4}	1.0×10^{3}
Ra-226	2.5×10^{-5}	6.5×10^{-1}	6.2×10^{3}	4.9×10^{2}
Th-229	9.6×10^{-7}	9.1×10^{-1}	1.1×10^{5}	9.0×10^{3}
Th-230	—	—	1.1×10^{4}	9.0×10^{2}
Ra-226	2.5×10^{-5}	3.5×10^{-1}		
Th-232[g]	4.9×10^{-5}	1.0	2.0×10^{3}	1.6×10^{2}
Pa-231[h]	4.0×10^{-7}	9.8×10^{-1}	2.6×10^{5}	2.0×10^{4}
U-233	—	—	1.2×10^{6}	9.1×10^{4}
Th-229	9.6×10^{-7}	9.0×10^{-2}		
U-234	—	—	2.4×10^{6}	1.9×10^{5}
Ra-226	2.5×10^{-5}	1.7×10^{-3}		
U-235	9.6×10^{-9}	1.0	5.6×10^{6}	4.4×10^{5}
Pa-231[h]	4.0×10^{-7}	2.1×10^{-2}		
U-236	—	—	No limit[i]	No limit
Th-232[g]	4.9×10^{-5}	4.9×10^{-8}		
U-238	1.2×10^{-7}	1.0	8.3×10^{5}	6.6×10^{4}
Np-237	1.4×10^{-7}	1.0	7.1×10^{5}	5.6×10^{4}
Pu-238	—	—	8.2×10^{9}	6.4×10^{8}
Ra-226	2.5×10^{-5}	4.9×10^{-7}		

Table is continued on following page.

Table 4-9. (continued)

Nuclide[b]	SDCF × G (rem/y per μCi/m^3)[c]	f_D[d]	$\overline{C}_W$ (μCi/m^3)[e]	Inventory (Ci per vault)[f]
Pu-239	—	—	No limit[i]	No limit
U-235	9.6×10^{-9}	9.7×10^{-7}		
Pa-231[h]	4.0×10^{-7}	1.0×10^{-8}		
Pu-240	—	—	No limit[i]	No limit
Th-232[g]	4.9×10^{-5}	7.0×10^{-13}		
Pu-241	—	—	1.3×10^{11}	1.1×10^{10}
Np-237	1.4×10^{-7}	5.3×10^{-6}		
Pu-242	—	—	No limit[i]	No limit
U-238	1.2×10^{-7}	1.5×10^{-7}		
Pu-244[j]	2.0×10^{-6}	1.0	5.0×10^{4}	4.0×10^{3}
Am-241	9.0×10^{-15}	2.0×10^{-1}	4.5×10^{9}	3.5×10^{8}
Np-237	1.4×10^{-7}	1.6×10^{-4}		
Am-242m	2.5×10^{-8}	7.3×10^{-3}	5.3×10^{8}	4.2×10^{7}
Ra-226	2.5×10^{-5}	4.3×10^{-7}		
Am-243[j]	3.2×10^{-8}	9.1×10^{-1}	3.4×10^{6}	2.7×10^{5}
Cm-242	—	—	1.6×10^{12}	1.3×10^{11}
Ra-226	2.5×10^{-5}	2.5×10^{-9}		
Cm-243	—	—	No limit[i]	No limit
U-235	9.6×10^{-9}	1.1×10^{-9}		
Pa-231[h]	4.0×10^{-7}	1.1×10^{-11}		
Cm-244	—	—	No limit[i]	No limit
Th-232[g]	4.9×10^{-5}	1.8×10^{-15}		
Cm-245	1.3×10^{-9}	9.2×10^{-1}	8.4×10^{7}	6.6×10^{6}
Cm-246	—	—	No limit[i]	No limit
U-238	1.2×10^{-7}	1.4×10^{-10}		

Table is continued on following page.

Table 4-9. (continued)

Nuclide[b]	SDCF × G (rem/y per μCi/m^3)[c]	f_D[d]	$\overline{C}_W$ (μCi/m^3)[e]	Inventory (Ci per vault)[f]
Cm-247	4.0×10^{-7}	1.0	2.5×10^{5}	2.0×10^{4}
Am-243	3.2×10^{-8}	9.0×10^{-2}		
Cm-248	—	—	5.8×10^{9}	4.6×10^{8}
Pu-244	2.0×10^{-6}	8.6×10^{-6}		
Bk-249	—	—	7.3×10^{8}	5.8×10^{7}
Cf-249	3.8×10^{-7}	3.6×10^{-4}		
Cf-249	3.8×10^{-7}	1.4×10^{-1}	1.9×10^{6}	1.5×10^{5}
Cf-250	—	—	No limit[i]	No limit
U-238	1.2×10^{-7}	3.6×10^{-13}		
Cf-251	8.4×10^{-9}	4.6×10^{-1}	2.6×10^{7}	2.0×10^{6}
Cf-252	—	—	7.6×10^{14}	6.0×10^{13}
Pu-244	2.0×10^{-6}	6.6×10^{-11}		

[a] Limits on allowable disposals are calculated using eqs. (4.5) and (4.8). Disposal limits are calculated for all photon-emitting radionuclides with a half-life on the order of 100 years or greater (see Sect. 4.2.2.2 and Table 4-5).

[b] Indented entries are radiologically significant decay products of parent radionuclide listed (see Tables 4-1 and 4-2). Omitted photon-emitting decay products do not contribute significantly to dose.

[c] Scenario dose conversion factors (SDCFs) are values for 0.5 m of shielding given in Table 4-7, and waste dilution factor (G) is 0.6 (see Sect. 4.4.2). When no value is given, radionuclide is not a significant photon emitter.

[d] Radioactive decay factor giving radionuclide inventory at time resident scenario is assumed to occur (1,000 years) relative to inventory of radionuclide or its parent at time of disposal. When no value is given, radionuclide is not a significant photon emitter or inventory at 1,000 years is negligible.

[e] Limits on allowable average concentrations of radionuclides in saltstone at time of disposal.

[f] Limits on allowable inventories of radionuclides in saltstone at time of disposal for Vaults 2-15 (see Fig. 3-3); limits on allowable inventories in Vault 1 are half the given values.

[g] Contribution from Ra-228 decay product is included, based on assumption of activity equilibrium with parent radionuclide.

[h] Contribution from Ac-227 decay product is included, based on assumption of activity equilibrium with parent radionuclide.

[i] Calculated concentration limit exceeds specific activity of isotope.

[j] Buildup of activity of long-lived photon-emitting decay products over 10,000 years is insignificant.

Table 4-10. Limits on allowable average concentrations and inventories of radionuclides per vault in SDF based on resident scenario for inadvertent intruders at 10,000 years after disposal[a]

Nuclide[b]	SDCF × G (rem/y per μCi/m^3)[c]	f_D[d]	$\overline{C}_w$ (μCi/m^3)[e]	Inventory (Ci per vault)[f]
Th-230	—	—	4.3×10^3	3.4×10^2
Ra-226	2.5×10^{-5}	9.2×10^{-1}		
U-233	—	—	1.7×10^5	1.4×10^4
Th-229	9.6×10^{-7}	6.0×10^{-1}		
U-234	—	—	5.9×10^4	4.6×10^3
Ra-226	2.5×10^{-5}	6.8×10^{-2}		
U-235	9.6×10^{-9}	1.0	1.2×10^6	9.2×10^4
Pa-231[g]	4.0×10^{-7}	1.9×10^{-1}		
U-236	—	—	No limit[i]	No limit
Th-232[h]	4.9×10^{-5}	4.9×10^{-7}		
U-238	1.2×10^{-7}	1.0	7.1×10^5	5.6×10^4
Ra-226	2.5×10^{-5}	8.1×10^{-4}		
Np-237	1.4×10^{-7}	1.0	6.7×10^5	5.3×10^4
Th-229	9.6×10^{-7}	1.4×10^{-2}		
Pu-238	—	—	1.7×10^8	1.3×10^7
Ra-226	2.5×10^{-5}	2.4×10^{-5}		
Pu-239	—	—	2.3×10^{11}	1.8×10^{10}
U-235	9.6×10^{-9}	8.6×10^{-6}		
Pa-231[g]	4.0×10^{-7}	8.8×10^{-7}		
Pu-240	—	—	No limit[i]	No limit
Th-232[h]	4.9×10^{-5}	5.3×10^{-11}		
Pu-241	—	—	1.0×10^{11}	7.9×10^9
Np-237	1.4×10^{-7}	6.6×10^{-6}		
Th-229	9.6×10^{-7}	9.0×10^{-8}		

Table is continued on following page.

Table 4-10. (continued)

Nuclide[b]	SDCF × G (rem/y per μCi/m^3)[c]	f_D[d]	$\overline{C}_w$ (μCi/m^3)[e]	Inventory (Ci per vault)[f]
Pu-242	—	—	No limit[i]	No limit
U-238	1.2×10^{-7}	1.5×10^{-6}		
Ra-226	2.5×10^{-5}	5.5×10^{-9}		
Am-241	—	—	3.2×10^{9}	2.5×10^{8}
Np-237	1.4×10^{-7}	2.0×10^{-4}		
Th-229	9.6×10^{-7}	2.7×10^{-6}		
Am-242m	2.5×10^{-8}	—	1.3×10^{8}	1.0×10^{7}
Ra-226	2.5×10^{-5}	3.1×10^{-5}		
Cm-242	—	—	3.3×10^{10}	2.6×10^{9}
Ra-226	2.5×10^{-5}	1.2×10^{-7}		
Cm-243	—	—	2.0×10^{14}	1.6×10^{13}
U-235	9.6×10^{-9}	1.0×10^{-8}		
Pa-231[g]	4.0×10^{-7}	1.0×10^{-9}		
Cm-244	—	—	No limit[i]	No limit
Th-232[h]	4.9×10^{-5}	1.5×10^{-13}		
Cm-245	1.3×10^{-9}	4.4×10^{-1}	1.1×10^{8}	9.1×10^{6}
Np-237	1.4×10^{-7}	2.1×10^{-3}		
Th-229	9.6×10^{-7}	1.2×10^{-5}		
Cm-246	—	—	No limit[i]	No limit
U-238	1.2×10^{-7}	9.2×10^{-9}		
Ra-226	2.5×10^{-5}	1.2×10^{-12}		
Cm-247	4.0×10^{-7}	1.0	2.4×10^{5}	1.9×10^{4}
Am-243	3.2×10^{-8}	6.1×10^{-1}		
Cm-248	—	—	5.8×10^{8}	4.6×10^{7}
Pu-244	2.0×10^{-6}	8.6×10^{-5}		

Table is continued on following page.

Table 4-10. (continued)

Nuclide[b]	SDCF × G (rem/y per μCi/m^3)[c]	f_D[d]	$\overline{C}_W$ (μCi/m^3)[e]	Inventory (Ci per vault)[f]
Bk-249	—	—	1.0×10^{12}	8.2×10^{10}
Cm-245	1.3×10^{-9}	4.9×10^{-5}		
Np-237	1.4×10^{-7}	2.2×10^{-7}		
Th-229	9.6×10^{-7}	1.1×10^{-9}		
Cf-249	—	—	2.7×10^{9}	2.1×10^{8}
Cm-245	1.3×10^{-9}	1.9×10^{-2}		
Np-237	1.4×10^{-7}	8.4×10^{-5}		
Th-229	9.6×10^{-7}	4.4×10^{-7}		
Cf-250	—	—	No limit[i]	No limit
U-238	1.2×10^{-7}	2.5×10^{-11}		
Ra-226	2.5×10^{-5}	3.1×10^{-15}		
Cf-251	8.4×10^{-9}	4.4×10^{-4}	3.6×10^{9}	2.8×10^{8}
Cm-247	4.0×10^{-7}	5.8×10^{-5}		
Am-243	3.2×10^{-8}	3.2×10^{-5}		
Cf-252	—	—	7.7×10^{13}	6.1×10^{12}
Pu-244	2.0×10^{-6}	6.5×10^{-10}		

Footnotes to table are given on following page.

Footnotes to Table 4-10

[a] Limits on allowable disposals are calculated using eqs. (4.5) and (4.8). Disposal limits are calculated only for photon-emitting radionuclides that have long-lived decay products whose activities increase with time beyond 1,000 years (see Sect. 4.2.2.2 and Table 4-5).

[b] Indented entries are radiologically significant decay products of parent radionuclide listed (see Tables 4-1 and 4-2). Omitted photon-emitting decay products do not contribute significantly to dose.

[c] Scenario dose conversion factors (SDCFs) are values for 0.5 m of shielding given in Table 4-7, and waste dilution factor (G) is 0.6 (see Sect. 4.4.2). When no value is given, radionuclide is not a significant photon emitter.

[d] Radioactive decay factor giving radionuclide inventory at time resident scenario is assumed to occur (10,000 years) relative to inventory of radionuclide or its parent at time of disposal. When no value is given, radionuclide is not a significant photon emitter or inventory at 10,000 years is negligible.

[e] Limits on allowable average concentrations of radionuclides in saltstone at time of disposal.

[f] Limits on allowable inventories of radionuclides in saltstone at time of disposal for Vaults 2-15 (see Fig. 3-3); limits on allowable inventories in Vault 1 are half the given values.

[g] Contribution from Ac-227 decay product is included, based on assumption of activity equilibrium with parent radionuclide.

[h] Contribution from Ra-228 decay product is included, based on assumption of activity equilibrium with parent radionuclide.

[i] Calculated concentration limit exceeds specific activity of isotope.

Second, for many radionuclides that are not significant photon emitters, limits on allowable disposals based on the resident scenario at 1,000 or 10,000 years (Tables 4-9 and 4-10) are very high. For these radionuclides, the external dose is due entirely to buildup of photon-emitting decay products that have half-lives much longer than the half-life of the parent and, thus, much lower activities at 1,000 or 10,000 years relative to the activity of the parent at the time of disposal. In some of these cases, limits on allowable average concentrations are greater than the specific activity of the isotope, and in all such cases, limits on allowable average concentrations are much higher than concentrations that could occur in any wastes.

Third, the limits on allowable disposals in Tables 4-8, 4-9, and 4-10 apply to each radionuclide individually. In order to comply with the performance measure for chronic exposure of inadvertent intruders of 0.1 rem per year when more than one radionuclide is present in waste, the sum-of-fractions rule should be applied. The sum-of-fractions rule states that the ratio of the average concentration of each radionuclide in saltstone at the time of disposal to its limit on allowable concentration, when summed over all radionuclides, should not exceed unity, and similarly for the total inventory per vault.

4.4.5 Summary of Limits on Allowable Disposals Based on Resident Scenario

For any radionuclide, limits on the allowable average concentration and total inventory per vault are the most restrictive of the limits for the resident scenario at different times after disposal. In this analysis, limits on allowable disposals based on two time frames for assessments of inadvertent intrusion are provided. The first set of disposal limits given in Table 4-11 is based on an assumption that the appropriate time frame is 10,000 years, to be consistent with the SRS DAS (Fiori and Frei, 1999), and these limits are the most restrictive of the values given in Tables 4-8, 4-9, and 4-10. The second set of disposal limits given in Table 4-12 is based on the time frame of 1,000 years specified by USDOE (1999a), and these limits are the more restrictive of the values given in Tables 4-8 and 4-9 only. Although disposal limits are given to two significant figures, only the leading figure should be considered significant in applying the results. This caution is based on consideration of uncertainties in the model and data used to estimate external dose and the simplistic nature of the assumed resident scenarios.

Of the 60 radionuclides considered in this analysis (see Sect. 4.1 and Table 4-1), 43 have disposal limits based on the resident scenario at 100, 1,000, or 10,000 years. The remaining radionuclides for which there is no disposal limit based on the resident scenarios are not significant photon emitters and do not have significant photon-emitting decay products.

Table 4-11. Limits on allowable average concentrations and inventories of radionuclides per vault in SDF based on resident scenarios and assumption of 10,000-year time frame for assessments of inadvertent intrusion[a]

Nuclide	Limiting scenario[b]	Concentration limit ($\mu Ci/m^3$)[c]	Inventory limit (Ci per vault)[d]
H-3	None	No limit	No limit
Be-10	None	No limit	No limit
C-14	None	No limit	No limit
Al-26	Resident scenario at 1,000 years	2.0×10^{3}	1.6×10^{2}
Co-60	Resident scenario at 100 years	3.8×10^{12}	3.0×10^{11}
Ni-59	None	No limit	No limit
Ni-63	None	No limit	No limit
Se-79	None	No limit	No limit
Sr-90	None	No limit	No limit
Zr-93	None	No limit	No limit
Nb-93m	None	No limit	No limit
Nb-94	Resident scenario at 1,000 years	1.1×10^{4}	8.5×10^{2}
Tc-99	None	No limit	No limit
Pd-107	None	No limit	No limit
Cd-113m	None	No limit	No limit
Sn-121m	None	No limit	No limit
Sn-126	Resident scenario at 1,000 years	1.3×10^{4}	1.0×10^{3}
I-129	None	No limit	No limit
Cs-135	None	No limit	No limit
Cs-137	Resident scenario at 100 years	1.3×10^{10}	1.0×10^{9}

Table is continued on following page.

Table 4-11. (continued)

Nuclide	Limiting condition[b]	Concentration limit ($\mu Ci/m^3$)[c]	Inventory limit (Ci per vault)[d]
Sm-151	None	No limit	No limit
Eu-152	Resident scenario at 100 years	4.2×10^{9}	3.3×10^{8}
Eu-154	Resident scenario at 100 years	9.5×10^{10}	7.5×10^{9}
Eu-155	Resident scenario at 100 years	No limit[e]	No limit
Pb-210	None	No limit	No limit
Ra-226	Resident scenario at 1,000 years	6.2×10^{3}	4.9×10^{2}
Ra-228	Resident scenario at 100 years	1.1×10^{11}	9.1×10^{9}
Ac-227	Resident scenario at 100 years	1.5×10^{11}	1.2×10^{10}
Th-229	Resident scenario at 1,000 years	1.1×10^{5}	9.0×10^{3}
Th-230	Resident scenario at 10,000 years	4.3×10^{3}	3.4×10^{2}
Th-232	Resident scenario at 1,000 years	2.0×10^{3}	1.6×10^{2}
Pa-231	Resident scenario at 1,000 years	2.6×10^{5}	2.0×10^{4}
U-232	Resident scenario at 100 years	1.9×10^{6}	1.5×10^{5}
U-233	Resident scenario at 10,000 years	1.7×10^{5}	1.4×10^{4}
U-234	Resident scenario at 10,000 years	5.9×10^{4}	4.6×10^{3}
U-235	Resident scenario at 10,000 years	1.2×10^{6}	9.2×10^{4}
U-236	Resident scenario at 10,000 years	No limit[e]	No limit
U-238	Resident scenario at 10,000 years	7.1×10^{5}	5.6×10^{4}
Np-237	Resident scenario at 10,000 years	6.7×10^{5}[f]	5.3×10^{4}
Pu-238	Resident scenario at 10,000 years	1.7×10^{8}[f]	1.3×10^{7}
Pu-239	Resident scenario at 10,000 years	2.3×10^{11}[f]	1.8×10^{10}

Table is continued on following page.

Table 4-11. (continued)

Nuclide	Limiting condition[b]	Concentration limit ($\mu Ci/m^3$)[c]	Inventory limit (Ci per vault)[d]
Pu-240	Resident scenario at 10,000 years	No limit[e,f]	No limit
Pu-241	Resident scenario at 10,000 years	1.0×10^{11}[f]	7.9×10^{9}
Pu-242	Resident scenario at 10,000 years	No limit[e,f]	No limit
Pu-244	Resident scenario at 1,000 years	5.0×10^{4}	4.0×10^{3}
Am-241	Resident scenario at 10,000 years	3.2×10^{9}[f]	2.5×10^{8}
Am-242m	Resident scenario at 10,000 years	1.3×10^{8}[f]	1.0×10^{7}
Am-243	Resident scenario at 1,000 years	3.4×10^{6}[f]	2.7×10^{5}
Cm-242	Resident scenario at 10,000 years	3.3×10^{10}[f]	2.6×10^{9}
Cm-243	Resident scenario at 100 years	1.6×10^{14}[f]	1.2×10^{13}
Cm-244	Resident scenario at 10,000 years	No limit[e,f]	No limit
Cm-245	Resident scenario at 1,000 years	8.4×10^{7}[f]	6.6×10^{6}
Cm-246	Resident scenario at 10,000 years	No limit[e,f]	No limit
Cm-247	Resident scenario at 10,000 years	2.4×10^{5}[f]	1.9×10^{4}
Cm-248	Resident scenario at 10,000 years	5.8×10^{8}[f]	4.6×10^{7}
Bk-249	Resident scenario at 1,000 years	7.3×10^{8}[f]	5.8×10^{7}
Cf-249	Resident scenario at 1,000 years	1.9×10^{6}[f]	1.5×10^{5}
Cf-250	Resident scenario at 10,000 years	No limit[e,f]	No limit
Cf-251	Resident scenario at 1,000 years	2.6×10^{7}[f]	2.0×10^{6}
Cf-252	Resident scenario at 10,000 years	7.7×10^{13}[f]	6.1×10^{12}

Footnotes to table are given on following page.

Footnotes to Table 4-11

[a] Limits on allowable disposals are most restrictive of limits given in Tables 4-8, 4-9, and 4-10; assumed time frame is discussed in Sect. 4.4.5. Sum-of-fractions rule applies to waste containing mixtures of radionuclides.

[b] When there is no limiting scenario, radionuclide and its decay products, if any, are not significant photon emitters.

[c] Limits on allowable average concentrations of radionuclides in saltstone at time of disposal.

[d] Limits on allowable inventories of radionuclides in saltstone at time of disposal for Vaults 2-15 (see Fig. 3-3); limits on allowable inventories in Vault 1 are half the given values.

[e] Calculated concentration limit exceeds specific activity of isotope.

[f] Calculated concentration limit or corresponding maximum concentration of alpha-emitting transuranic decay product exceeds value of 100 nCi/g for alpha-emitting transuranic radionuclides with half-lives greater than 20 years used to define transuranic waste (see Sect. 4.4.5).

Table 4-12. Limits on allowable average concentrations and inventories of radionuclides per vault in SDF based on resident scenarios and assumption of 1,000-year time frame for assessments of inadvertent intrusion[a]

Nuclide	Limiting scenario[b]	Concentration limit ($\mu Ci/m^3$)[c]	Inventory limit (Ci per vault)[d]
H-3	None	No limit	No limit
Be-10	None	No limit	No limit
C-14	None	No limit	No limit
Al-26	Resident scenario at 1,000 years	2.0×10^3	1.6×10^2
Co-60	Resident scenario at 100 years	3.8×10^{12}	3.0×10^{11}
Ni-59	None	No limit	No limit
Ni-63	None	No limit	No limit
Se-79	None	No limit	No limit
Sr-90	None	No limit	No limit
Zr-93	None	No limit	No limit
Nb-93m	None	No limit	No limit
Nb-94	Resident scenario at 1,000 years	1.1×10^4	8.5×10^2
Tc-99	None	No limit	No limit
Pd-107	None	No limit	No limit
Cd-113m	None	No limit	No limit
Sn-121m	None	No limit	No limit
Sn-126	Resident scenario at 1,000 years	1.3×10^4	1.0×10^3
I-129	None	No limit	No limit
Cs-135	None	No limit	No limit
Cs-137	Resident scenario at 100 years	1.3×10^{10}	1.0×10^9

Table is continued on following page.

Table 4-12. (continued)

Nuclide	Limiting condition[b]	Concentration limit ($\mu Ci/m^3$)[c]	Inventory limit (Ci per vault)[d]
Sm-151	None	No limit	No limit
Eu-152	Resident scenario at 100 years	4.2×10^9	3.3×10^8
Eu-154	Resident scenario at 100 years	9.5×10^{10}	7.5×10^9
Eu-155	Resident scenario at 100 years	No limit[e]	No limit
Pb-210	None	No limit	No limit
Ra-226	Resident scenario at 1,000 years	6.2×10^3	4.9×10^2
Ra-228	Resident scenario at 100 years	1.1×10^{11}	9.1×10^9
Ac-227	Resident scenario at 100 years	1.5×10^{11}	1.2×10^{10}
Th-229	Resident scenario at 1,000 years	1.1×10^5	9.0×10^3
Th-230	Resident scenario at 1,000 years	1.1×10^4	9.0×10^2
Th-232	Resident scenario at 1,000 years	2.0×10^3	1.6×10^2
Pa-231	Resident scenario at 1,000 years	2.6×10^5	2.0×10^4
U-232	Resident scenario at 100 years	1.9×10^6	1.5×10^5
U-233	Resident scenario at 1,000 years	1.2×10^6	9.1×10^4
U-234	Resident scenario at 1,000 years	2.4×10^6	1.9×10^5
U-235	Resident scenario at 1,000 years	5.6×10^6	4.4×10^5
U-236	Resident scenario at 1,000 years	No limit[e]	No limit
U-238	Resident scenario at 1,000 years	8.3×10^5	6.6×10^4
Np-237	Resident scenario at 1,000 years	7.1×10^5[f]	5.6×10^4
Pu-238	Resident scenario at 1,000 years	8.2×10^9[f]	6.4×10^8
Pu-239	Resident scenario at 1,000 years	No limit[e,f]	No limit

Table is continued on following page.

Table 4-12. (continued)

Nuclide	Limiting condition[b]	Concentration limit ($\mu Ci/m^3$)[c]	Inventory limit (Ci per vault)[d]
Pu-240	Resident scenario at 1,000 years	No limit[e,f]	No limit
Pu-241	Resident scenario at 1,000 years	1.3×10^{11}[f]	1.1×10^{10}
Pu-242	Resident scenario at 1,000 years	No limit[e,f]	No limit
Pu-244	Resident scenario at 1,000 years	5.0×10^{4}	4.0×10^{3}
Am-241	Resident scenario at 1,000 years	4.5×10^{9}[f]	3.5×10^{8}
Am-242m	Resident scenario at 1,000 years	5.3×10^{8}[f]	4.2×10^{7}
Am-243	Resident scenario at 1,000 years	3.4×10^{6}[f]	2.7×10^{5}
Cm-242	Resident scenario at 1,000 years	1.6×10^{12}[f]	1.3×10^{11}
Cm-243	Resident scenario at 100 years	1.6×10^{14}[f]	1.2×10^{13}
Cm-244	Resident scenario at 1,000 years	No limit[e,f]	No limit
Cm-245	Resident scenario at 1,000 years	8.4×10^{7}[f]	6.6×10^{6}
Cm-246	Resident scenario at 1,000 years	No limit[e,f]	No limit
Cm-247	Resident scenario at 1,000 years	2.5×10^{5}[f]	2.0×10^{4}
Cm-248	Resident scenario at 1,000 years	5.8×10^{9}[f]	4.6×10^{8}
Bk-249	Resident scenario at 1,000 years	7.3×10^{8}[f]	5.8×10^{7}
Cf-249	Resident scenario at 1,000 years	1.9×10^{6}[f]	1.5×10^{5}
Cf-250	Resident scenario at 1,000 years	No limit[e,f]	No limit
Cf-251	Resident scenario at 1,000 years	2.6×10^{7}[f]	2.0×10^{6}
Cf-252	Resident scenario at 1,000 years	7.6×10^{14}[f]	6.0×10^{13}

Footnotes to table are given on following page.

Footnotes to Table 4-12

[a] Limits on allowable disposals are more restrictive of limits given in Tables 4-8 and 4-9; assumed time frame is value specified by USDOE (1999a). Sum-of-fractions rule applies to waste containing mixtures of radionuclides.

[b] When there is no limiting scenario, radionuclide and its decay products, if any, are not significant photon emitters.

[c] Limits on allowable average concentrations of radionuclides in saltstone at time of disposal.

[d] Limits on allowable inventories of radionuclides in saltstone at time of disposal for Vaults 2-15 (see Fig. 3-3); limits on allowable inventories in Vault 1 are half the given values.

[e] Calculated concentration limit exceeds specific activity of isotope.

[f] Calculated concentration limit or corresponding maximum concentration of alpha-emitting transuranic decay product exceeds value of 100 nCi/g for alpha-emitting transuranic radionuclides with half-lives greater than 20 years used to define transuranic waste (see Sect. 4.4.5).

For each transuranic radionuclide in Tables 4-11 and 4-12 except Pu-244, the limits on allowable average concentration in saltstone exceed the value of 100 nCi/g for alpha-emitting transuranic radionuclides with half-lives greater than 20 years used to define transuranic waste (USDOE, 1999b), or the corresponding maximum concentration of a longer-lived alpha-emitting transuranic decay product calculated from the half-lives of the parent and the decay product exceeds 100 nCi/g. Based on a bulk density of solidified saltstone of 1.7×10^3 kg/m^3 (see Sect. 3.3), the activity per unit mass of 100 nCi/g used to define transuranic waste corresponds to a concentration of 1.7×10^5 μCi/m^3. The maximum concentration of an alpha-emitting transuranic decay product is important for several transuranic radionuclides including Pu-241, Am-242m, Cm-242, Cm-244, Bk-249, Cf-250, and Cf-252; these radionuclides have half-lives less than 20 years or they are not alpha emitters. The importance of the definition of transuranic waste to development of waste acceptance criteria at the SDF is discussed in Sect. 8.10.

As emphasized previously, the sum-of-fractions rule should be applied in determining allowable disposals of waste containing mixtures of radionuclides. The easiest approach would be to apply the sum-of-fractions rule to all radionuclides without regard for when the limiting exposure scenario is assumed to occur. This approach should be conservative in ensuring compliance with the performance measure for chronic exposure of inadvertent intruders, because the resident scenarios assumed in this analysis do not occur at the same time and, thus, the calculated doses at the different times generally are not additive. Alternatively, the sum-of-fractions rule could be applied more rigorously by taking into account the time of occurrence of the most restrictive resident scenario for each radionuclide of concern, as well as the radionuclide half-lives and the change in assumed thickness of shielding over time. However, the alternative approach is more difficult to apply, and it is not expected that a straightforward application of the sum-of-fractions rule to all radionuclides, without taking into account differences in the times of occurrence of the assumed resident scenarios, would disqualify significant amounts of waste that would be acceptable if the sum-of-fractions rule were applied more rigorously.

4.4.6 Sensitivity of Disposal Limits to Time Frame for Assessment

In this analysis, limits on allowable disposals of radionuclides in the SDF are developed based on assumed time frames for assessments of inadvertent intrusion of 1,000 and 10,000 years. The disposal limits given in Tables 4-11 and 4-12 depend on the assumed time frame only in cases where a longer-lived radionuclide has photon-emitting decay products whose activities increase with time beyond 1,000 years. Radionuclides of this type include Th-230, long-lived isotopes of uranium, and many longer-lived transuranic radionuclides. When disposal limits based on the analysis at 10,000 years are more restrictive than limits based on the analysis at 1,000 years, a comparison of results in the two tables indicates that the disposal limits are relatively insensitive to the assumed time frame for some radionuclides (the limits differ by less than a factor of two) but are substantially more sensitive to the assumed time frame for other radionuclides (the limits differ by more than an order of magnitude).

However, there are two additional considerations which indicate that the sensitivity of disposal limits to the assumed time frame for assessments of inadvertent intrusion is

unimportant. First, for all radionuclides for which disposal limits can depend on the assumed time frame, the more restrictive concentration limits given in Table 4-11 are at least three orders of magnitude higher than expected concentrations in saltstone (see Sect. 3.3 and Table 3-1). Second, for all transuranic radionuclides of this type, the more restrictive concentration limits exceed the value of 100 nCi/g used to define transuranic waste (see Sect. 8.10).

Therefore, as a practical matter, decisions about the acceptability of waste for disposal in the SDF based on assessments of inadvertent intrusion should not depend on whether the time frame for the assessments is assumed to be 1,000 or 10,000 years. This conclusion is a consequence of the design of the engineered barriers and cover system above the waste and their assumed performance over 10,000 years.

4.4.7 Sensitivity and Uncertainty Analysis of Model to Estimate Dose

Current USDOE guidance on performance assessment (USDOE, 1996) indicates that a sensitivity and uncertainty analysis of the assumed exposure scenarios for inadvertent intrusion should be limited to qualitative arguments including, for example, discussions of the rationale for selecting particular scenarios and parameter values. These issues have been addressed in Sects. 4.2 and 4.3. The guidance on sensitivity and uncertainty analysis of intrusion scenarios is based on an assumption that active institutional control will be maintained over disposal sites until they can be safely released in accordance with requirements in Order 5400.5 (USDOE, 1990) or 10 CFR Part 834 (USDOE, 1993) when it is promulgated.

The following discussion of uncertainties in the model used to estimate external dose to inadvertent intruders in the resident scenarios goes beyond the requirements in current guidance (USDOE, 1996). The purpose of these discussions is to provide insight into the magnitude of uncertainties in estimates of annual effective dose equivalents per unit concentration of radionuclides in the assumed resident scenarios, i.e., the SDCFs in Table 4-7. Because a simple linear model is used to estimate external dose in these scenarios, estimates of uncertainty in the parameter values essentially provide estimates of uncertainty in the SDCFs.

Limits on allowable disposals in Tables 4-11 and 4-12 are based on estimates of dose for the pathway involving external exposure during indoor residence in a home located on top of a disposal vault. As indicated by eq. (4.1) in Sect. 4.3.2, the external dose during indoor residence per unit concentration of a radionuclide in a disposal vault is proportional to the exposure time, the shielding factor during indoor residence, and the external dose coefficient. For purposes of radiation protection, external dose coefficients are treated as fixed parameters for reference individuals and reference conditions of exposure and, thus, essentially have no uncertainty. Even if uncertainties in external dose coefficients were treated rigorously, they probably are less than a factor of two for radionuclides that are high-energy photon emitters (Eckerman and Ryman, 1993; ICRP, 1996). The exposure time is uncertain, but the value of 50% of the time during the year assumed in this analysis is intended to be conservative for most individuals and would not underestimate the exposure time in the worst case by more than a factor of two. The shielding factor during indoor residence also is uncertain, but the assumed value of 0.7 would not underestimate the actual

value by more than 50%, and it probably does not overestimate values for typical homes in the vicinity of the SRS by more than a factor of two (Kocher, 1980). Substantially lower shielding factors and, thus, lower external doses would result if a home were assumed to be constructed with a thick foundation of uncontaminated material, such as concrete (see Sect. A.2.3). However, it does not seem reasonable to assume that such a foundation would be used in any home that would be constructed at the disposal site.

Based on these considerations, uncertainties in the model used to assess external dose to inadvertent intruders should not be significant. Rather, the definition of exposure scenarios and an ability to obtain sufficiently accurate estimates of radionuclide inventories in waste prior to disposal are likely to be more important factors in determining acceptable disposals of radionuclides based on the results of an intruder dose analysis. These factors are discussed in the following section.

4.4.8 General Consideration of Uncertainties in Determining Acceptable Disposals

In evaluating uncertainties in models used to estimate dose to an inadvertent intruder, the most important consideration may be the definitions of the exposure scenarios. Dose assessments for the different scenarios are based on an assumption that the scenarios will occur as postulated, but many of the assumptions used in defining the scenarios are likely to be pessimistic.

In defining exposure scenarios for inadvertent intruders at the SDF, it is reasonable to assume that individuals will establish a homestead within the site boundary at some time after loss of active institutional control. However, several of the assumptions used in developing the exposure scenarios are less certain and probably pessimistic. For example, all scenarios for inadvertent intrusion assume that individuals will have no prior knowledge of waste disposal activities at the site, but this assumption seems unreasonable at times soon after loss of active institutional control. Furthermore, all exposure scenarios assume that an intruder will build a home at the location of disposed waste, but there is some probability that all homes constructed on the SRS at future times will be located elsewhere.

An assumption that active institutional control will be relinquished at 100 years after disposal also may be pessimistic. Controls over waste disposal or other contaminated sites will not be relinquished until such sites can be released safely (USDOE, 1996), and conditions that define safe release may be more restrictive that the performance measure for inadvertent intruders assumed in this analysis (Luftig and Weinstock, 1997). Thus, active institutional control may be required for considerably longer than 100 years to provide adequate protection of the public, and some assumed scenarios, such as the resident scenario at 100 years, may be precluded.

By the way they are defined, assumed exposure scenarios for inadvertent intruders tend to maximize estimates of dose that reasonably could be experienced by individuals who might come onto the disposal site after loss of active institutional control. It is important to recognize that a dose assessment for inadvertent intruders is not intended to provide best estimates of doses that likely would be received and a quantification of uncertainties in estimated doses. Rather, the primary purpose of a dose assessment is to indicate whether

planned disposal practices at a site would be adequately protective of future inadvertent intruders. This is accomplished mainly by using the results of a dose assessment to establish waste acceptance criteria in the form of limits on average concentrations or inventories of radionuclides in waste prior to disposal. If these limits are not exceeded, it may be assumed that future inadvertent intruders will be protected, and there is no need to be concerned about the magnitude of doses that any such individuals might actually receive. Furthermore, quantitative estimates of uncertainties in calculated doses based on parameter uncertainty analysis may not be particularly meaningful, because the results are conditional on the occurrence of assumed exposure scenarios. Therefore, an important factor in determining the acceptability of waste disposals is the credibility of assumed exposure scenarios for inadvertent intruders, rather than any estimates of uncertainties in the results due to uncertainties in parameters in the dose assessment models.

A second important factor in determining the acceptability of waste disposals is the capability of estimating inventories of radionuclides in disposed waste with sufficient accuracy. In a PA for a disposal facility at another site (ORNL, 1997), uncertainties in estimating inventories of radionuclides in waste were judged to be an important, if not the most important, source of uncertainty in assessing long-term performance, even when all sources of uncertainty in the various models and parameters were considered. It usually is not difficult to estimate the inventories of high-energy photon-emitting radionuclides, such as Co-60 and Cs-137. However, it can be a major challenge to estimate inventories of beta- and alpha-emitting radionuclides based on measurement, rather than process knowledge. Furthermore, depending on the scenarios for inadvertent intrusion that are assumed to be credible occurrences, inventories of some of these radionuclides, such as Tc-99, uranium, and alpha-emitting transuranic radionuclides (e.g., Pu-239, Am-241) may be comparable to or a substantial fraction of the limits on allowable inventories, in which case there is a clear need of accurate estimates of inventories. Therefore, the problem of determining inventories of important radionuclides in waste may override any considerations of uncertainties in the models used to assess dose to inadvertent intruders.

5. ANALYSIS OF GROUNDWATER PATHWAY

As shown in Table 2-1, the USDOE requirements for low-level waste disposal contain a performance objective based on the exposure from all pathways and a requirement to assess impacts on water resources. This section presents an assessment of potential dose to a future member of the general public from the groundwater pathway and potential impacts on water resources. Taking into account the current design of the facility, the results of this assessment are used to derive limits on the total inventory of radionuclides in waste at the time of disposal.

5.1 RADIONUCLIDES CONSIDERED IN DOSE ANALYSIS

Low-level radioactive waste that may be sent to the SDF contains many radionuclides. However, the number of radionuclides that need to be included in this groundwater analysis can be reduced substantially based on considerations of radionuclide half-lives and the processes by which low-level waste at the SDF is generated. The radionuclides considered in this analysis were selected on the basis of screening analyses for the groundwater pathway in the ELLWF PA (Table 4.1-2 of McDowell-Boyer et al., 2000) and on their expected presence in salt waste (Table 2.6-2 of MMES et al., 1992 and Table 3-1 of this study). Sixty radionuclides were selected for the analysis as shown in Table 5-1.

5.2 SCENARIOS

For protection of the public and the assessment of impacts to water resources, exposure pathways involving direct ingestion of groundwater are the pathways of dominant concern (McDowell-Boyer et al., 2000). The scenario analyzed in this study assumes water is drawn from a well located at the point of highest concentration in groundwater at least 100 meters from the disposed waste. The concentration at this point is compared with either the MCL or the concentration that would result in an effective dose equivalent of 25 mrem/year (assuming consumption of two liters of water per day for a year), whichever is lower.

5.3 COMPUTER MODELING

The PATHRAE computer program (RAE, 1998) was used to extend the analysis reported in the Z-Area Performance Assessment (MMES et al., 1992).

5.3.1 Selection of Computer Program

The PATHRAE computer program was selected for this study because it has been extensively tested and because it has been widely applied at SRS to a variety of waste management and environmental restoration assessments. The PATHRAE code was developed for the EPA to assess disposal of low-level radioactive waste at shallow-land disposal facilities (Merrell et al., 1986). The PATHRAE code was checked to ensure that it

Table 5-1. Radionuclides considered in groundwater impacts assessment

Nuclide	Nuclide	Nuclide
H-3	Cs-135	Np-237
C-14	Cs-137	Pu-238
Co-60	Ce-144	Pu-239
Ni-59	Pr-144	Pu-240
Ni-63	Pm-147	Pu-241
Se-79	Sm-151	Pu-242
Sr-90	Eu-152	Pu-244
Y-90	Eu-154	Am-241
Zr-93	Eu-155	Am-242
Tc-99	Th-228	Am-242m
Ru-106	Th-231	Am-243
Pd-107	Th-232	Cm-242
Ag-110m	Th-234	Cm-243
Sn-121m	Pa-234	Cm-244
Sn-126	U-232	Cm-245
Sb-125	U-233	Cm-246
Sb-126	U-234	Cm-247
Te-125m	U-235	Cm-248
I-129	U-236	Cf-249
Cs-134	U-238	Cf-252

fundamentally solves the groundwater flow and transport equations properly (i.e., PATHRAE gives the same results for test problems as a manual solution) (Looney et al., 1987). PATHRAE was verified according to the WSRC E-7 manual (Shuman et al., 1998a). PATHRAE was also benchmarked against a number of other analytical and numerical codes (Shuman et al., 1986, Wood et al., 1994, and Shuman et al., 1998b).

5.3.2 Benchmarking

To ensure that the PATHRAE computer program would give results comparable to the PORFLOW (Runchal and Sagar, 1992) numerical program used in the PA, PATHRAE was benchmarked against the performance assessment. This means determining if the PATHRAE program could provide results similar to those reported in the PA using the same input parameters plus values for additional parameters that were reasonable. Both the intact and degraded cases from the PA were used. The intact case assumes that the saltstone and vault have not cracked (i.e., lost their ability to delay the flow of water through them). In the degraded case, the saltstone and vault are assumed to have cracked completely through, which results in a much greater flow of water through the saltstone and vault. The physical state of the vault and saltstone in the groundwater degraded case is much different than the failed case in the intruder analysis. Cracking of the vault and saltstone will result in hydrologic failure (i.e., will not delay water flow through them) but would not result in failure in the sense used in the intruder analysis in Sect. 4. Failure in the intruder analysis means that the saltstone, grout, or vault roof have lost ability to prevent excavating into or through them.

5.3.2.1 Intact Case. The intact case (i.e., uncracked vault and saltstone) was simulated using the PATHRAE code. Input parameters are shown in Table 5-2. The radionuclide inventories from Table 2.6-2 in the saltstone PA (MMES et al., 1992) were used. Results are presented in Table 5-3. The PATHRAE results provide a reasonably conservative representation of the PA results. The time of the peak concentration in the PATHRAE results is within about 14% of the time in the PA results. The peak concentration in the PATHRAE results is greater (i.e., more conservative) than the PA result.

5.3.2.2 Degraded Case. The degraded case (i.e., cracked vault and saltstone) was also simulated using the PATHRAE code. The radionuclide inventories from Table 2.6-2 in the saltstone PA (MMES et al. 1992) were used. The input parameters are shown in Table 5-4. The results are presented in Table 5-5. For the radionuclides that gave a PATHRAE peak value greater than 1×10^{-21} pCi/L, the PATHRAE peak concentrations are greater than the corresponding PA concentrations (i.e., the PATHRAE peak concentrations are conservative versus the PA peak concentrations), with the exception of ^{129}I. The PATHRAE peak times are within about 40% of the corresponding PA peak times.

5.4 DEVELOPMENT OF GROUNDWATER INVENTORY LIMITS

The PATHRAE results for the degraded case, shown in Table 5-6, confirm, with one exception, the groundwater screening analysis in the original PA (Sect. 3.2.3.4 of MMES et al., 1992). Other than tritium, ^{14}C, ^{79}Se, ^{99}Tc, ^{126}Sn, and ^{129}I (i.e., the radionuclides for which groundwater concentrations were calculated in the PA), only ^{237}Np peaks at a time near 10,000 years. Neptunium-237 peaks at 25,000 years at a value of 2.8×10^{-2} pCi/L per curie.

Since the PA screening analysis is confirmed, the peak groundwater concentrations from the PA or PA Addendum (WSRC, 1998), as appropriate, should be used to establish saltstone disposal limits for the groundwater pathway, until the PA can be revised. Since the PATHRAE results show that ^{237}Np should not be screened out, the PATHRAE result for ^{237}Np should be used to establish an interim inventory limit for saltstone disposal, until the PA can be revised. For all of these limits, the radionuclide inventory is assumed to be distributed uniformly over the disposed waste.

The drinking water standards (40 CFR 141) (i.e., MCLs) are the appropriate measure for groundwater protection (Cook, 2002). The inventory limit for each radionuclide is calculated by dividing the MCL by the peak groundwater concentration, expressed in units of pCi/L/Ci. The limits are shown in Table 5-7. For tritium, the peak concentration value for the intact case in the PA was conservatively used because the degraded case peak value was $<10^{-12}$ pCi/L. For ^{237}Np, values are shown both for the PATHRAE peak concentration and for the concentration at 10,000 years. The limit based on the concentration at 10,000 years should be used to be consistent with the SRS DAS (Fiori and Frei, 1999).

To provide a comparison of limits derived from concentrations within 1,000 years (i.e., the time of compliance required by USDOE Order 435.1) with those derived from concentrations within 10,000 years (i.e., those consistent with the SRS DAS), PATHRAE was used to determine the maximum concentration up to 1,000 years and the maximum concentration up to 10,000 years for all radionuclides listed in Table 5-6 with peak concentration greater than 1×10^{-21} pCi/L/Ci. The results are shown in Table 5-8. Limiting the time of compliance to 1,000 years results in no radionuclide limits being required for the groundwater pathway. Using a 10,000-year time of compliance results in limits being required for ^{14}C, ^{99}Tc, ^{129}I, and ^{237}Np.

Table 5-2. Saltstone intact case benchmarking input parameters

Property	Value	Source
Length of Facility	650 m	Map Measurement
Width of Facility	1000 m	Map Measurement
Density of Aquifer	1600 kg/m^3	Z-Area PA, page 3-77
Longitudinal Dispersivity	3 m	Z-Area PA, page A-40
Transverse Dispersion	0	Assumption
Vertical Dispersion	0	Assumption
Residual Saturation	0.7	Z-Area PA, page C-12
Sat. Conductivity of Vertical Zone	3.2 m/y (1 x 10^{-5} cm/s)	Z-Area PA, Table 3.3-1
No. of Mesh Points	20	PATHRAE Suggestion
Cover Thickness	3.6 m	Z-Area PA, pages 2-61, 62, 71
Waste Thickness	7.3 m	Z-Area PA, page 2-61
Waste Volume	1.14 x 10^6 m^3	Z-Area PA, page 2-58 (30x30x7.3x174)
Effective Diffusion Length in Waste	7.3 m	Saltstone thickness, Z-Area PA page 2-61
Effective Diffusion Length in Vault Wall	0.76	Vault thickness, Z-Area PA, page 2-61
Effective Diffusion Coefficient Waste	5 x 10^{-9} cm^2/sec / R_d	Z-Area PA, page A-34
Effective Diffusion Coefficient Vault	1 x 10^{-8} cm^2/sec / R_d	Z-Area PA, page A-34
Distance to Well	100 m	USDOE Order 435.1
Well Distance Off Centerline	0 m	Assumption
Density of Waste	1700 kg/m^3	Z-Area PA, page 2-56
Water Infiltration to Waste	1.75 x 10^{-3} m/y	Z-Area PA, Table 4.1-1
Horizontal Velocity of Aquifer	1.2 m/y	Calibrated vs. NO_3 arrival time
Porosity of Aquifer	0.40	Z-Area PA, Table 3.3-3
Distance from waste to Aquifer	6.1 m (20 ft.)	Z-Area PA, Fig. A.1-9
Vertical velocity in Unsaturated Zone	9.8 x 10^{-4} m/y	Calibrated vs. NO_3 arrival time
Well Screen Length	10 m	Thickness of top node
Porosity of Unsaturated Zone	0.44	Z-Area PA, Table 3.3-1
Bulk Density of Soil	1.6 g/m^3	Z-Area PA, page. 3-77

Table 5-3. Results of benchmarking of intact case

	PA Results		PATHRAE Results	
Radionuclide	Peak Concentration, pCi/L	Time of Peak, y	Peak Concentration, pCi/L	Time of Peak, y
Se-79	1.2×10^{-2}	2.1×10^{5}	3.5×10^{-1}	2.0×10^{5}
Tc-99	6.7×10^{-7}	1.6×10^{6}	6.4×10^{-7}	1.4×10^{6}
Sn-126	4.0×10^{-11}	9.2×10^{5}	3.0×10^{-10}	7.9×10^{5}

Table 5-4. Saltstone degraded case benchmarking input parameters

Property	Value	Source
Length of Facility	650 m	Map Measurement
Width of Facility	1000 m	Map Measurement
Density of Aquifer	1600 kg/m^3	Z-Area PA, page 3-77
Longitudinal Dispersivity	3 m	Z-Area PA, page A-40
Transverse Dispersion	0	Assumption
Vertical Dispersion	0	Assumption
Residual Saturation	0.7	Z-Area PA, page C-12
Sat. Conductivity of Vertical Zone	3.2 m/y (1 x 10^{-5} cm/s)	Z-Area PA, Table 3.3-1
No. of Mesh Points	20	PATHRAE Suggestion
Cover Thickness	3.6 m	Z-Area PA, pages 2-61, 62, 71
Waste Thickness	7.3 m	Z-Area PA, page 2-61
Waste Volume	1.14 x 10^6 m^3	Z-Area PA, page 2-58 (30x30x7.3x174)
Effective Diffusion Length in Waste	1.5 m	½ distance between cracks
Effective Diffusion Length in Vault Wall	0.76	Vault thickness
Effective Diffusion Coefficient Waste	5 x 10^{-9} cm^2/sec / R_d	Z-Area PA, page A-34
Effective Diffusion Coefficient Vault	1 x 10^{-8} cm^2/sec	Assumption – No vault in PA degraded model
Distance to Well	100 m	USDOE Order 435.1
Well Distance Off Centerline	0 m	Assumption
Density of Waste	1700 kg/m^3	Z-Area PA, page. 2-56
Water Infiltration to Waste	1.75 x 10^{-3} m/y	Z-Area PA, Table 4.1-1
Horizontal Velocity of Aquifer	1.2 m/y	Calibrated vs. NO_3 arrival time
Porosity of Aquifer	0.40	Z-Area PA, Table 3.3-3
Distance from waste to Aquifer	6.1 m	Z-Area PA, Fig. A.1-9
Vertical velocity in Unsaturated Zone	1.8 x 10^{-2} m/y	Calibrated vs. NO_3 arrival time
Well Screen Length	10 m	Thickness of top node
Porosity of Unsaturated Zone	0.44	Z-Area PA, Table 3.3-1
Bulk Density of Soil	1.6 g/m^3	Z-Area PA, page 3-77

Table 5-5. Results of benchmarking of degraded case

	PA Results		PATHRAE Results	
Radionuclide	Peak Concentration, pCi/L	Time of Peak, y	Peak Concentration, pCi/L	Time of Peak, y
H-3	$<10^{-12}$		3.6×10^{-21}	4.3×10^{2}
C-14	7.8×10^{-5} (PA*13)[a]	7.3×10^{3}	2.1×10^{-3}	1.0×10^{4}
Se-79	5.7×10^{1}	1.5×10^{4}	1.5×10^{2}	2.3×10^{4}
Sr-90	$<10^{-12}$		$< 10^{-21}$	
Tc-99	1.5×10^{2}	2.4×10^{3}	2.5×10^{2}	2.8×10^{3}
Sn-126	2.9×10^{-2}	9.2×10^{5}	6.8×10^{-1}	3.4×10^{5}
I-129	9.9×10^{-1}	3.2×10^{3}	3.6×10^{-1}	4.3×10^{3}
Cs-137	$<10^{-12}$		$< 10^{-21}$	
Pu-238	$<10^{-12}$		$< 10^{-21}$	
Am-241	$<10^{-12}$		$< 10^{-21}$	

[a] Per WSRC, 1998.

Table 5-6. PATHRAE degraded case results

Nuclide	Peak Concentration, PCi/L/Ci	Peak Time, Years	Nuclide	Peak Concentration, PCi/L/Ci	Peak Time, Years
H-3	1.9×10^{-25}	4.3×10^{2}	Th-231	$<1 \times 10^{-21}$	
C-14	3.2×10^{-4}	9.2×10^{3}	Th-232	$<1 \times 10^{-21}$	
Co-60	$<1 \times 10^{-21}$		Th-234	$<1 \times 10^{-21}$	
Ni-59	$<1 \times 10^{-21}$		Pa-234	$<1 \times 10^{-21}$	
Ni-63	$<1 \times 10^{-21}$		U-232	$<1 \times 10^{-21}$	
Se-79	4.7×10^{-1}	1.8×10^{5}	U-233	$<1 \times 10^{-21}$	
Sr-90	$<1 \times 10^{-21}$		U-234	$<1 \times 10^{-21}$	
Y-90	$<1 \times 10^{-21}$		U-235	$<1 \times 10^{-21}$	
Zr-93	$<1 \times 10^{-21}$		U-236	$<1 \times 10^{-21}$	
Tc-99	3.8×10^{-3}	2.8×10^{3}	U-238	$<1 \times 10^{-21}$	
Ru-106	$<1 \times 10^{-21}$		Np-237	2.8×10^{-2}	2.5×10^{4}
Pd-107	$<1 \times 10^{-21}$		Pu-238	$<1 \times 10^{-21}$	
Ag-110m	$<1 \times 10^{-21}$		Pu-239	$<1 \times 10^{-21}$	
Sn-121m	$<1 \times 10^{-21}$		Pu-240	$<1 \times 10^{-21}$	
Sn-126	5.2×10^{-3}	3.5×10^{5}	Pu-241	$<1 \times 10^{-21}$	
Sb-125	$<1 \times 10^{-21}$		Pu-242	$<1 \times 10^{-21}$	
Sb-126	$<1 \times 10^{-21}$		Pu-244	$<1 \times 10^{-21}$	
Te-125m	$<1 \times 10^{-21}$		Am-241	$<1 \times 10^{-21}$	
I-129	1.8×10^{-2}	4.3×10^{3}	Am-242	$<1 \times 10^{-21}$	
Cs-134	$<1 \times 10^{-21}$		Am-242m	$<1 \times 10^{-21}$	
Cs-135	$<1 \times 10^{-21}$		Am-243	$<1 \times 10^{-21}$	
Cs-137	$<1 \times 10^{-21}$		Cm-242	$<1 \times 10^{-21}$	
Ce-144	$<1 \times 10^{-21}$		Cm-243	$<1 \times 10^{-21}$	
Pr-144	$<1 \times 10^{-21}$		Cm-244	$<1 \times 10^{-21}$	
Pm-147	$<1 \times 10^{-21}$		Cm-245	$<1 \times 10^{-21}$	
Sm-151	$<1 \times 10^{-21}$		Cm-246	$<1 \times 10^{-21}$	
Eu-152	$<1 \times 10^{-21}$		Cm-247	$<1 \times 10^{-21}$	
Eu-154	$<1 \times 10^{-21}$		Cm-248	$<1 \times 10^{-21}$	
Eu-155	$<1 \times 10^{-21}$		Cf-249	$<1 \times 10^{-21}$	
Th-228	$<1 \times 10^{-21}$		Cf-252	$<1 \times 10^{-21}$	

Table 5-7. Radionuclide limits for the groundwater pathway

Nuclide	Source of Peak	PA Peak pCi/L	PA Inventory Ci[a]	PA Peak pCi/L/Ci	PATHRAE Peak pCi/L/Ci	MCL pCi/L[b]	Inventory Limit Ci
H-3	PA	3.50×10^{-5}[c]	1.90×10^{4}	1.84×10^{-9}		2.00×10^{4}	1.09×10^{13}
C-14	PA*13	7.80×10^{-5}	6.50	1.20×10^{-5}		2.00×10^{3}	1.67×10^{8}
Se-79	Addendum	5.70×10^{1}	3.20×10^{2}	1.78×10^{-1}		7.00×10^{2}	3.93×10^{3}
Tc-99	Addendum	1.47×10^{2}	6.50×10^{4}	2.26×10^{-3}		9.00×10^{2}	3.98×10^{5}
Sn-126	Addendum	2.90×10^{-2}	1.30×10^{2}	2.23×10^{-4}		3.00×10^{2}	1.34×10^{6}
I-129	Addendum	9.90×10^{-1}	2.00×10^{1}	4.95×10^{-2}		1.00	2.02×10^{1}
Np-237	PATHRAE				2.8×10^{-2}[d]	8.90	4.94×10^{2}[d]
Np-237	PATHRAE				8.7×10^{-5}[e]	8.90	1.02×10^{5}[e]

[a] From Table 2.6-2 of MMES et al.,1992.

[b] From Table 4.3-1 of Cook, 2002.

[c] From Table 4.1-5 of MMES et al., 1992.

[d] Based on peak concentration (i.e., at 2.5×10^{4} years).

[e] Based on concentration at 10,000 years.

Table 5-8. Comparison of disposal limits derived from PATHRAE results at 1,000 years and at 10,000 years

Nuclide	Maximum concentration up to 1000 years, pCi/L-Ci	Limit based on maximum concentration up to 1,000 years, Ci	Maximum concentration up to 10,000 years, pCi/L-Ci	Limit based on maximum concentration up to 10,000 years, Ci
H-3	0	No Limit	0	No Limit
C-14	0	No Limit	4.1×10^{-4}	4.9×10^{6}
Ni-59	0	No Limit	0	No Limit
Se-79	0	No Limit	0	No Limit
Tc-99	0	No Limit	3.8×10^{-3}	2.3×10^{5}
Pd-107	0	No Limit	0	No Limit
Sn-126	0	No Limit	0	No Limit
I-129	0	No Limit	1.8×10^{-2}	5.6×10^{1}
Np-237	0	No Limit	8.7×10^{-5}	1.0×10^{5}
Pu-239	0	No Limit	0	No Limit
Pu-240	0	No Limit	0	No Limit
Pu-242	0	No Limit	0	No Limit
Pu-244	0	No Limit	0	No Limit

6. ANALYSIS OF AIR PATHWAY

As shown in Table 2-1, the USDOE requirements for low-level waste disposal contain a performance objective based on the release of radionuclides to the atmosphere. This section presents an assessment of potential dose to a future member of the general public from the air pathway. Taking into account the current design of the facility, the results of the dose assessment are used to derive limits on the total inventory of radionuclides in waste at the time of disposal.

6.1 RADIONUCLIDES CONSIDERED IN DOSE ANALYSIS

Earlier screening analyses (Cook and Wilhite, 1998) show that only tritium and ^{14}C could produce doses of concern via the air pathway.

6.2 DOSE ANALYSIS

For the saltstone vaults, the analysis of the ^{3}H and ^{14}C source term to the atmosphere considers the partitioning of ^{3}H and ^{14}C between the volatile water vapor phase, and pore water in the vaults, the nonvolatile phase.

Since ^{3}H in the vapor form is derived from the concentrations in the pore water, the ratio of the concentration in the vapor form to that in the water phase can be determined as follows:

$$H = \frac{C_{vapor}\left(g / m^3 of\ air\right)}{C_{water}\left(g / m^3 of\ water\right)}$$

The concentration of water in air at 10 ^{o}C and 100 percent relative humidity can be determined as follows:

$$C\ (g/m^3) = \frac{Avp}{R\,x\,T}\,x\,MW$$

where

Avp	=	actual vapor pressure (0.012 bars),
R	=	gas constant (8.314×10^{-5} bar m^3 mol^{-1} $^{0}K^{-1}$),
T	=	temperature (283 ^{0}K),
MW	=	molecular weight (18 g mol^{-1}).

This results in a water concentration in the air of 9.2 g m^{-3}. Assuming that concentration of ^{3}H in water is equal to the concentration in the vapor:

$$\frac{1Ci}{m^3_{(H2O)}}\,x\,\frac{1m^3_{(H2O)}}{10^6\,g_{(H2O)}}\,x\,\frac{9.2\,g_{(H2O)}}{m^3_{(air)}} = \frac{9.2\,x\,10^{-6}\,Ci\ ^{3}H}{m^3_{(air)}}$$

Therefore,

$$H = \frac{9.2\ x\ 10^{-6}\ Ci / m^3_{(air)}}{1\ Ci / m^3_{(H2O)}}$$

Using this relationship, the ^3H concentration in the vapor phase can be determined from the concentration in the water phase in the vault.

Assuming that this is the ^3H concentration in the vapor phase at the top surface of the vault, then the flux at the ground surface is given by:

$$J = D\left(\frac{C_0}{x}\right) x\ L_{vault}\ x\ W_{vault}$$

where

J	=	annual flux at the soil surface, Ci y^{-1},
D	=	diffusion coefficient in air, 754 m^2 y^{-1}, (CRC, 1981),
C_O	=	source concentration in the vapor phase, Ci m^{-3},
x	=	cover thickness, m
L_{vault}	=	Length of vault, m
W_{vault}	=	Width of vault, m

For the saltstone vaults, using a volume of 8.2 x 10^4 m^3 (i.e., one vault), a porosity of 0.46 and an inventory of 1 Ci, the water concentration is 2.6 x 10^{-5} Ci/m^3. This gives a vapor concentration of 2.4 x 10^{-10} Ci/m^3. Using a cover thickness of 3.6 m, a tritium diffusion rate of 754 m^2/y (CRC, 1981), a length of 180m and a width of 60 m gives an annual release rate of 5.5 x 10^{-4} Ci/y.

A similar analysis for ^{14}C waste forms for the LAW and IL vaults shows that the annual release fraction is unity, because the partitioning between the water and vapor phases cannot be considered.

Two points and times of exposure were used in the analysis. For the first 100 years, the minimum time per USDOE Order 435.1 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 Disposal Facility at 100 years after closure.

The dose limit used in the air pathway analysis is 10 mrem/y, effective dose equivalent, as required by 40 CFR 61.

The air pathway analysis results in the dose factors given in Table 6-1.

The disposal limit, in terms of curies per vault, can be calculated from the equation,

Limit (Ci/vault) = dose limit (mrem/year) / [dose factor (mrem/Ci) * release fraction (1/y-vault)]

Radionuclide limits for the air pathway for both the 100 meter and site boundary locations are shown in Table 6-2. The most restrictive of the air pathway limits (i.e., those at the 100 meter location) should be used in establishing Waste Acceptance Criteria. Since the calculations were performed at times no later than 100 years after closure, limits derived for the 1,000-year time frame will be the same as those derived for the 10,000-year time frame.

Table 6-1. Dose factors for the air pathway

Nuclide	100 m Location (mrem/Ci Released)	Site Boundary Location (mrem/Ci Released)
H-3	8.5×10^{-3}	2.4×10^{-6}
C-14	3.8	1.0×10^{-3}

Table 6-2. Inventory limits for the air pathway

Nuclide	100 m Location Ci/vault	Site Boundary Location Ci/vault
H-3	2.1×10^{6}	7.6×10^{9}
C-14	2.6	1.0×10^{4}

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7. ANALYSIS OF RADON EMANATION

As shown in Table 2-1, the USDOE requirements for low-level waste disposal contain a performance objective based on the release of radon from the waste. An analysis of the release of radon from disposed saltstone has not been conducted. However, the results of a radon release analysis from the E-Area Low-Level Waste Facility (ELLWF) PA (McDowell-Boyer et al., 2000) can be applied to saltstone. The saltstone vaults, when filled with saltstone and the uncontaminated grout above the saltstone, are essentially solid concrete monoliths. The ELLWF vaults are concrete structures filled with containers of waste. In the intermediate level vaults, the waste containers are surrounded by grout. Nonetheless, the ELLWF vaults have considerably more void space than the saltstone vaults. The additional void space enhances the vapor phase migration of contaminants. Thus, the ELLWF radon analysis is a conservative (i.e., over predicts impacts from radon) representation of the saltstone vaults.

In Sect. 4.3.4 of the ELLWF PA (McDowell-Boyer et al., 2000), an analysis was made of the release of radon from ^{234}U and its migration to the surface of the disposal facility. The analysis concludes that a ^{234}U concentration in the vault-disposed waste of 2.6×10^{-3} Ci/m^3 will produce a radon flux of 20 pCi/m^2-sec, which is the limit on radon emanation. The volume of saltstone in a typical vault is 78,840 cubic meters (MMES et al., 1992). Therefore the ^{234}U limit in saltstone, based on radon emanation, is 2.05×10^2 Curies per vault. This limit was derived for the 10,000-year time frame. No analysis was performed for the 1,000-year time frame.

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8. CONCLUSIONS

In this section, the radionuclide disposal limits for the intruder, groundwater, air, and radon emanation pathways are converted into common units (i.e., Ci per liter of salt solution feed to saltstone) to establish, for each radionuclide, the lowest (i.e., most restrictive) limit. The lowest limits are then compared with the estimated average composition of low curie salt to assess the feasibility of low curie salt disposal. Limits assuming a time of assessment of 1,000 years are also developed for comparison with those developed assuming a time of assessment of 10,000 years (i.e., those consistent with the SRS DAS). The limits developed by assuming a 10,000-year time of assessment are converted into units consistent with saltstone waste acceptance criteria (i.e., pCi/mL of salt solution feed to saltstone).

8.1 INADVERTENT INTRUDER ANALYSIS

The radionuclide disposal limits derived from the inadvertent intruder analysis for the two assumed times of assessment are listed in Tables 4-11 and 4-12. Limits are presented in units of $\mu Ci/m^3$ and curies per vault. The curies-per-vault limits are reproduced in Tables 8-1 and 8-2.

8.2 GROUNDWATER ANALYSIS

The radionuclide disposal limits derived from the groundwater pathway analysis for the two assumed times of assessment are listed in Tables 5-7 and 5-8. For the purpose of establishing interim saltstone disposal limits (i.e., until the saltstone PA is revised), limits derived from groundwater concentrations determined in the saltstone PA (MMES et al., 1992) (i.e., at the peak concentration, which for several radionuclides, occurs at greater than 10,000 years) are used rather than those derived from the PATHRAE code at 10,000 years, with the exception of limits for Np-237. The limits are presented in terms of the total curie inventory that can be disposed in the saltstone landfill (i.e., Ci per facility). These total curie limits are reproduced in Tables 8-1 and 8-2.

8.3 AIR ANALYSIS

The radionuclide disposal limits derived from the air pathway analysis are listed in Table 6-2. The air pathway analysis did not consider the time of assessment. Rather, the limits were derived from a simple and conservative model. The limits are presented in terms of curies per vault and are shown in Tables 8-1 and 8-2.

8.4 RADON ANALYSIS

The radionuclide disposal limit derived from the radon emanation pathway is presented in Sect. 7. This limit was derived at 10,000 years following disposal. The limit is stated in terms of curies per vault. This limit is shown in Table 8-1.

8.5 CONVERSION OF LIMIT UNITS

For the purpose of comparison of the disposal limits with the currently estimated average composition of low curie salt solution, and for the purpose of providing input to development of saltstone WAC, the limits must be converted into terms of curies per liter of salt solution feed to saltstone. The following assumptions are made:

1. A saltstone vault has 12 cells.
2. Each cell will contain saltstone produced from 4.2×10^6 liters of salt solution (MMES et al., 1992).
3. The entire saltstone landfill, as analyzed in the saltstone PA, will contain saltstone produced from 7.3×10^8 liters of salt solution (MMES et al., 1992).

The limits stated in terms of curies per vault are converted into curies per liter of salt solution by dividing by 5.04×10^7 liters (i.e., the volume of salt solution that will be disposed in a 12-cell vault). The limits stated in terms of total curies in the saltstone landfill are converted into curies per liter of salt solution by dividing by 7.3×10^8 liters.

8.6 LIMITS FOR A 10,000-YEAR TIME OF ASSESSMENT

The limits for a 10,000-year time of assessment (i.e., those consistent with the SRS DAS and that should be used to establish saltstone WAC until the saltstone PA is revised) are shown in Table 8-3 in terms of curies per liter of salt solution feed to saltstone.

8.7 LIMITS FOR A 1,000-YEAR TIME OF ASSESSMENT

The limits for a 1,000-year time of assessment are shown in Table 8-4 in terms of curies per liter of salt solution feed to saltstone.

8.8 MOST RESTRICTIVE LIMITS

Tables 8-5 and 8-6 show the most restrictive of the limits from Tables 8-3 and 8-4, respectively. For the 10,000-year time of assessment, 41 of the 75 radionuclides selected for analysis have disposal limits. Of those, 34 are derived from the intruder analysis, four from the groundwater pathway analysis, two from the air pathway analysis, and one from the radon emanation analysis.

For the 1,000-year time of assessment, 37 of the 75 radionuclides have disposal limits. Of those, 35 are derived from the intruder analysis, none from the groundwater pathway analysis, two from the air pathway analysis, and none from the radon emanation analysis.

8.9 COMPARISON WITH ESTIMATED LOW CURIE SALT CONCENTRATIONS

Tables 8-5 and 8-6 also show the ratio of the currently estimated average radionuclide concentration in low curie salt to the most restrictive of the limits for each radionuclide (i.e., the fraction of the limit due to disposal of low curie salt). For the 10,000-year time of assessment, the greatest fraction is 0.038 for ^{126}Sn and the total sum-of-fractions of all the limits is 0.084. This shows that low curie salt can be disposed in the SDF without exceeding any of the USDOE performance objectives.

In comparison, if the 1,000-year time of assessment limits were used, the greatest fraction would remain 0.038 for ^{126}Sn and the total sum-of-fractions would decrease to 0.048.

To facilitate comparison, Table 8-7 shows the limits derived at the two assessment time frames.

8.10 SALTSTONE WASTE ACCEPTANCE CRITERIA

Waste Acceptance Criteria for disposal of saltstone derived from this Special Analysis are listed in Table 8-8. These WAC are taken from Table 8-5, with units converted from Ci to pCi by multiplying by 1 x 10^{12} and from liters to milliliters by multiplying by 1000 (i.e., Table 8-5 values are multiplied by 1 x 10^{9}). The limits developed using a 10,000-year time frame for analysis should be used for developing WAC to be consistent with the SRS DAS (Fiori and Frei, 1999).

These WAC only represent limits derived from this special analysis. Many other criteria (e.g., personnel protection, facility safety, State and Federal regulations) must be considered in establishing WAC for the SDF. Among these are restrictions on the disposal of alpha-emitting transuranic radionuclides with half-lives greater than 20 years, for which USDOE has established a limit of less than 100 nCi/g in the waste being disposed. This would equate to 2.7 x 10^{5} pCi/mL of salt solution feed to saltstone, assuming that saltstone contains 47 % by weight salt solution and the density of salt solution is 1.25 grams per milliliter.

Table 8-1. Radionuclide limits for a 10,000-year assessment period, in Ci/vault or Ci/facility

Nuclide	Intruder Limit, Ci/Vault[a]	Groundwater Limit, Total Ci[b]	Air Limit, Ci/Vault[c]	Radon Limit, Ci/Vault[d]
H-3	No limit	1.1×10^{13}	2.1×10^{6}	—[e]
Be-10	No limit	—[e]	—[e]	—[e]
C-14	No limit	1.7×10^{8}	2.6	—[e]
Al-26	1.6×10^{2}	—[e]	—[e]	—[e]
Co-60	3.0×10^{11}	No limit	—[e]	—[e]
Ni-59	No limit	No limit	—[e]	—[e]
Ni-63	No limit	No limit	—[e]	—[e]
Se-79	No limit	3.9×10^{3}	—[e]	—[e]
Sr-90	No limit	No limit	—[e]	—[e]
Y-90	—[f]	No limit	—[e]	—[e]
Zr-93	No limit	No limit	—[e]	—[e]
Nb-93m	No limit	—[e]	—[e]	—[e]
Nb-94	8.5×10^{2}	—[e]	—[e]	—[e]
Tc-99	No limit	4.0×10^{5}	—[e]	—[e]
Ru-106	—[f]	No limit	—[e]	—[e]
Pd-107	No limit	No limit	—[e]	—[e]
Ag-110m	—[f]	No limit	—[e]	—[e]
Cd-113m	No limit	—[e]	—[e]	—[e]
Sn-121m	No limit	No limit	—[e]	—[e]
Sn-126	1.0×10^{3}	1.3×10^{6}	—[e]	—[e]
Sb-125	—[f]	No limit	—[e]	—[e]
Sb-126	—[f]	No limit	—[e]	—[e]
Te-125m	—[f]	No limit	—[e]	—[e]
I-129	No limit	2.0×10^{1}	—[e]	—[e]
Cs-134	—[f]	No limit	—[e]	—[e]

Table is continued on following page

Table 8-1. (continued)

Nuclide	Intruder Limit, Ci/Vault[a]	Groundwater Limit, Total Ci[b]	Air Limit, Ci/Vault[c]	Radon Limit, Ci/Vault[d]
Cs-135	No limit	No limit	—[e]	—[e]
Cs-137	1.0×10^{9}	No limit	—[e]	—[e]
Ce-144	—[f]	No limit	—[e]	—[e]
Pr-144	—[f]	No limit	—[e]	—[e]
Pm-147	—[f]	No limit	—[e]	—[e]
Sm-151	No limit	No limit	—[e]	—[e]
Eu-152	3.3×10^{8}	No limit	—[e]	—[e]
Eu-154	7.5×10^{9}	No limit	—[e]	—[e]
Eu-155	No limit	No limit	—[e]	—[e]
Pb-210	No limit	—[e]	—[e]	—[e]
Ra-226	4.9×10^{2}	—[e]	—[e]	—[e]
Ra-228	9.1×10^{9}	—[e]	—[e]	—[e]
Ac-227	1.2×10^{10}	—[e]	—[e]	—[e]
Th-228	—[f]	No limit	—[e]	—[e]
Th-229	9.0×10^{3}	—[e]	—[e]	—[e]
Th-230	3.4×10^{2}	—[e]	—[e]	—[e]
Th-231	—[f]	No limit	—[e]	—[e]
Th-232	1.6×10^{2}	No limit	—[e]	—[e]
Th-234	—[f]	No limit	—[e]	—[e]
Pa-231	2.0×10^{4}	—[e]	—[e]	—[e]
Pa-234	—[f]	No limit	—[e]	—[e]
U-232	1.5×10^{5}	No limit	—[e]	—[e]
U-233	1.4×10^{4}	No limit	—[e]	—[e]
U-234	4.6×10^{3}	No limit	—[e]	2.0×10^{2}
U-235	9.2×10^{4}	No limit	—[e]	—[e]

Table is continued on following page

Table 8-1. (continued)

Nuclide	Intruder Limit, Ci/Vault[a]	Groundwater Limit, Total Ci[b]	Air Limit, Ci/Vault[c]	Radon Limit, Ci/Vault[d]
U-236	No limit	No limit	—[e]	—[e]
U-238	5.6×10^{4}	No limit	—[e]	—[e]
Np-237	5.3×10^{4}	1.0×10^{5}[g]	—[e]	—[e]
Pu-238	1.3×10^{7}	No limit	—[e]	—[e]
Pu-239	1.8×10^{10}	No limit	—[e]	—[e]
Pu-240	No limit	No limit	—[e]	—[e]
Pu-241	7.9×10^{9}	No limit	—[e]	—[e]
Pu-242	No limit	No limit	—[e]	—[e]
Pu-244	4.0×10^{3}	No limit	—[e]	—[e]
Am-241	2.5×10^{8}	No limit	—[e]	—[e]
Am-242	—[f]	No limit	—[e]	—[e]
Am-242m	1.0×10^{7}	No limit	—[e]	—[e]
Am-243	2.7×10^{5}	No limit	—[e]	—[e]
Cm-242	2.6×10^{9}	No limit	—[e]	—[e]
Cm-243	1.2×10^{13}	No limit	—[e]	—[e]
Cm-244	No limit	No limit	—[e]	—[e]
Cm-245	6.6×10^{6}	No limit	—[e]	—[e]
Cm-246	No limit	No limit	—[e]	—[e]
Cm-247	1.9×10^{4}	No limit	—[e]	—[e]
Cm-248	4.6×10^{7}	No limit	—[e]	—[e]
Bk-249	5.8×10^{7}	—[e]	—[e]	—[e]
Cf-249	1.5×10^{5}	No limit	—[e]	—[e]
Cf-250	No limit	—[e]	—[e]	—[e]
Cf-251	2.0×10^{6}	—[e]	—[e]	—[e]
Cf-252	6.1×10^{12}	No limit	—[e]	—[e]

[a] From Table 4-11.

[b] From Tables 5-6 and 5-7.

[c] From Table 6-2, 100 m Location.

[d] From Sect. 7.

[e] Not considered in analysis.

[f] Screened from analysis.

[g] Based on concentration at 10,000 years.

Table 8-2. Radionuclide limits for a 1,000-year assessment period, in Ci/vault or Ci/facility

Nuclide	Intruder Limit, Ci/Vault[a]	Groundwater Limit, Total Ci[b]	Air Limit, Ci/Vault[c]	Radon Limit, Ci/Vault[d]
H-3	No limit	No limit	2.1×10^{6}	—[e]
Be-10	No limit	—[e]	—[e]	—[e]
C-14	No limit	No limit	2.6	—[e]
Al-26	1.6×10^{2}	—[e]	—[e]	—[e]
Co-60	3.0×10^{11}	No limit	—[e]	—[e]
Ni-59	No limit	No limit	—[e]	—[e]
Ni-63	No limit	No limit	—[e]	—[e]
Se-79	No limit	No limit	—[e]	—[e]
Sr-90	No limit	No limit	—[e]	—[e]
Y-90	—[f]	No limit	—[e]	—[e]
Zr-93	No limit	No limit	—[e]	—[e]
Nb-93m	No limit	—[e]	—[e]	—[e]
Nb-94	8.5×10^{2}	—[e]	—[e]	—[e]
Tc-99	No limit	No limit	—[e]	—[e]
Ru-106	—[f]	No limit	—[e]	—[e]
Pd-107	No limit	No limit	—[e]	—[e]
Ag-110m	—[f]	No limit	—[e]	—[e]
Cd-113m	No limit	—[e]	—[e]	—[e]
Sn-121m	No limit	No limit	—[e]	—[e]
Sn-126	1.0×10^{3}	No limit	—[e]	—[e]
Sb-125	—[f]	No limit	—[e]	—[e]
Sb-126	—[f]	No limit	—[e]	—[e]
Te-125m	—[f]	No limit	—[e]	—[e]
I-129	No limit	No limit	—[e]	—[e]
Cs-134	—[f]	No limit	—[e]	—[e]

Table is continued on following page

Table 8-2. (continued)

Nuclide	Intruder Limit, Ci/Vault[a]	Groundwater Limit, Total Ci[b]	Air Limit, Ci/Vault[c]	Radon Limit, Ci/Vault[d]
Cs-135	No limit	No limit	—[e]	—[e]
Cs-137	1.0×10^{9}	No limit	—[e]	—[e]
Ce-144	—[f]	No limit	—[e]	—[e]
Pr-144	—[f]	No limit	—[e]	—[e]
Pm-147	—[f]	No limit	—[e]	—[e]
Sm-151	No limit	No limit	—[e]	—[e]
Eu-152	3.3×10^{8}	No limit	—[e]	—[e]
Eu-154	7.5×10^{9}	No limit	—[e]	—[e]
Eu-155	No limit	No limit	—[e]	—[e]
Pb-210	No limit	—[e]	—[e]	—[e]
Ra-226	4.9×10^{2}	—[e]	—[e]	—[e]
Ra-228	9.1×10^{9}	—[e]	—[e]	—[e]
Ac-227	1.2×10^{10}	—[e]	—[e]	—[e]
Th-228	—[f]	No limit	—[e]	—[e]
Th-229	9.0×10^{3}	—[e]	—[e]	—[e]
Th-230	9.0×10^{2}	—[e]	—[e]	—[e]
Th-231	—[f]	No limit	—[e]	—[e]
Th-232	1.6×10^{2}	No limit	—[e]	—[e]
Th-234	—[f]	No limit	—[e]	—[e]
Pa-231	2.0×10^{4}	—[e]	—[e]	—[e]
Pa-234	—[f]	No limit	—[e]	—[e]
U-232	1.5×10^{5}	No limit	—[e]	—[e]
U-233	9.1×10^{4}	No limit	—[e]	—[e]
U-234	1.9×10^{5}	No limit	—[e]	—[g]
U-235	4.4×10^{5}	No limit	—[e]	—[e]

Table is continued on following page

Table 8-2. (continued)

Nuclide	Intruder Limit, Ci/Vault[a]	Groundwater Limit, Total Ci[b]	Air Limit, Ci/Vault[c]	Radon Limit, Ci/Vault[d]
U-236	No limit	No limit	—[e]	—[e]
U-238	6.6×10^{4}	No limit	—[e]	—[e]
Np-237	5.6×10^{4}	No limit	—[e]	—[e]
Pu-238	6.4×10^{8}	No limit	—[e]	—[e]
Pu-239	No limit	No limit	—[e]	—[e]
Pu-240	No limit	No limit	—[e]	—[e]
Pu-241	1.1×10^{10}	No limit	—[e]	—[e]
Pu-242	No limit	No limit	—[e]	—[e]
Pu-244	4.0×10^{3}	No limit	—[e]	—[e]
Am-241	3.5×10^{8}	No limit	—[e]	—[e]
Am-242	—[f]	No limit	—[e]	—[e]
Am-242m	4.2×10^{7}	No limit	—[e]	—[e]
Am-243	2.7×10^{5}	No limit	—[e]	—[e]
Cm-242	1.3×10^{11}	No limit	—[e]	—[e]
Cm-243	1.2×10^{13}	No limit	—[e]	—[e]
Cm-244	No limit	No limit	—[e]	—[e]
Cm-245	6.6×10^{6}	No limit	—[e]	—[e]
Cm-246	No limit	No limit	—[e]	—[e]
Cm-247	2.0×10^{4}	No limit	—[e]	—[e]
Cm-248	4.6×10^{8}	No limit	—[e]	—[e]
Bk-249	5.8×10^{7}	—[e]	—[e]	—[e]
Cf-249	1.5×10^{5}	No limit	—[e]	—[e]
Cf-250	No limit	—[e]	—[e]	—[e]
Cf-251	2.0×10^{6}	—[e]	—[e]	—[e]
Cf-252	6.0×10^{13}	No limit	—[e]	—[e]

[a] From Table 4-12.

[b] From Tables 5-6 and 5-8.

[c] From Table 6-2, 100 m Location.

[d] From Sect. 7.

[e] Not considered in analysis.

[f] Screened from analysis.

[g] Not analyzed at 1,000 years.

Table 8-3. Radionuclide limits for a 10,000-year assessment period, in Ci/L salt solution

Nuclide	Intruder Limit, Ci/L salt solution[a]	Groundwater Limit, Ci/L salt solution[b]	Air Limit, Ci/L salt solution[c]	Radon Limit, Ci/L salt solution[d]
H-3	No limit	1.5×10^{4}	4.2×10^{-2}	—[e]
Be-10	No limit	—[e]	—[e]	—[e]
C-14	No limit	2.3×10^{-1}	5.2×10^{-8}	—[e]
Al-26	3.2×10^{-6}	—[e]	—[e]	—[e]
Co-60	6.0×10^{3}	No limit	—[e]	—[e]
Ni-59	No limit	No limit	—[e]	—[e]
Ni-63	No limit	No limit	—[e]	—[e]
Se-79	No limit	5.3×10^{-6}	—[e]	—[e]
Sr-90	No limit	No limit	—[e]	—[e]
Y-90	—[f]	No limit	—[e]	—[e]
Zr-93	No limit	No limit	—[e]	—[e]
Nb-93m	No limit	—[e]	—[e]	—[e]
Nb-94	1.7×10^{-5}	—[e]	—[e]	—[e]
Tc-99	No limit	5.5×10^{-4}	—[e]	—[e]
Ru-106	—[f]	No limit	—[e]	—[e]
Pd-107	No limit	No limit	—[e]	—[e]
Ag-110m	—[f]	No limit	—[e]	—[e]
Cd-113m	No limit	—[e]	—[e]	—[e]
Sn-121m	No limit	No limit	—[e]	—[e]
Sn-126	2.0×10^{-5}	1.8×10^{-3}	—[e]	—[e]
Sb-125	—[f]	No limit	—[e]	—[e]
Sb-126	—[f]	No limit	—[e]	—[e]
Te-125m	—[f]	No limit	—[e]	—[e]
I-129	No limit	2.7×10^{-8}	—[e]	—[e]
Cs-134	—[f]	No limit	—[e]	—[e]

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Table 8-3. (continued)

Nuclide	Intruder Limit, Ci/L salt solution[a]	Groundwater Limit, Ci/L salt solution[b]	Air Limit, Ci/L salt solution[c]	Radon Limit, Ci/L salt solution[d]
Cs-135	No limit	No limit	—[e]	—[e]
Cs-137	2.0×10^{1}	No limit	—[e]	—[e]
Ce-144	—[f]	No limit	—[e]	—[e]
Pr-144	—[f]	No limit	—[e]	—[e]
Pm-147	—[f]	No limit	—[e]	—[e]
Sm-151	No limit	No limit	—[e]	—[e]
Eu-152	6.5	No limit	—[e]	—[e]
Eu-154	1.5×10^{2}	No limit	—[e]	—[e]
Eu-155	No limit	No limit	—[e]	—[e]
Pb-210	No limit	—[e]	—[e]	—[e]
Ra-226	9.7×10^{-6}	—[e]	—[e]	—[e]
Ra-228	1.8×10^{2}	—[e]	—[e]	—[e]
Ac-227	2.4×10^{2}	—[e]	—[e]	—[e]
Th-228	—[f]	No limit	—[e]	—[e]
Th-229	1.8×10^{-4}	—[e]	—[e]	—[e]
Th-230	6.7×10^{-6}	—[e]	—[e]	—[e]
Th-231	—[f]	No limit	—[e]	—[e]
Th-232	3.2×10^{-6}	No limit	—[e]	—[e]
Th-234	—[f]	No limit	—[e]	—[e]
Pa-231	4.0×10^{-4}	—[e]	—[e]	—[e]
Pa-234	—[f]	No limit	—[e]	—[e]
U-232	3.0×10^{-3}	No limit	—[e]	—[e]
U-233	2.8×10^{-4}	No limit	—[e]	—[e]
U-234	9.1×10^{-5}	No limit	—[e]	4.1×10^{-6}
U-235	1.8×10^{-3}	No limit	—[e]	—[e]

Table is continued on following page

Table 8-3. (continued)

Nuclide	Intruder Limit, Ci/L salt solution[a]	Groundwater Limit, Ci/L salt solution[b]	Air Limit, Ci/L salt solution[c]	Radon Limit, Ci/L salt solution[d]
U-236	No limit	No limit	—[e]	—[e]
U-238	1.1×10^{-3}	No limit	—[e]	—[e]
Np-237	1.1×10^{-3}	1.4×10^{-4}[g]	—[e]	—[e]
Pu-238	2.6×10^{-1}	No limit	—[e]	—[e]
Pu-239	3.6×10^{2}	No limit	—[e]	—[e]
Pu-240	No limit	No limit	—[e]	—[e]
Pu-241	1.6×10^{2}	No limit	—[e]	—[e]
Pu-242	No limit	No limit	—[e]	—[e]
Pu-244	7.9×10^{-5}	No limit	—[e]	—[e]
Am-241	5.0	No limit	—[e]	—[e]
Am-242	—[f]	No limit	—[e]	—[e]
Am-242m	2.0×10^{-1}	No limit	—[e]	—[e]
Am-243	5.4×10^{-3}	No limit	—[e]	—[e]
Cm-242	5.2×10^{1}	No limit	—[e]	—[e]
Cm-243	2.4×10^{5}	No limit	—[e]	—[e]
Cm-244	No limit	No limit	—[e]	—[e]
Cm-245	1.3×10^{-1}	No limit	—[e]	—[e]
Cm-246	No limit	No limit	—[e]	—[e]
Cm-247	3.8×10^{-4}	No limit	—[e]	—[e]
Cm-248	9.1×10^{-1}	No limit	—[e]	—[e]
Bk-249	1.2	—[e]	—[e]	—[e]
Cf-249	3.0×10^{-3}	No limit	—[e]	—[e]
Cf-250	No limit	—[e]	—[e]	—[e]
Cf-251	4.0×10^{-2}	—[e]	—[e]	—[e]
Cf-252	1.2×10^{5}	No limit	—[e]	—[e]

[a] From Table 4-11.

[b] From Tables 5-6 and 5-7.

[c] From Table 6-2, 100 m Location.

[d] From Sect. 7.

[e] Not considered in analysis.

[f] Screened from analysis.

[g] Based on concentration at 10,000 years.

Table 8-4. Radionuclide limits for a 1,000-year assessment period

Nuclide	Intruder Limit, Ci/L salt solution[a]	Groundwater Limit, Ci/L salt solution[b]	Air Limit, Ci/L salt solution[c]	Radon Limit, Ci/L salt solution[d]
H-3	No limit	No limit	4.2×10^{-2}	—[e]
Be-10	No limit	—[e]	—[e]	—[e]
C-14	No limit	No limit	5.2×10^{-8}	—[e]
Al-26	3.2×10^{-6}	—[e]	—[e]	—[e]
Co-60	6.0×10^{3}	No limit	—[e]	—[e]
Ni-59	No limit	No limit	—[e]	—[e]
Ni-63	No limit	No limit	—[e]	—[e]
Se-79	No limit	No limit	—[e]	—[e]
Sr-90	No limit	No limit	—[e]	—[e]
Y-90	—[f]	No limit	—[e]	—[e]
Zr-93	No limit	No limit	—[e]	—[e]
Nb-93m	No limit	—[e]	—[e]	—[e]
Nb-94	1.7×10^{-5}	—[e]	—[e]	—[e]
Tc-99	No limit	No limit	—[e]	—[e]
Ru-106	—[f]	No limit	—[e]	—[e]
Pd-107	No limit	No limit	—[e]	—[e]
Ag-110m	—[f]	No limit	—[e]	—[e]
Cd-113m	No limit	—[e]	—[e]	—[e]
Sn-121m	No limit	No limit	—[e]	—[e]
Sn-126	2.0×10^{-5}	No limit	—[e]	—[e]
Sb-125	—[f]	No limit	—[e]	—[e]
Sb-126	—[f]	No limit	—[e]	—[e]
Te-125m	—[f]	No limit	—[e]	—[e]
I-129	No limit	No limit	—[e]	—[e]
Cs-134	—[f]	No limit	—[e]	—[e]

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Table 8-4. (continued)

Nuclide	Intruder Limit, Ci/L salt solution[a]	Groundwater Limit, Ci/L salt solution[b]	Air Limit, Ci/L salt solution[c]	Radon Limit, Ci/L salt solution[d]
Cs-135	No limit	No limit	—[e]	—[e]
Cs-137	2.0×10^{1}	No limit	—[e]	—[e]
Ce-144	—[f]	No limit	—[e]	—[e]
Pr-144	—[f]	No limit	—[e]	—[e]
Pm-147	—[f]	No limit	—[e]	—[e]
Sm-151	No limit	No limit	—[e]	—[e]
Eu-152	6.5	No limit	—[e]	—[e]
Eu-154	1.5×10^{2}	No limit	—[e]	—[e]
Eu-155	No limit	No limit	—[e]	—[e]
Pb-210	No limit	—[e]	—[e]	—[e]
Ra-226	9.7×10^{-6}	—[e]	—[e]	—[e]
Ra-228	1.8×10^{2}	—[e]	—[e]	—[e]
Ac-227	2.4×10^{2}	—[e]	—[e]	—[e]
Th-228	—[f]	No limit	—[e]	—[e]
Th-229	1.8×10^{-4}	—[e]	—[e]	—[e]
Th-230	1.8×10^{-5}	—[e]	—[e]	—[e]
Th-231	—[f]	No limit	—[e]	—[e]
Th-232	3.2×10^{-6}	No limit	—[e]	—[e]
Th-234	—[f]	No limit	—[e]	—[e]
Pa-231	4.0×10^{-4}	—[e]	—[e]	—[e]
Pa-234	—[f]	No limit	—[e]	—[e]
U-232	3.0×10^{-3}	No limit	—[e]	—[e]
U-233	1.8×10^{-3}	No limit	—[e]	—[e]
U-234	3.8×10^{-3}	No limit	—[e]	—[e]
U-235	8.7×10^{-3}	No limit	—[e]	—[e]

Table is continued on following page

Table 8-4. (continued)

Nuclide	Intruder Limit, Ci/L salt solution[a]	Groundwater Limit, Ci/L salt solution[b]	Air Limit, Ci/L salt solution[c]	Radon Limit, Ci/L salt solution[d]
U-236	No limit	No limit	—[e]	—[e]
U-238	1.3×10^{-3}	No limit	—[e]	—[e]
Np-237	1.1×10^{-3}	No limit	—[e]	—[e]
Pu-238	1.3×10^{1}	No limit	—[e]	—[e]
Pu-239	No limit	No limit	—[e]	—[e]
Pu-240	No limit	No limit	—[e]	—[e]
Pu-241	2.2×10^{2}	No limit	—[e]	—[e]
Pu-242	No limit	No limit	—[e]	—[e]
Pu-244	7.9×10^{-5}	No limit	—[e]	—[e]
Am-241	6.9	No limit	—[e]	—[e]
Am-242	—[f]	No limit	—[e]	—[e]
Am-242m	8.3×10^{-1}	No limit	—[e]	—[e]
Am-243	5.4×10^{-3}	No limit	—[e]	—[e]
Cm-242	2.6×10^{3}	No limit	—[e]	—[e]
Cm-243	2.4×10^{5}	No limit	—[e]	—[e]
Cm-244	No limit	No limit	—[e]	—[e]
Cm-245	1.3×10^{-1}	No limit	—[e]	—[e]
Cm-246	No limit	No limit	—[e]	—[e]
Cm-247	4.0×10^{-4}	No limit	—[e]	—[e]
Cm-248	9.1	No limit	—[e]	—[e]
Bk-249	1.2	—[e]	—[e]	—[e]
Cf-249	3.0×10^{-3}	No limit	—[e]	—[e]
Cf-250	No limit	—[e]	—[e]	—[e]
Cf-251	4.0×10^{-2}	—[e]	—[e]	—[e]
Cf-252	1.2×10^{6}	No limit	—[e]	—[e]

[a] From Table 4-12.

[b] From Tables 5-6 and 5-8.

[c] From Table 6-2, 100 m Location.

[d] From Sect. 7.

[e] Not considered in analysis.

[f] Screened from analysis.

Table 8-5. Most restrictive radionuclide limits for a 10,000-year assessment period compared with currently estimated average low curie salt concentrations

Nuclide	Limiting Pathway	Limit, Ci/L salt solution	Low Curie Salt, Ci/L[a]	Low Curie Salt/Limit
H-3	Air	4.2×10^{-2}	0.0	0.0
Be-10	None	No limit		
C-14	Air	5.2×10^{-8}	4.46×10^{-10}	8.7×10^{-3}
Al-26	Intruder	3.2×10^{-6}		
Co-60	Intruder	6.0×10^{3}	4.08×10^{-5}	6.9×10^{-9}
Ni-59	None	No limit	2.57×10^{-7}	
Ni-63	None	No limit	3.98×10^{-10}	
Se-79	Groundwater	5.3×10^{-6}	1.51×10^{-7}	2.8×10^{-2}
Sr-90	None	No limit	8.94×10^{-3}	
Y-90	None	No limit	8.94×10^{-3}	
Zr-93	None	No limit		
Nb-93m	None	No limit		
Nb-94	Intruder	1.7×10^{-5}		
Tc-99	Groundwater	5.5×10^{-4}	2.57×10^{-6}	4.7×10^{-3}
Ru-106	None	No limit	9.50×10^{-7}	
Pd-107	None	No limit		
Ag-110m	None	No limit		
Cd-113m	None	No limit		
Sn-121m	None	No limit		
Sn-126	Intruder	2.0×10^{-5}	7.50×10^{-7}	3.8×10^{-2}
Sb-125	None	No limit	2.43×10^{-5}	
Sb-126	None	No limit	2.02×10^{-7}	
Te-125m	None	No limit	0.00	
I-129	Groundwater	2.7×10^{-8}	2.37×10^{-11}	8.6×10^{-4}
Cs-134	None	No limit	1.06×10^{-6}	

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Table 8-5. (continued)

Nuclide	Limiting Pathway	Limit, Ci/L salt solution	Low Curie Salt, Ci/L	Low Curie Salt/Limit
Cs-135	None	No limit	1.81×10^{-9}	
Cs-137	Intruder	2.0×10^{1}	2.26×10^{-2}	1.1×10^{-3}
Ce-144	None	No limit	4.90×10^{-7}	
Pr-144	None	No limit	4.90×10^{-7}	
Pm-147	None	No limit	5.13×10^{-4}	
Sm-151	None	No limit	0.00	
Eu-152	Intruder	6.5		
Eu-154	Intruder	1.5×10^{2}	1.00×10^{-4}	6.7×10^{-7}
Eu-155	None	No limit	0.00	
Pb-210	None	No limit		
Ra-226	Intruder	9.7×10^{-6}		
Ra-228	Intruder	1.8×10^{2}		
Ac-227	Intruder	2.4×10^{2}		
Th-228	None	No limit		
Th-229	Intruder	1.8×10^{-4}		
Th-230	Intruder	6.7×10^{-6}		
Th-231	None	No limit		
Th-232	Intruder	3.2×10^{-6}	2.88×10^{-10}	9.1×10^{-5}
Th-234	None	No limit		
Pa-231	Intruder	4.0×10^{-4}		
Pa-234	None	No limit		
U-232	Intruder	3.0×10^{-3}	6.23×10^{-11}	2.1×10^{-8}
U-233	Intruder	2.8×10^{-4}	1.72×10^{-8}	6.2×10^{-5}
U-234	Radon	4.1×10^{-6}	6.34×10^{-9}	1.6×10^{-3}
U-235	Intruder	1.8×10^{-3}	2.00×10^{-10}	1.1×10^{-7}

Table is continued on following page

Table 8-5. (continued)

Nuclide	Limiting Pathway	Limit, Ci/L salt solution	Low Curie Salt, Ci/L	Low Curie Salt/Limit
U-236	None	No limit	9.63×10^{-10}	
U-238	Intruder	1.1×10^{-3}	4.78×10^{-9}	4.3×10^{-6}
Np-237	Groundwater	1.4×10^{-4}	8.92×10^{-9}	6.5×10^{-5}
Pu-238	Intruder	2.6×10^{-1}	2.18×10^{-4}	8.4×10^{-4}
Pu-239	Intruder	3.6×10^{2}	3.32×10^{-6}	9.3×10^{-9}
Pu-240	None	No limit	1.55×10^{-6}	
Pu-241	Intruder	1.6×10^{2}	1.06×10^{-4}	6.8×10^{-7}
Pu-242	None	No limit	3.60×10^{-9}	
Pu-244	Intruder	7.9×10^{-5}		
Am-241	Intruder	5.0	2.22×10^{-5}	4.5×10^{-6}
Am-242	None	No limit		
Am-242m	Intruder	2.0×10^{-1}	2.11×10^{-8}	1.1×10^{-7}
Am-243	Intruder	5.4×10^{-3}		
Cm-242	Intruder	5.2×10^{1}	2.20×10^{-5}	4.3×10^{-7}
Cm-243	Intruder	2.4×10^{5}	0.00	
Cm-244	None	No limit	2.20×10^{-5}	
Cm-245	Intruder	1.3×10^{-1}	1.64×10^{-9}	1.3×10^{-8}
Cm-246	None	No limit		
Cm-247	Intruder	3.8×10^{-4}		
Cm-248	Intruder	9.1×10^{-1}		
Bk-249	Intruder	1.2		
Cf-249	Intruder	3.0×10^{-3}		
Cf-250	None	No limit		
Cf-251	Intruder	4.0×10^{-2}		
Cf-252	Intruder	1.2×10^{5}		

[a] From Table 3-1.

Table 8-6. Most restrictive radionuclide limits for a 1,000-year assessment period compared with currently estimated average low curie salt concentrations

Nuclide	Limiting Pathway	Limit, Ci/L salt solution	Low Curie Salt, Ci/L[a]	Low Curie Salt/Limit
H-3	Air	4.2×10^{-2}	0.00	0.0
Be-10	None	No limit		
C-14	Air	5.2×10^{-8}	4.46×10^{-10}	8.7×10^{-3}
Al-26	Intruder	3.2×10^{-6}		
Co-60	Intruder	6.0×10^{3}	4.08×10^{-5}	6.9×10^{-9}
Ni-59	None	No limit	2.57×10^{-7}	
Ni-63	None	No limit	3.98×10^{-10}	
Se-79	None	No limit	1.51×10^{-7}	
Sr-90	None	No limit	8.94×10^{-3}	
Y-90	None	No limit	8.94×10^{-3}	
Zr-93	None	No limit		
Nb-93m	None	No limit		
Nb-94	Intruder	1.7×10^{-5}		
Tc-99	None	No limit	2.57×10^{-6}	
Ru-106	None	No limit	9.50×10^{-7}	
Pd-107	None	No limit		
Ag-110m	None	No limit		
Cd-113m	None	No limit		
Sn-121m	None	No limit		
Sn-126	Intruder	2.0×10^{-5}	7.50×10^{-7}	3.8×10^{-2}
Sb-125	None	No limit	2.43×10^{-5}	
Sb-126	None	No limit	2.02×10^{-7}	
Te-125m	None	No limit	0.00	
I-129	None	No limit	2.37×10^{-11}	
Cs-134	None	No limit	1.06×10^{-6}	

Table is continued on following page

Table 8-6. (continued)

Nuclide	Limiting Pathway	Limit, Ci/L salt solution	Low Curie Salt, Ci/L	Low Curie Salt/Limit
Cs-135	None	No limit	1.81×10^{-9}	
Cs-137	Intruder	2.0×10^{1}	2.26×10^{-2}	1.1×10^{-3}
Ce-144	None	No limit	4.90×10^{-7}	
Pr-144	None	No limit	4.90×10^{-7}	
Pm-147	None	No limit	5.13×10^{-4}	
Sm-151	None	No limit	0.00	
Eu-152	Intruder	6.5		
Eu-154	Intruder	1.5×10^{2}	1.00×10^{-4}	6.7×10^{-7}
Eu-155	None	No limit	0.00	
Pb-210	None	No limit		
Ra-226	Intruder	9.7×10^{-6}		
Ra-228	Intruder	1.8×10^{2}		
Ac-227	Intruder	2.4×10^{2}		
Th-228	None	No limit		
Th-229	Intruder	1.8×10^{-4}		
Th-230	Intruder	1.8×10^{-5}		
Th-231	None	No limit		
Th-232	Intruder	3.2×10^{-6}	2.88×10^{-10}	9.1×10^{-5}
Th-234	None	No limit		
Pa-231	Intruder	4.0×10^{-4}		
Pa-234	None	No limit		
U-232	Intruder	3.0×10^{-3}	6.23×10^{-11}	2.1×10^{-8}
U-233	Intruder	1.8×10^{-3}	1.72×10^{-8}	9.5×10^{-6}
U-234	Intruder	3.8×10^{-3}	6.34×10^{-9}	1.7×10^{-6}
U-235	Intruder	8.7×10^{-3}	2.00×10^{-10}	2.3×10^{-8}

Table is continued on following page

Table 8-6. (continued)

Nuclide	Limiting Pathway	Limit, Ci/L salt solution	Low Curie Salt, Ci/L	Low Curie Salt/Limit
U-236	None	No limit	9.63×10^{-10}	
U-238	Intruder	1.3×10^{-3}	4.78×10^{-9}	3.7×10^{-6}
Np-237	Intruder	1.1×10^{-3}	8.92×10^{-9}	8.0×10^{-6}
Pu-238	Intruder	1.3×10^{1}	2.18×10^{-4}	1.7×10^{-5}
Pu-239	None	No limit	3.32×10^{-6}	
Pu-240	None	No limit	1.55×10^{-6}	
Pu-241	Intruder	2.2×10^{2}	1.06×10^{-4}	4.9×10^{-7}
Pu-242	None	No limit	3.60×10^{-9}	
Pu-244	Intruder	7.9×10^{-5}		
Am-241	Intruder	6.9	2.22×10^{-5}	3.2×10^{-6}
Am-242	None	No limit		
Am-242m	Intruder	8.3×10^{-1}	2.11×10^{-8}	2.5×10^{-8}
Am-243	Intruder	5.4×10^{-3}		
Cm-242	Intruder	2.6×10^{3}	2.20×10^{-5}	8.5×10^{-9}
Cm-243	Intruder	2.4×10^{5}	0.00	
Cm-244	None	No limit	2.20×10^{-5}	
Cm-245	Intruder	1.3×10^{-1}	1.64×10^{-9}	1.2×10^{-8}
Cm-246	None	No limit		
Cm-247	Intruder	4.0×10^{-4}		
Cm-248	Intruder	9.1		
Bk-249	Intruder	1.2		
Cf-249	Intruder	3.0×10^{-3}		
Cf-250	None	No limit		
Cf-251	Intruder	4.0×10^{-2}		
Cf-252	Intruder	1.2×10^{6}		

[a] From Table 3-1.

Table 8-7. Comparison of disposal limits derived from 1,000-year and 10,000-year time frames

Nuclide	1,000-year limit, Ci/L[a]	10,000-year limit, Ci/L[b]	Nuclide	1,000-year limit, Ci/L[a]	10,000-year limit, Ci/L[b]
H-3	4.2×10^{-2}	4.2×10^{-2}	U-234	3.8×10^{-3}	4.1×10^{-6}
C-14	5.2×10^{-8}	5.2×10^{-8}	U-235	8.7×10^{-3}	1.8×10^{-3}
Al-26	3.2×10^{-6}	3.2×10^{-6}	U-238	1.3×10^{-3}	1.1×10^{-3}
Co-60	6.0×10^{3}	6.0×10^{3}	Np-237	1.1×10^{-3}	1.4×10^{-4}
Se-79	No limit	5.3×10^{-6}	Pu-238	1.3×10^{1}	2.6×10^{-1}
Nb-94	1.7×10^{-5}	1.7×10^{-5}	Pu-239	No limit	3.6×10^{2}
Tc-99	No limit	5.5×10^{-4}	Pu-241	2.2×10^{2}	1.6×10^{2}
Sn-126	2.0×10^{-5}	2.0×10^{-5}	Pu-244	7.9×10^{-5}	7.9×10^{-5}
I-129	No limit	2.7×10^{-8}	Am-241	6.9	5.0
Cs-137	2.0×10^{1}	2.0×10^{1}	Am-242m	8.3×10^{-1}	2.0×10^{-1}
Eu-152	6.5	6.5	Am-243	5.4×10^{-3}	5.4×10^{-3}
Eu-154	1.5×10^{2}	1.5×10^{2}	Cm-242	2.6×10^{3}	5.2×10^{1}
Ra-226	9.7×10^{-6}	9.7×10^{-6}	Cm-243	2.4×10^{5}	2.4×10^{5}
Ra-228	1.8×10^{2}	1.8×10^{2}	Cm-245	1.3×10^{-1}	1.3×10^{-1}
Ac-227	2.4×10^{2}	2.4×10^{2}	Cm-247	4.0×10^{-4}	3.8×10^{-4}
Th-229	1.8×10^{-4}	1.8×10^{-4}	Cm-248	9.1	9.1×10^{-1}
Th-230	1.8×10^{-5}	6.7×10^{-6}	Bk-249	1.2	1.2
Th-232	3.2×10^{-6}	3.2×10^{-6}	Cf-249	3.0×10^{-3}	3.0×10^{-3}
Pa-231	4.0×10^{-4}	4.0×10^{-4}	Cf-251	4.0×10^{-2}	4.0×10^{-2}
U-232	3.0×10^{-3}	3.0×10^{-3}	Cf-252	1.2×10^{6}	1.2×10^{5}
U-233	1.8×10^{-3}	2.8×10^{-4}			

[a] From Table 8-6

[b] From Table 8-5

Table 8-8. Saltstone Waste Acceptance Criteria derived from this Special Analysis

Nuclide	WAC, pCi/mL[a]	Nuclide	WAC, pCi/mL[a]
H-3	4.2×10^{7}	U-234	4.1×10^{3}
C-14	5.2×10^{1}	U-235	1.8×10^{6}
Al-26	3.2×10^{3}	U-238	1.1×10^{6}
Co-60	6.0×10^{12}	Np-237[b]	1.4×10^{5}
Se-79	5.3×10^{3}	Pu-238[b]	2.6×10^{8}
Nb-94	1.7×10^{4}	Pu-239[b]	3.6×10^{11}
Tc-99	5.5×10^{5}	Pu-241[b]	1.6×10^{11}
Sn-126	2.0×10^{4}	Pu-244[b]	7.9×10^{4}
I-129	2.7×10^{1}	Am-241[b]	5.0×10^{9}
Cs-137	2.0×10^{10}	Am-242m[b]	2.0×10^{8}
Eu-152	6.5×10^{9}	Am-243[b]	5.4×10^{6}
Eu-154	1.5×10^{11}	Cm-242[b]	5.2×10^{10}
Ra-226	9.7×10^{3}	Cm-243[b]	2.4×10^{14}
Ra-228	1.8×10^{11}	Cm-245[b]	1.3×10^{8}
Ac-227	2.4×10^{11}	Cm-247[b]	3.8×10^{5}
Th-229	1.8×10^{5}	Cm-248[b]	9.1×10^{8}
Th-230	6.7×10^{3}	Bk-249[b]	1.2×10^{9}
Th-232	3.2×10^{3}	Cf-249[b]	3.0×10^{6}
Pa-231	4.0×10^{5}	Cf-251[b]	4.0×10^{7}
U-232	3.0×10^{6}	Cf-252[b]	1.2×10^{14}
U-233	2.8×10^{5}		

[a] From Table 8-5

[b] Actual disposals will be limited by the definition of TRU waste of 100 nCi/gram (i.e., 2.7×10^{5} pCi/mL).

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9. REFERENCES

Baes, C.F. III, and Sharp, R.D. 1983. "A Proposal for Estimation of Soil Leaching Constants for Use in Assessment Models," *J. Environ. Qual.* 12, 17.

Cook, J. R. 1983. Estimation of High Water Table Levels at the Saltstone Disposal Site (Z-Area). DPST-83-607, Savannah River Laboratory, E.I. du Pont de Nemours & Company, Inc., Aiken, SC.

Cook, J.R. 2002. Special Analysis: Correction and Update of E-Area Disposal Limits, WSRC-TR-2002-00047, Rev. 1, Westinghouse Savannah River Company, Savannah River Site, Aiken, SC.

Cook, J. R., and S. K. Salvo. 1992. *Final Vegetative Cover for Closed Waste Sites.* WSRC-RP-92-1361. Westinghouse Savannah River Company, Savannah River Site, Aiken, SC.

Cook, J.R. and Wilhite, E.L. 1998. *Radionuclide Screening and Preliminary Scoping Study for EAV Disposal of APT and TEF Wastes*, WSRC-RP-98-00084, Westinghouse Savannah River Company, Savannah River Site, Aiken, SC.

Cook, J.R., Wilhite, E.L., and Young, K.E. 2000. *Closure Plan for the Z-Area Saltstone Disposal Facility*, WSRC-RP-2000-00426, Westinghouse Savannah River Company, Aiken, South Carolina.

CRC Press, Inc. 1981. CRC Handbook of Chemistry and Physics. Robert C. Weast and Melvin J. Astle, eds. Boca Raton, FL.

Dennehy, K. F., D. C. Prowell, and P. B. McMahon. 1989. *Reconnaissance Hydrological Investigation of the Defense Waste Processing Facility and Vicinity,* E. I. du Pont de Nemours and Company Savannah River Laboratory, Aiken, SC.

Drumm, M.D. 2002. *Low Curie Salt Material Balance*. X-CLC-H-00201, Rev. 1. Westinghouse Savannah River Company, Savannah River Site, Aiken, SC.

Eckerman, K.F., and Ryman, J.C. 1993. *External Exposure to Radionuclides in Air, Water, and Soil*, Federal Guidance Report No. 12, USEPA 402-R-93-081, U.S. Environmental Protection Agency, Washington, DC.

Evans, R.D. 1955. "Radioactive-Series Decay," Chapter 15 in *The Atomic Nucleus*, Mc-Graw Hill, New York.

Fiori, J.J. and Frei, M.W. to Rudy, G. 1999. *Disposal Authorization Statement for the Savannah River Site E-Area Vaults and Saltstone Disposal Facilities*, 9/28/99.

Horton, J.H., and Wilhite, E.L. 1978. *Estimated Erosion Rate at the SRP Burial Ground*, E.I. du Pont de Nemours and Company, Inc., Savannah River Laboratory, Aiken, South Carolina.

ICRP (International Commission on Radiological Protection). 1959. *Recommendation of the International Commission on Radiological Protection,* ICRP Publication 2, Pergamon Press, New York.

ICRP (International Commission on Radiological Protection). 1979. *Limits for Intakes of Radionuclides by Workers,* ICRP Publication 30, Part 1, *Ann. ICRP* 2, No. 3/4.

ICRP (International Commission on Radiological Protection). 1996. *Conversion Coefficients for Use in Radiological Protection Against External Exposure,* ICRP Publication 74, *Ann. ICRP* 26, No. 3/4.

Kocher, D.C. 1980. "Effects of Indoor Residence on Radiation Doses from Routine Releases of Radionuclides to the Atmosphere," *Nucl. Technol.* 48, 171.

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

Kocher, D.C., and Sjoreen, A.L. 1985. "Dose-Rate Conversion Factors for Photon Emitters in Soil," *Health Phys.* 48, 193.

Looney, B.B., C.M. King, and D.E. Stephenson, 1987. *Quality Assurance Program for Environmental Assessment of Savannah River Plant Waste Sites*, DPST-86-725, E.I. du Pont de Nemours & Co., Savannah River Laboratory, Aiken, SC.

Luftig, S.D., and Weinstock, L. 1997. "Establishment of Cleanup Levels for CERCLA Sites with Radioactive Contamination," Directive 9200.4-18, Office of Solid Waste and Emergency Response, U.S. Environmental Protection Agency, Washington, DC.

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

Merrell, G.B, V.C. Rogers, and M. K. Bollenbacher 1986. *The PATHRAE-RAD Performance Assessment Code for the Land Disposal of Radioactive Wastes,* RAE-8511-28, Rogers and Associates Engineering Company, Salt Lake City, Utah.

MMES (Martin Marietta Energy Systems, Inc., EG&G Idaho, Inc., Westinghouse Hanford Company, and Westinghouse Savannah River Company). 1992. *Radiological Performance Assessment for the Z-Area Saltstone Disposal Facility,* WSRC-RP-92-1360, Westinghouse Savannah River Company, Aiken, South Carolina.

NCRP (National Council on Radiation Protection and Measurements). 1999. *Recommended Screening Limits for Contaminated Surface Soil and Review of Factors Relevant to Site-Specific Studies*, NCRP Report No. 129, National Council on Radiation Protection and Measurements, Bethesda, Maryland.

ORNL (Oak Ridge National Laboratory). 1997. *Performance Assessment for Continuing and Future Operations at Solid Waste Storage Area 6*, ORNL-6783/R1, Volume 1 and 2, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Oztunali, O.I., Re, G.C., Moskowitz, P.M., Picazo, E.D., and Pitt, C.J. 1981. *Data Base for Radioactive Waste Management, Volume 3: Impacts Analyses Methodology Report*, NUREG/CR-1759, Vol. 3, Dames and Moore, Inc., U.S. Nuclear Regulatory Commission, Washington, DC.

Oztunali, O.I., and Roles, G.W. 1986. *Update of Part 61 Impacts Analysis Methodology – Methodology Report*, Vol. 1, NUREG/CR-4370, Envirosphere Company and U.S. Nuclear Regulatory Commission, Washington, DC.

RAE (Rogers & Associates Engineering Corporation). 1998. The PATHRAE-RAD Performance Assessment Code for the Land Disposal of Radioactive Wastes Version 3.0, RAE-9740/1-2.

Runchal, A.K. and B. Sagar. 1992. PORFLOW: *A Model for Fluid Flow, Heat and Mass Transport in Multifluid, Multiphase Fractured or Porous Media, Users Manual, Version 2.4*. ACRi/016/Rev. G, Analytic and Computational Research, Inc., Los Angeles, CA.

Tuli, J.K. 2000. *Nuclear Wallet Cards*, sixth edition, Brookhaven National Laboratory, Upton, New York.

Reboul, S.H. 2002. *Conclusions and Path Forward for Sn-126 in Low Curie Salt*. HLW-PRE-2002-0030, Rev. 0. Westinghouse Savannah River Company, Savannah River Site, Aiken, SC.

Shuman, R. G. Merrell, and V.C. Rogers 1986 *Benchmark Calculations for Modeling Contaminant Transport in the Unsaturated Zone Using PATHRAE*, RAE-8511-27, Rogers & Associates Engineering Corporation, Salt Lake City, Utah.

Shuman, R. and G. Merrell 1998a *Verification of the PATHRAE-RAD Computer Code*, RAE-9740/1-1, Rogers & Associates Engineering Corporation, Salt Lake City, Utah.

Shuman, R. and G. Merrell 1998b *Modifications made to the PATHRAE Computer Codes to Accommodate Diffusion and Time-Dependent Source Release Modeling*, RAE-9740/1-4, Rogers & Associates Engineering Corporation, Salt Lake City, Utah.

U.S. Department of Agriculture. 1985. *Site Specific Cropland Erosion Inventory*, Soil Conservation Service, Columbia, South Carolina.

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 No. 22.

USDOE (U.S. Department of Energy). 1988. "Management of Low-Level Waste," Chapter III in *Radioactive Waste Management*, USDOE Order 5820.2A, U.S. Department of Energy, Washington, DC.

USDOE (U.S. Department of Energy). 1990. *Radiation Protection of the Public and the Environment*, USDOE Order 5400.5, U.S. Department of Energy, Washington, DC.

USDOE (U.S. Department of Energy). 1993. "10 CFR Part 834 – Radiation Protection of the Public and the Environment," Proposed Rule, *Fed. Register.* 58, 16268.

USDOE (U.S. Department of Energy). 1996. *Interim Format and Content Guide and Standard Review Plan for U.S. Department of Energy Low-Level Waste Disposal Facility Performance Assessments*, U.S. Department of Energy, Washington, DC (October 31).

USDOE (U.S. Department of Energy). 1999a. "Low-Level Waste Requirements," Chapter IV in *Radioactive Waste Management Manual*, USDOE M 435.1-1, U.S. Department of Energy, Washington, DC (July 9).

USDOE (U.S. Department of Energy). 1999b. "Transuranic Waste Requirements," Chapter III in *Radioactive Waste Management Manual*, DOE M 435.1-1, U.S. Department of Energy, Washington, DC (July 9).

USEPA (U.S. Environmental Protection Agency). 2000. *National Primary Drinking Water Regulations; Radionuclides; Final Rule*. 40 CFR Parts 141 and 142. U.S. Environmental Protection Agency, Washington, DC, (December 7).

USEPA (U.S. Environmental Protection Agency). 2001. *Implementation Guidance for Radionuclides, Appendix I, Comparison of Derived Values of Beta and Photon Emitters.* EPA 816-D-00-002. U.S. Environmental Protection Agency, Washington, DC, (August).

USEPA (U.S. Environmental Protection Agency). 2002. Identification and Listing of Hazardous Waste. 40 CFR Part 261. U.S. Environmental Protection Agency, Washington, DC.

USNRC (U.S. Nuclear Regulatory Commission). 1977. *Regulatory Guide 1.109. Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I*, Washington, DC.

USNRC (U.S. Nuclear Regulatory Commission). 1982. "10 CFR Part 61 – Licensing Requirements for Land Disposal of Radioactive Waste," Final Rule, *Fed. Register*. 47, 57446.

USNRC and CEC (U.S. Nuclear Regulatory Commission and Commission of the European Communities). 1997. *Probabilistic Accident and Consequence Uncertainty Analysis: Food Chain Uncertainty Assessment*, NUREG/CR-6523, EUR 16771, SAND97-0335, Vol. 1 and 2, Commission of the European Communities, Luxembourg.

Wood, D.E. et al 1994 *Performance Assessment Task Team Progress Report,* DOE/LLW-157, Rev. 1, Idaho National Enginering Laboratory, EG&G Idaho, Inc., Idaho Falls, ID.

WSRC (Westinghouse Savannah River Company). 1992. *Safety Analysis 200 Area Replacement Tritium Facility.* WSRC-SA-1-1. Savannah River Laboratory, Aiken, SC.

WSRC (Westinghouse Savannah River Company). 1998. *Addendum to the Radiological Performance Assessment for the Z-Area Saltstone Disposal Facility at the Savannah River Site*, WSRC-RP-98-00156, Westinghouse Savannah River Company, Aiken, South Carolina.

40 CFR 61. *Code of Federal Regulations*, Title 40, Protection of Environment, Part 61, "National Emissions Standards for Hazardous Air Pollutants".

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

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

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APPENDIX A

DISCUSSION OF PREVIOUS INTRUDER ANALYSIS

Sect. A.1 discusses the exposure scenarios that were assumed in the existing PA for the SDF (MMES et al., 1992) and summarizes the results of the analysis. Sect. A.2 presents a reevaluation of the results of the previous dose analysis for inadvertent intruders taking into account, first, changes in estimates of the inventories of important radionuclides in waste intended for disposal in the SDF and, second, certain assumptions used in the previous analysis that are not justified on technical grounds. Sect. A.3 discusses changes in the design of the cover system for disposal vaults, compared with the design assumed in the existing PA (Cook et al., 2000), that were necessitated by the reevaluation of the previous intruder dose analysis.

A.1 ANALYSIS OF INADVERTENT INTRUSION IN EXISTING PA

In the existing PA for the SDF (MMES et al., 1992), two scenarios for chronic exposure of inadvertent intruders to residual solid waste in the facility, referred to as the agriculture and resident scenarios, were found to be the most important and were evaluated. The following sections discuss the assumed exposure scenarios and the results of the dose analysis for inadvertent intruders based on the assumed scenarios. These scenarios also are summarized in Table A-1. Additional scenarios for chronic exposure of inadvertent intruders that were evaluated but found to be unimportant and other scenarios for acute exposure of inadvertent intruders that were discussed but could be dismissed as unimportant (MMES et al., 1992) are not considered further in this report.

A.1.1 Description of Agriculture Scenario

In the agriculture scenario, an inadvertent intruder is assumed to construct a home on top of a disposal vault, with the foundation of the home extending into the waste itself. Some of the waste and other uncontaminated cover materials above the waste that are exhumed during digging of the foundation are assumed to be mixed with native soil in the intruder's vegetable garden. The agriculture scenario then occurs after construction of the home is completed and the vegetable garden established, and the following pathways involving chronic exposure are assumed to occur:

- ingestion of contaminated vegetables grown in contaminated garden soil;
- direct ingestion of contaminated soil from the garden in conjunction with intakes of vegetables;
- external exposure to contaminated soil while working in the garden or residing in the home on top of uncovered waste in a disposal vault;
- inhalation of radionuclides suspended into air from contaminated soil while working in the garden or residing in the home.

Table A-1. Summary of important scenarios for exposure of inadvertent intruders evaluated in existing PA for SDF[a]

Scenario	Time of occurrence	Summary description
Agriculture	10,000 to 20,000 years	Intruder excavates into saltstone that has weathered to soil-like material and mixes some exhumed waste with native soil in vegetable garden Exposure pathways involve ingestion, external, and inhalation exposures from radionuclides in garden soil and external and inhalation exposures during indoor residence in home on top of uncovered (unshielded) saltstone Scenario is assumed to be precluded by presence of physical barriers to excavation including intact vault roof, intact grout layer above waste, and intact saltstone monolith; scenario is not credible until all barriers above waste have lost their physical integrity (i.e., failed) and a substantial layer of saltstone has weathered to soil-like material
Resident	100 years	Intruder constructs home on top of intact vault roof and receives external exposure during indoor residence Thickness of vault roof and layer of grout above waste was assumed to be 0.75 m
	Few thousand years	Intruder constructs home on top of intact saltstone monolith and receives external exposure during indoor residence Scenario is assumed to be precluded by presence of physical barriers to excavation including intact vault roof and intact grout layer above waste; scenario is not credible until all barriers above waste have lost their physical integrity (i.e., failed)

[a]See Sect. 3.2.4.1 of MMES et al. (1992).

The credibility of the agriculture scenario depends on two conditions. First, excavation into waste in a disposal vault is assumed not to be a credible occurrence until the engineered barriers above the waste, including the vault roof and layer of uncontaminated grout, have lost their physical integrity (i.e., failed) and a substantial thickness of saltstone has weathered to soil-like material. This condition is based on an assumption that an intact engineered barrier above the waste or an unweathered saltstone monolith cannot readily be penetrated by the types of excavation equipment normally used in the vicinity of the SRS, and that an intruder who excavates at the site would recognize the presence of such intact materials and either take suitable precautions or decide to excavate elsewhere. Second, when all engineered barriers have failed and excavation into saltstone would be a credible occurrence, the minimum depth of saltstone below the ground surface must be less than a typical maximum depth of an excavation in digging a foundation for a home. Thus, either the presence of an intact engineered barrier or a sufficient depth of burial of waste below the ground surface would preclude occurrence of the agriculture scenario. The key assumption in the agriculture scenario is that excavation into buried waste occurs.

In the previous analysis (MMES et al., 1992), only the presence of intact engineered barriers, including unweathered saltstone, was assumed to preclude the agriculture scenario. When the barriers were assumed to fail, it was assumed that an excavation at the ground surface could reach well below the minimum depth of buried waste.

A.1.2 Description of Resident Scenario

As in the agriculture scenario described above, the resident scenario assumes that an inadvertent intruder excavates a foundation for a home on top of a disposal vault. However, excavation into saltstone is assumed to be precluded, either because the intruder encounters an intact engineered barrier (e.g., vault roof) that cannot be readily penetrated by the types of excavation equipment normally used in the vicinity of the SRS, or because the depth of buried waste is greater than a typical maximum depth of an excavation in digging a foundation for a home (i.e., 3 m). The resident scenario then occurs after the home is constructed, and the only relevant exposure pathway is external exposure to photon-emitting radionuclides in the waste while residing in the home. The presence of uncontaminated material above the waste would preclude inhalation or ingestion exposures.

In the previous analysis (MMES et al., 1992), only the presence of intact engineered barriers was assumed to preclude excavation into the waste. Therefore, an intruder's home was assumed to be constructed either on top of an intact vault roof or, at a later time, on top of an intact saltstone monolith. The analysis in either case assumed that an excavation at the ground surface could reach the depth of the intact barrier.

A.1.3 Summary of Results of Previous Intruder Dose Analysis

In the existing PA for the SDF (MMES et al., 1992), estimated inventories of radionuclides in waste intended for disposal in the facility and the assumed scenarios described above were used to estimate annual effective dose equivalents to an inadvertent intruder. The estimated doses were compared with the performance objective of 100 mrem per year for chronic

exposure of inadvertent intruders (USDOE, 1988). The results of the previous analysis are summarized as follows.

A.1.3.1 Results of previous analysis of agriculture scenario. In the previous analysis (MMES et al., 1992), the agriculture scenario was assumed not to be a credible occurrence for at least 10,000 years after disposal (see Table A-1). The assumed time of occurrence of the agriculture scenario was based on an analysis of the long-term performance of engineered barriers with respect to deterring excavation into saltstone.

First, based on an analysis of the rate of degradation of the reinforced concrete roof on a disposal vault, the roof was assumed to maintain its physical integrity and, thus, provide a barrier to excavation for at least 1,000 years after disposal, and perhaps for as long as 10,000 years.

Second, the layer of uncontaminated grout above the waste and the saltstone monolith itself were assumed to weather to soil-like material at the same rate as carbonate rock (limestone) in regions near the SRS. Furthermore, weathering was assumed to be a surface phenomenon, and the surface of the saltstone monolith was assumed not to degrade until the entire layer of grout had weathered. The weathering rate of grout and saltstone was assumed to be 0.1 m per 1,000 years; this value is at the upper end of the range of weathering rates for carbonate rock summarized by Ketelle and Huff (1984). Based on the assumed weathering rate, the time required for the entire layer of grout and an appreciable thickness (0.5 m) of saltstone to weather to soil-like material was estimated to be about 10,000 years.

The assumed lifetime of the barriers to excavation into saltstone of at least 10,000 years was sufficiently long that only long-lived radionuclides were important in the agriculture scenario. A specific assumption about the lifetime of the barriers to excavation was not needed because, first, a time frame for an intruder dose analysis was not specified in the requirements that applied at the time the analysis was performed (USDOE, 1988) and, second, the estimates of dose are not sensitive to the assumed time frame when only long-lived radionuclides are important. At 10,000 years and beyond, erosion of the cover material above a vault was assumed to be sufficient that an excavation would reach the assumed depth of weathered saltstone of 0.5 m.

When the agriculture scenario at 10,000 to 20,000 years after disposal was applied to the expected inventories of radionuclides in waste intended for disposal in the SDF, the annual effective dose equivalent was estimated to be 450 mrem. This result was more than four times higher than the performance objective of 100 mrem (USDOE, 1988). The estimated dose was due primarily to the assumed inventory of Sn-126, and the dose from Sn-126 was determined almost entirely by the dose from external exposure during indoor residence on top of uncovered (unshielded) saltstone. However, the model used to estimate external dose during indoor residence was believed to be conservative, and two approaches were taken in an attempt to provide a more realistic estimate of dose from this exposure pathway.

In one approach, a correction factor to account for the finite lateral extent of an excavation was applied. Based on a previous analysis (USNRC, 1981; Oztunali and Roles, 1986), it was

assumed that the external dose during indoor residence would be reduced by a factor of about four when the finite size of an excavation was taken into account.

In an alternative approach, it was assumed that an intruder would cover the exposed saltstone outside the home with a layer of uncontaminated soil in order to grow grass or other ground cover. The minimum thickness of soil required to provide a sufficient root zone was assumed to be 30 cm, and calculations indicated that a layer of uncontaminated soil of this thickness above the waste would reduce the external dose during indoor residence by a factor of about 30.

When these corrections were applied individually, the result was an estimated annual effective dose equivalent to an inadvertent intruder in the agriculture scenario of 50-110 mrem. Based on consideration of the two correction factors described above, which were not combined, it was concluded that the dose probably would be somewhat less than the upper end of this range and, therefore, that compliance with the performance objective of 100 mrem was demonstrated.

A.1.3.2 Results of previous analysis of resident scenario. In the previous analysis (MMES et al., 1992), the resident scenario was assumed to occur at two different times after disposal (see Table A-1). The first was at 100 years, when institutional control over the disposal site was assumed to be relinquished and all engineered barriers above the waste were assumed to be intact, and the second was between 1,000 and 10,000 years, when the reinforced concrete roof on a disposal vault was assumed to have failed (i.e., lost its physical integrity) and the entire layer of uncontaminated grout was assumed to have weathered to soil-like material. A specific assumption about the time beyond 1,000 years when residence on top of uncovered saltstone could occur was not made. Again, such an assumption was not needed because the dose from exposure to long-lived radionuclides is insensitive to the assumed time of occurrence of the scenario.

At 100 years after disposal, the resident scenario was evaluated based on an assumption that the intruder's home was constructed directly on top of an intact vault roof. It was assumed that the thickness of the vault roof would be about 0.45 m and the thickness of the underlying grout layer would be 0.3 m. Therefore, the resident scenario at 100 years was evaluated assuming a thickness of shielding of 0.75 m above the waste. The estimated annual effective dose equivalent was only about 0.6 mrem, or more than a factor of 100 less than the performance objective of 100 mrem (USDOE, 1988). The estimated dose from Cs-137 was 0.5 mrem, and the remainder was due almost entirely to Sn-126. The corrections described in the previous section to take into account the finite size of an excavation or the likely presence of at least 30 cm of uncontaminated soil around the home were not included in the analysis of the resident scenario.

At 1,000 years after disposal and beyond, the resident scenario was evaluated based on an assumption that the intruder's home was constructed directly on top of an intact saltstone monolith. Thus, the dose during indoor residence at these times was the same as the dose from this exposure pathway in the agriculture scenario. As described in the previous section, a presumably conservative estimate of dose was more than four times higher than the performance objective of 100 mrem, and application of correction factors to account for the

finite size of an excavation or the likelihood that at least 30 cm of uncontaminated soil would be placed around the home reduced the estimated dose to an extent sufficient to demonstrate compliance with the performance objective.

A.2 REEVALUATION OF ANALYSIS OF INADVERTENT INTRUSION IN EXISTING PA

The dose analysis for inadvertent intruders presented in this report was undertaken partly in response to the results of a reevaluation of the previous analysis summarized in Sect. A.1. A reevaluation of the previous analysis was prompted by three concerns. First, estimated inventories of important radionuclides in waste intended for disposal in the SDF have changed substantially. Second, the design thickness of the concrete roof and layer of grout above the saltstone has changed. Third, the two corrections that were applied in an effort to reduce the doses in the agriculture and resident scenarios could not be justified. These concerns are discussed in the following three sections.

A.2.1 Inventories of Important Radionuclides

In the existing PA for the SDF (MMES et al., 1992), the most important radionuclides in regard to potential doses to inadvertent intruders were the photon emitters Sn-126 and Cs-137. The assumed inventory of Sn-126 was especially important, given that the best estimates of dose in the agriculture and resident scenarios at times long after disposal were nearly equal to the performance objective of 100 mrem per year.

The expected inventories of Sn-126 and Cs-137 in disposed waste have increased substantially since the time of the previous analysis (see Sect. 3.3 and Table 3-1). The expected inventory of Sn-126 now is about four times higher than assumed previously. As a result, the estimated dose from Sn-126 in the agriculture and resident scenarios at times long after disposal, based on the assumptions and results described in Sect. A.1.3, would exceed the performance measure of 100 mrem per year (USDOE, 1999), even when the corrections that take into account the finite size of an excavation or the likely presence of at least 30 cm of uncontaminated soil around the home are included.

The expected inventory of Cs-137 now is more than 800 times higher than assumed previously. As a result, the estimated dose from Cs-137 in the resident scenario at 100 years after disposal based on the thickness of shielding of 0.75 m assumed in the previous analysis would exceed the performance measure of 100 mrem per year (USDOE, 1999) by about a factor of four (see Sect. A.1.3.2). If the corrections that take into account the finite size of an excavation or the likely presence of at least 30 cm of uncontaminated soil around the home were included, the estimated dose could be about as high as the performance measure.

A.2.2 Design of Disposal Vault

In the existing PA for the SDF (MMES et al., 1992), the dose analysis for the resident scenario at 100 years after disposal was based on an assumption that the thickness of an intact vault roof would be about 0.45 m and the thickness of the grout layer above the saltstone would be 0.3 m. Thus, the total thickness of shielding above the waste was assumed to be 0.75 m. However, the design thickness of the vault roof and the grout layer

above the saltstone has changed. In the current design, the thickness of a vault roof is 0.1 m and the thickness of the grout layer is 0.4 m (Cook et al., 2000). Thus, the assumed thickness of shielding in the resident scenario at 100 years should be 0.5 m. A reduction in the assumed thickness of shielding from 0.75 to 0.5 m increases the external dose from Cs-137 by a factor of about 13 (Kocher and Sjoreen, 1985).

Taking into account the increase in the expected inventory of Cs-137 by a factor of about 800 noted in the previous section and the increase in external dose due to the decrease in shielding, the dose from Cs-137 in the resident scenario at 100 years would exceed the performance measure of 100 mrem per year (USDOE, 1999) by about a factor of 50. Even if the corrections to take into account the finite size of an excavation or the likelihood that at least 30 cm of uncontaminated soil would be placed around the home are included, the dose from Cs-137 in the resident scenario at 100 years would exceed the performance measure by a substantial amount.

A.2.3 Corrections to Estimated Doses in Agriculture and Resident Scenarios

As described in Sect. A.1.3.2, two corrections were applied to the model used to estimate external dose during indoor residence on top of a disposal facility in the agriculture and resident scenarios in an effort to obtain more realistic estimates of dose. One correction took into account that the source region would be finite in lateral extent, as defined by the size of a typical excavation in digging a foundation for a home. The saltstone outside this area would remain covered and, thus, would be shielded completely. Based on an analysis by the USNRC (Oztunali and Roles, 1986; Sect. 4.2.4), the finite size of an excavation was assumed to reduce the external dose during indoor residence by a factor of four. The second correction was based on an assumption that an inadvertent intruder, in an effort to grow grass or other ground cover around the home, would add a layer of uncontaminated topsoil to the uncovered saltstone outside the home. The minimum thickness of soil cover required to sustain plant growth was assumed to be 30 cm, and this amount of shielding reduced the external dose by a factor of 30.

The two corrections used in the previous analysis to reduce estimates of external dose during indoor residence could be considered for use in the present analysis. However, further review of the bases for the assumed reductions indicates that they are not justified on technical grounds.

The main difficulty with the corrections used in the previous analysis is in the assumptions used by the USNRC to develop a correction to account for the finite size of an excavation (Oztunali and Roles, 1986; Sect. 4.2.4). This correction was based on two assumptions. First, the USNRC assumed that the home located on top of uncovered waste is constructed with a foundation slab consisting of a 30-cm thick layer of concrete. The presence of such a slab substantially reduces the external dose from exposure to sources beneath the home, and most of the external dose then is due to sources outside the home. In contrast, the model used in the intruder dose analysis for the SDF (MMES et al., 1992) assumes that the home is constructed on top of uncovered saltstone, and that the flooring used in the home is a thin layer of wood or other material that provides only a small amount of shielding from sources beneath the home. Assumptions about how a home is constructed clearly are subjective, but

excavation for a basement and the use of a minimal foundation or flooring is not uncommon and, thus, is assumed to provide a reasonably conservative basis for a dose assessment.

The second assumption used by the USNRC in correcting for the finite size of an excavation was that the sources located outside the home are confined to the ground surface, rather than distributed with depth in soil. This assumption greatly overestimates the reduction in external dose when the finite extent of an excavation is taken into account, because it ignores the shielding provided by the source region itself. The USNRC assumed that the radius of a typical home is 8 m and that the radius of an excavation is 25 m. These assumptions combined with the assumption described above that the sources beneath the home are shielded by 30 cm of concrete led to the calculated reduction in dose by a factor of four compared with the dose from exposure to an infinite, unshielded source region. However, when the distribution of sources with depth in soil is taken into account in estimating the dose from a finite source region (Blizard et al., 1968; Sect. 6.4.3.2), only about 15% of the dose is due to sources located more than 8 m away and the contribution from sources beyond 25 m is nearly zero. Thus, when a minimal foundation or flooring for a home is assumed, most of the dose during indoor residence is due to exposure to sources beneath the home, and the dose is insensitive to an assumption about the lateral extent of the source region outside the home. Even if the home were constructed on top of a thick concrete slab, as assumed by the USNRC, the dose from a source region of radius 25 m would be nearly the same as the dose from an infinite source region.

Now consider the reduction in external dose used in the previous analysis that was based on an assumption that a 30-cm thick layer of uncontaminated soil would be placed on top of saltstone outside the home (MMES et al., 1992). A reevaluation of the analysis indicates that this reduction is unreasonable on two counts. First, it was mistakenly assumed that the added layer of soil covered all saltstone, including saltstone beneath the home. Second, based on the results described in the previous paragraph, it is evident that the external dose during indoor residence is insensitive to the thickness of the soil cover outdoors when the home is assumed to be constructed on top of uncovered saltstone, because nearly all of the dose is due to sources beneath the house in any case.

This discussion shows that the reductions in dose from external exposure during indoor residence that were applied in the previous analysis of the agriculture and resident scenarios in an effort to obtain more realistic estimates of dose (MMES et al., 1992) are unjustified on technical grounds and, therefore, should not be applied in a dose analysis for these scenarios. That is, estimates of dose based on an assumption that the source region is unshielded and infinite in lateral extent are not evidently conservative. This result has two consequences. First, based on the expected inventory of Sn-126, the dose during indoor residence in the agriculture scenario or in the resident scenario involving exposure on top of uncovered waste would greatly exceed the performance measure of 100 mrem per year (USDOE, 1999) if either scenario were a credible occurrence within the time frame of concern for an intruder dose analysis. Second, based on the expected inventory of Cs-137 and the thickness of a vault roof and layer of uncontaminated grout documented in a closure plan for the SDF (Cook et al., 2000), the dose in the resident scenario at 100 years after disposal would greatly exceed the performance measure.

To mitigate the unacceptably high doses from disposal of Sn-126 and Cs-137 that would be obtained based on the previous intruder dose analysis, a change in the documented design of the cover system on a disposal vault (Cook et al., 2000) was made in this analysis. This change is described in the following section. The impact of the change on the selection of credible scenarios for exposure of inadvertent intruders is described in Sect. 4.2.2.

A.3 CHANGE IN DESIGN OF COVER SYSTEM FOR DISPOSAL VAULT

The discussions in the previous section indicate that there are two problems that need to be addressed to allow disposal of wastes that are intended for placement in the SDF based on a dose analysis for inadvertent intruders.

First, the disposal system must be designed and constructed in such a way that the agriculture scenario and the resident scenario involving exposure on top of uncovered waste at times long after disposal are not credible occurrences within the time frame of concern for an intruder dose analysis. If either scenario is a credible occurrence, the results of an analysis described above indicate that the expected inventories of Sn-126 in waste intended for disposal in the SDF are sufficiently high that disposal of the waste would be unacceptable.

In the previous analysis (MMES et al., 1992), a time frame for an intruder dose analysis was not specified, but a time frame of as long as 10,000 years is assumed in the present analysis (see Sect. 2.3). As discussed in Sect. A.1.1, the agriculture scenario and the resident scenario involving exposure on top of uncovered waste can be precluded either by including engineered barriers in the cover system above the waste that would maintain their physical integrity and deter excavation to the depth of saltstone for a period longer than 10,000 years or by including a sufficient depth of cover material above the waste that the depth of saltstone below ground during the 10,000-year time frame would be greater than a typical maximum depth of an excavation in digging a foundation for a home.

The second problem is that the estimated dose from Cs-137 in the resident scenario at 100 years after disposal greatly exceeds the performance measure of 100 mrem per year (USDOE, 1999) when the vault roof and layer of uncontaminated grout above the waste are assumed to provide 0.5 m of shielding. Similar to the case of the agriculture scenario and the resident scenario involving exposure on top of uncovered waste discussed above, the dose in the resident scenario at 100 years can be reduced either by increasing the thickness of engineered barriers above the waste or by including a sufficient depth of cover material above the waste that the depth of the engineered barriers below ground would be greater than a typical maximum depth of an excavation in digging a foundation for a home (i.e., 3 m).

Based on the reevaluation of the previous dose analysis for inadvertent intruders and the resulting estimates of dose in the agriculture and resident scenarios described above, the design of the cover system above a disposal vault documented previously (Cook et al., 2000) has been modified to include an additional layer of grout above the reinforced concrete roof on a vault. The design thickness of the additional grout layer is 1 m. No other changes in the documented design of the cover system have been made. With this addition, the design of the cover system, including all layers between the ground surface and the buried waste (saltstone), is as summarized in Table 4-3 (see Sect. 4.2.1).

A.4 REFERENCES

Blizard, E.P., Foderaro, A., Goussev, N.G., and Kovalev, E.E. 1968. "Extended Radiation Sources (Point Kernel Integrations)," Chapter 6 in *Engineering Compendium on Radiation Shielding*, Vol. 1, ed. by Jaeger, R.G., Springer-Verlag, New York.

Cook, J.R., Wilhite, E.L., and Young, K.E. 2000. *Closure Plan for the Z-Area Saltstone Disposal Facility*, WSRC-RP-2000-00426, Westinghouse Savannah River Company, Aiken, South Carolina.

Ketelle, R.H, and Huff, D.D. 1984. *Site Characterization of the West Chestnut Ridge Site*, ORNL/TM-9229, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Kocher, D.C., and Sjoreen, A.L. 1985. "Dose-Rate Conversion Factors for Photon Emitters in Soil," *Health Phys.* 48, 193.

MMES (Martin Marietta Energy Systems, Inc., EG&G Idaho, Inc., Westinghouse Hanford Company, and Westinghouse Savannah River Company). 1992. *Radiological Performance Assessment for the Z-Area Saltstone Disposal Facility*, WSRC-RP-92-1360, Westinghouse Savannah River Company, Aiken, South Carolina.

Oztunali, O.I., and Roles, G.W. 1986. *Update of Part 61 Impacts Analysis Methodology – Methodology Report*, Vol. 1, NUREG/CR-4370, Envirosphere Company and U.S. Nuclear Regulatory Commission, Washington, DC.

USDOE (U.S. Department of Energy). 1988. "Management of Low-Level Waste," Chapter III in *Radioactive Waste Management*, USDOE Order 5820.2A, U.S. Department of Energy, Washington, DC.

USDOE (U.S. Department of Energy). 1999. "Low-Level Waste Requirements," Chapter IV in *Radioactive Waste Management Manual*, USDOE M 435.1-1, U.S. Department of Energy, Washington, DC (July 9).

USNRC (U.S. Nuclear Regulatory Commission). 1981. *Draft Environmental Impact Statement on 10 CFR Part 61 "Licensing Requirements for Land Disposal of Radioactive Waste"*, NUREG-0782, U.S. Nuclear Regulatory Commission, Washington, DC.

APPENDIX B

QUALITY ASSURANCE

To satisfy quality assurance requirements for this Special Analysis, a careful check of data entered into all tables was made to lessen the possibility of transcription errors when copying data from one table to another. For the intruder analysis, it was deemed prudent to apply two methods for computing limits on inventory and concentration and comparing the results to illuminate any possible errors by either method.

The results presented in Tables 4-7 through 4-10 were derived by hand calculations using the appropriate equations described in the text of Sect. 4. As a check of the accuracy of these calculations, data from Tables 4-1, 4-5 and 4-6 were entered into a Microsoft® Excel (Microsoft Corporation, 2001) spreadsheet, and the values in Tables 4-7 through 4-10 were recalculated electronically. The inventory limits in Tables 4-8 through 4-10 for each radionuclide and exposure scenario evaluated were then entered back into another column of the spreadsheet, and these results were compared to the electronically-derived values.

The results of the comparison of hand-calculated versus electronically-generated values are presented in Tables B-1, B-2, and B-3 for each exposure scenario evaluated. A separate column in each of these tables shows the calculated percent difference when comparing the values from each method for each radionuclide. The percent difference reported was generally on the order of a few percent or less, and always less than 7%, with the maximum difference occurring for radionuclides with multiple decay products contributing to the calculated dose per unit activity. The discrepancies between the two sets of numbers for each exposure scenario were determined to be a result of the carrying-through of excess significant digits for each intermediate calculation in the spreadsheet. The hand calculations were carried out by rounding intermediate results in a chain of calculations to two significant digits. While rounding is possible using the spreadsheet program, it is a very cumbersome procedure, and the agreement between the two sets of numbers did not warrant further refinement of the check for accuracy.

Additional design checks of Sects. 4, 5, 6, and 7 are documented separately in Wilhite, 2002 and Lee, 2002.

REFERENCES

Lee, P.L, 2002 *Technical Report Design Check of Intruder Analysis in WSRC-TR-2002-00456*, SRT-EST-2002-00-178, Westinghouse Savannah River Company, Savannah River Site, Aiken, SC

Microsoft Corporation. 2001. *Microsoft ® Excel 2002*.

Wilhite, E.L., 2002 *Design Check of Saltstone Special Analysis*, WSRC-TR-2002-00456, Westinghouse Savannah River Company, Savannah River Site, Aiken, SC

Table B-1. Comparison of hand calculated limits on inventory with spreadsheet calculation of same limits for 100-year intruder resident scenario

	Inventory Limit		
	Excel Spreadsheet	Hand Calculation[a]	Percent
Radionuclide	Ci/vault	Ci/vault	difference
Co-60	2.8E+11	3.0E+11	5.2
Cs-137	1.0E+09	1.0E+09	-4.7
Ba-137m			
Eu-152	3.3E+08	3.3E+08	-1.1
Eu-154	7.5E+09	7.5E+09	0.2
Eu-155	8.0E+23	no limit	na
Ra-228	9.2E+09	9.1E+09	-1.2
Ac-228			
Th-228			
Ra-224			
Rn-220			
Pb-212			
Bi-212			
Tl-208			
Ac-227	1.2E+10	1.2E+10	1.4
Th-227			
Ra-223			
Pb-211			
Bi-211			
Tl-207			
U-232	1.5E+05	1.5E+05	0.0
Th-228			
Ra-224			
Rn-220			
Pb-212			
Bi-212			
Tl-208			
Am-242m	3.0E+09	3.1E+09	4.8
Am-242			
Cm-242			
Np-238			
Pu-238			
U-234			
Ra-226			
Cm-243	1.3E+13	1.2E+13	-4.9
Pu-239			
U-235			
Pa-231			
Ac-227			

[a] From Table 4-8.

Table B-2. Comparison of hand calculated limits on inventory with spreadsheet calculation of same limits for 1,000-year intruder resident scenario

	Inventory Limit		
	Excel Spreadsheet	Hand Calculation[a]	Percent
Radionuclide	Ci/vault	Ci/vault	difference
Al-26	1.6E+02	1.6E+02	-2.3
Nb-94	8.7E+02	8.5E+02	-1.8
Sn-126	1.0E+03	1.0E+03	-1.8
Sb-126m			
Sb-126			
Cs-137	3.5E+13		
Ba-137m			
Ra-226	4.8E+02	4.9E+02	1.3
Rn-222			
Pb-214			
Bi-214			
Pb-210			
Po-210			
Th-229	8.7E+03	9.0E+03	3.0
Ra-225			
Ac-225			
Fr-221			
Bi-213			
Tl-209			
Th-230	9.0E+02	9.0E+02	0.5
Ra-226			
Pb-210			
Po-210			
Th-232	1.6E+02	1.6E+02	-2.1
Ra-228			
Ac-228			
Th-228			
Ra-224			
Rn-220			
Pb-212			
Bi-212			
Tl-208			
Pa-231	2.0E+04	2.0E+04	-0.7
Ac-227			
Th-227			
Ra-223			
Pb-211			
Bi-211			
Tl-207			

Table B-2. (continued)

Radionuclide	Inventory Limit Excel Spreadsheet Ci/vault	Inventory Limit Hand Calculation[a] Ci/vault	Percent difference
U-233	8.8E+04	9.1E+04	3.0
Th-229			
U-234	1.9E+05	1.9E+05	2.4
Th-230			
Ra-226			
Pb-210			
Po-210			
U-235	4.4E+05	4.4E+05	-0.9
Th-231			
Pa-231			
Ac-227			
U-236	3.3E+09	no limit	na
Th-232			
U-238	6.7E+04	6.6E+04	-1.8
Th-234			
Pa-234m			
Pa-234			
U-234			
Th-230			
Ra-226			
Pb-210			
Po-210			
Np-237	5.8E+04	5.6E+04	-3.4
Pa-233			
Th-229			
Pu-238	6.5E+08	6.4E+08	-2.0
U-234			
Th-230			
Ra-226			
Pb-210			
Po-210			
Pu-239	8.3E+11	no limit	na
U-235			
Pa-231			
Ac-227			
Pu-240	3.4E+13	no limit	na
U-236			
Th-232			

Table B-2. (continued)

Radionuclide	Inventory Limit Excel Spreadsheet Ci/vault	Hand Calculation[a] Ci/vault	Percent difference
Pu-241	1.1E+10	1.1E+10	1.3
Am-241			
Np-237			
Th-229			
Pu-242	4.3E+11	no limit	na
U-238			
Ra-226			
Pu-244	4.0E+03	4.0E+03	-0.1
Np-240m			
Am-241	3.6E+08	3.5E+08	-2.7
Np-237			
Th-229			
Am-242m	4.0E+07	4.2E+07	3.8
Am-242			
Cm-242			
Np-238			
Pu-238			
U-234			
Ra-226			
Am-243	2.8E+05	2.7E+05	-2.0
Np-239			
Pu-239			
Cm-242	1.3E+11	1.3E+11	-0.5
Pu-238			
U-234			
Th-230			
Ra-226			
Pb-210			
Po-210			
Cm-243	6.9E+14	no limit	na
Pu-239			
U-235			
Pa-231			
Ac-227			
Cm-244	1.3E+16	no limit	na
Pu-240			
U-236			
Th-232			

Table B-2. (continued)

Radionuclide	Inventory Limit Excel Spreadsheet Ci/vault	Hand Calculation[a] Ci/vault	Percent difference
Cm-245	6.7E+06	6.6E+06	-1.2
Pu-241			
Am-241			
Np-237			
Th-229			
Cm-246	4.9E+14	no limit	na
Pu-242			
U-238			
Ra-226			
Cm-247	2.0E+04	2.0E+04	2.4
Pu-243			
Am-243			
Np-239			
Pu-239			
Cm-248	4.7E+08	4.6E+08	-1.2
Pu-244			
Bk-249	6.0E+07	5.8E+07	-2.9
Cf-249			
Cm-245			
Pu-241			
Am-241			
Np-237			
Th-229			
Cf-249	1.5E+05	1.5E+05	-0.3
Cm-245			
Pu-241			
Am-241			
Np-237			
Th-229			
Cf-250	1.9E+17	no limit	na
Cm-246			
Pu-242			
U-238			
Ra-226			
Cf-251	2.1E+06	2.0E+06	-3.8
Cm-247			
Am-243			
Cf-252	6.1E+13	6.0E+13	-1.1
Cm-248			
Pu-244			

[a] From Table 4-9.

Table B-3. Comparison of hand calculated limits on inventory with spreadsheet calculations of same limits for 10,000-year intruder resident scenario

	Inventory Limits		
	Excel Spreadsheet	Hand Calculations[a]	Percent
Radionuclide	Ci/vault	Ci/vault	difference
Th-230	3.4E+02	3.4E+02	-0.2
Ra-226			
Pb-210			
Po-210			
U-233	1.3E+04	1.4E+04	5.4
Th-229			
U-234	4.7E+03	4.6E+03	-2.9
Th-230			
Ra-226			
Pb-210			
Po-210			
U-235	9.2E+04	9.2E+04	0.3
Th-231			
Pa-231			
Ac-227			
U-236	3.3E+08	no limit	na
Th-232			
U-238	5.7E+04	5.6E+04	-2.6
Th-234			
Pa-234m			
Pa-234			
U-234			
Th-230			
Ra-226			
Pb-210			
Po-210			
Np-237	5.2E+04	5.3E+04	1.3
Pa-233			
Th-229			
Pu-238	1.4E+07	1.3E+07	-4.8
U-234			
Th-230			
Ra-226			
Pb-210			
Po-210			
Pu-239	1.8E+10	1.8E+10	-0.9
U-235			
Pa-231			
Ac-227			

Table B-3. (continued)

Radionuclide	Inventory Limits Excel Spreadsheet Ci/vault	Hand Calculations[a] Ci/vault	Percent difference
Pu-240	3.1E+12	no limit	na
U-236			
Th-232			
Pu-241	7.9E+09	7.9E+09	0.6
Am-241			
Np-237			
Th-229			
Pu-242	2.5E+10	no limit	na
U-238			
Ra-226			
Pu-244	4.0E+03		
Np-240m			
Am-241	2.6E+08	2.5E+08	-4.8
Np-237			
Th-229			
Am-242m	9.6E+06	1.0E+07	3.8
Am-242			
Cm-242			
Np-238			
Pu-238			
U-234			
Ra-226			
Cm-242	2.6E+09	2.6E+09	-0.5
Pu-238			
U-234			
Th-230			
Ra-226			
Pb-210			
Po-210			
Cm-243	1.6E+13	1.6E+13	0.7
Pu-239			
U-235			
Pa-231			
Ac-227			
Cm-244	1.1E+15	no limit	na
Pu-240			
U-236			
Th-232			

Table B-3. (continued)

Radionuclide	Inventory Limits Excel Spreadsheet Ci/vault	Hand Calculations[a] Ci/vault	Percent difference
Cm-245	9.2E+06	9.1E+06	-1.2
Pu-241			
Am-241			
Np-237			
Th-229			
Cm-246	7.0E+12	no limit	na
Pu-242			
U-238			
Ra-226			
Cm-247	1.9E+04	1.9E+04	1.2
Pu-243			
Am-243			
Np-239			
Pu-239			
Cm-248	4.7E+07	4.6E+07	-1.2
Pu-244			
Bk-249	8.8E+10	8.2E+10	-6.8
Cf-249			
Cm-245			
Pu-241			
Am-241			
Np-237			
Th-229			
Cf-249	2.2E+08	2.1E+08	-5.0
Cm-245			
Pu-241			
Am-241			
Np-237			
Th-229			
Cf-250	2.6E+15	no limit	na
Cm-246			
Pu-242			
U-238			
Ra-226			
Cf-251	2.9E+08	2.8E+08	-1.9
Cm-247			
Am-243			
Cf-252	6.2E+12	6.1E+12	-0.9
Cm-248			
Pu-244			

[a] From Table 4-10.

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