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Revision 1

Guidance for Conducting Technical Analyses for 10 CFR Part 61

Draft Report for Comment

Office of Nuclear Material Safety and Safeguards

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ABSTRACT

This document provides guidance on conducting technical analyses (i.e., performance assessment, inadvertent intruder assessment, operational safety assessment, site stability assessment, and performance period analyses) to demonstrate compliance with the performance objectives in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 61, "Licensing Requirements for Land Disposal of Radioactive Waste." This document provides implementing guidance for the amendments to 10 CFR Part 61 that are detailed in the U.S. Nuclear Regulatory Commission's (NRC's) "Integrated Low-Level Radioactive Waste Disposal" (ILLRWD) rule (NRC, 2026). Though this guidance emphasizes technical analyses, it also provides guidance on many other topics relevant to technical analyses.

The information in this document is intended to supplement existing guidance on low-level radioactive waste on issues pertinent to conducting technical analyses to demonstrate compliance with the performance objectives. This document provides detailed guidance in new areas, such as the inadvertent intruder assessment and analyses for the compliance and performance periods. This guidance discusses approaches to selecting the compliance period (1,000 years or 10,000 years after disposal site closure, depending on the quantities of long-lived radionuclides disposed), and covers the performance period analyses (beyond 10,000 years) that should be performed for analysis of long-lived waste.

This guidance should facilitate implementation of the rule by existing licensees and potential applicants and assist regulators in reviewing the technical analyses. This guidance applies to waste streams disposed at 10 CFR Part 61 low-level waste disposal facilities, including significant quantities of depleted uranium, blended waste, and Greater-Than-Class-C (GTCC) waste. Additional topics covered in this document include (1) the use of defense-in-depth to enhance the resiliency of the disposal site to meet performance objectives, (2) the identification and screening of the features, events, and processes to develop scenarios and conceptual models for technical analyses (the scope of the analyses), (3) the use of the waste classification tables or the results of the technical analyses to develop generic or site-specific waste acceptance criteria, and (4) the development of a safety case for a low-level waste disposal facility. This guidance includes a discussion of specific considerations for GTCC waste disposal in both a near-surface disposal facility and a specialized land disposal facility, provides examples for determining the appropriate compliance period for the technical analyses based on the quantities of long-lived radionuclides, and the criteria for disposal facility licensees with licenses in effect before the effective date of the final ILLRWD rule that do not plan to dispose of significant quantities of long-lived radionuclides and do not plan to apply to the NRC to dispose of GTCC waste after the effective date of this rule.

PREFACE

The U.S. Nuclear Regulatory Commission (NRC) published the rule, "Integrated Low-Level Radioactive Waste Disposal," (ILLRWD) (NRC, 2026) with amendments to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 61, "Licensing Requirements for Land Disposal of Radioactive Waste." This rule includes detailed requirements for 10 CFR Part 61 licensees and applicants for performing technical analyses to demonstrate compliance with Subpart C, "Performance Objectives" (i.e., 10 CFR 61.41, "Protection of the general population from releases of radioactivity," through 10 CFR 61.44, "Stability of the disposal site after closure"). Appendix A to this document highlights revisions to the 10 CFR Part 61 performance objectives that were made in the ILLRWD rule and identifies the new requirements, particularly relating to technical analyses, in comparison to the original 1982 requirements that may continue to apply to existing sites that meet the criteria in 10 CFR 61.1(b). The most significant changes in the current rule revisions are the adoption of requirements for an inadvertent intruder assessment and specific technical requirements for the disposal of Greater-Than-Class-C (GTCC) waste in both near-surface disposal facilities and specialized land disposal facilities. These changes allow the revised 10 CFR Part 61 regulations to be more flexible to account for different wastes while maintaining technical rigor and continuing to ensure the adequate protection of public health and safety.

The purpose of this guidance document is to provide licensees and applicants with the tools to develop high-quality technical analyses—the performance assessment, inadvertent intruder assessment, operational safety assessment, site-stability assessment, and performance period analyses. NRC or Agreement State regulators may use this document to assist in reviewing these technical analyses. This document refers to the reviewing authority as "the NRC" or "the NRC staff," but these terms should be interpreted to mean Agreement State regulators as well, if applicable. The document provides methods and information the NRC finds generally acceptable to demonstrate the licensee's and applicant's ability to meet the regulatory requirements. This document uses the term "licensees" to refer to licensees or license applicants, as applicable. In instances where a regulatory requirement is being referenced, the reader should refer to the regulation to determine whether the guidance applies in a particular case.

In addition to supporting development of the technical analyses, licensees may use this document to assist in (1) identifying where defense-in-depth protections could enhance the resiliency of the disposal site to meet performance objectives and (2) developing waste acceptance criteria for their disposal sites and acceptable processes for waste characterization and certification.

In this document, the NRC provides reference material for licensees yet allows flexibility to adapt this guidance to the specifics for each low-level waste disposal site. The appendices to this document provide additional details and examples, such as (1) hazard maps in Appendix B that may assist applicants and reviewers in determining site suitability for a proposed disposal site, and (2) lists of example features, events, and processes in Appendix C that may assist licensees in defining the scope of the technical analyses. Appendix G includes specific disposal considerations for GTCC waste, and Appendix H details considerations for determining the appropriate compliance period based on the quantities of long-lived radionuclides disposed.

While this guidance document provides acceptable methodologies that licensees can use to perform the analyses, which are then incorporated, as appropriate, into the licensing basis for the facility, the use of this guidance document is optional, and licensees may propose alternative means of complying with NRC's regulations in 10 CFR Part 61.

To develop a succinct document, the NRC staff has written this guidance with the assumption that the reader has some level of proficiency in conducting technical analyses. For example, the NRC staff describes the application of model support and model abstraction for the performance assessment and inadvertent intruder assessment and references existing documents for background information on these terms. Therefore, the NRC staff recommends that the reader become familiar with documents such as NUREG-1573, “A Performance Assessment Methodology for Low-Level Radioactive Waste Disposal Facilities,” issued October 2000 (NRC, 2000a), and other guidance documents referenced in Section 1.2, as background for this document.

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ACRONYMS AND ABBREVIATIONS

Am	americium
ACAP	Alternative Cover Assessment Project
ADAMS	Agencywide Documents Access and Management System
ASTM	American Society for Testing and Materials
Bq	becquerel
BTP	branch technical position
BTP CA	"Concentration Averaging and Encapsulation Branch Technical Position, Revision 1" (NRC, 2015a)
C	carbon
Cf	californium
CFR	*Code of Federal Regulations*
Ci	curie
Cl	chlorine
Cm	curium
CNWRA	Center for Nuclear Waste Regulatory Analyses
Cs	cesium
CSDMS	Community Surface Dynamics Modeling System
DOE	U.S. Department of Energy
DQO	data quality objective
DU	depleted uranium
EC	European Commission
EPA	U.S. Environmental Protection Agency
ET	evapotranspiration
FEPs	features, events, and processes
FGR	Federal Guidance Report
FR	*Federal Register*
ft	foot
g	gram
GTCC	Greater than Class C
HLW	high-level waste
I	iodine
IAEA	International Atomic Energy Agency
ICRP	International Commission on Radiological Protection
IEEE	Institute of Electrical and Electronics Engineers
ILW	intermediate-level waste
ILLRWD	Integrated Low-Level Radioactive Waste Disposal
IPCC	Intergovernmental Panel on Climate Change
kBq	kilobecquerel
K_d	distribution coefficient
Kr	krypton
LiDAR	Light Detection and Ranging
LLW	low-level radioactive waste
LLW-LL	low-level waste—long lived
m	meter
mm	millimeter

mrem	millirem
mSv	millisievert
nCi	nanocuries
NCRP	National Council on Radiation Protection and Measurements
NEA	Nuclear Energy Agency
NOAA	National Oceanic and Atmospheric Administration
Np	neptunium
NRC	U.S. Nuclear Regulatory Commission
NYSDEC	New York State Department of Environmental Conservation
OAT	one-at-a-time
PA	performance assessment
PMF	probable maximum flood
PMP	probable maximum precipitation
PRA	probabilistic risk assessment
Pu	plutonium
QA	quality assurance
QC	quality control
Rn	radon
SOF	sum of fractions
Sr	strontium
SRM	staff requirements memorandum
Tc	technetium
TRU	transuranic waste
U	uranium
USGS	U.S. Geological Survey
UWM	Uniform Low-Level Radioactive Waste Manifest
yr	year

In this document, the first use of a technical term that is defined in the glossary is shown in *italics.*

1.0 INTRODUCTION

The U.S. Nuclear Regulatory Commission (NRC) establishes licensing requirements, performance objectives, and technical criteria for the *disposal* of commercial *low-level radioactive waste* (LLW) in *land disposal facilities*. Land disposal facilities include both *near-surface disposal facilities* and *specialized land disposal facilities*. These requirements can be found in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 61, "Licensing Requirements for Land Disposal of Radioactive Waste," and in the final rule in the *Federal Register* (FR) (47 FR 57446) (NRC, 1982a).

The NRC previously issued this guidance in NUREG-2175, "Guidance for Conducting Technical Analyses for 10 CFR Part 61, Draft Report for Comment," issued March 2015 (NRC, 2015c), as implementing guidance for the Low-Level Radioactive Waste Disposal proposed rule. Both the proposed rule and draft guidance were issued for a 120-day public comment period (NRC, 2015d). After considering and responding to stakeholder comments, the NRC staff developed final rule language and submitted it to the Commission for review in SECY-16-0106, "Final Rule: Low-Level Radioactive Waste Disposal (10 CFR Part 61)," dated September 15, 2016 (NRC, 2016c). The staff also developed the interim draft final NUREG-2175 in October 2016 (NRC, 2016b). Appendix F to the 2016 draft final NUREG-2175 contains responses to the public comments received on the 2015 draft NUREG-2175.

The Commission directed the staff on the path forward in Staff Requirements Memorandum (SRM)-SECY-16-0106, dated September 8, 2017 (NRC, 2017a). The Commission directed the staff to make substantive revisions to the draft final rule and subsequently republish it as a supplemental proposed rule for a 90-day public comment period. The Commission also directed the staff to revise NUREG-2175 and publish the guidance for comment along with the supplemental proposed rule.

On a separate track, the NRC staff issued the "Draft Regulatory Basis for Disposal of Greater-Than-Class C (GTCC) and Transuranic Waste" for public comment on July 22, 2019 (NRC, 2019), which discusses a possible rulemaking to address the disposal requirements for near-surface disposal of GTCC waste. However, on October 21, 2020, in SECY-20-0098, "Path Forward and Recommendations for Certain Low-Level Radioactive Waste Disposal Rulemakings," the staff provided the Commission with options and a recommendation for the path forward on these separate activities (NRC, 2020c). The staff recommended combining the Low-Level Radioactive Waste Disposal supplemental proposed rule with the GTCC proposed rulemaking into one consolidated and integrated rulemaking for 10 CFR Part 61. On April 5, 2022, the Commission issued SRM-SECY-20-0098 (NRC, 2022a), approving the staff's recommendation.

The Integrated Low-Level Radioactive Waste Disposal (ILLRWD) rule (NRC, 2026) amended the NRC's regulations to revise the licensing requirements for LLW disposal and consolidate and integrate criteria for licensing the land disposal of GTCC waste. The ILLRWD rule ensures that the LLW streams that are significantly different from those considered during the development of 10 CFR Part 61, such as depleted uranium (DU), would continue to be disposed

of safely and meet the performance objectives for land disposal of LLW.[1] This document provides implementing guidance for the amendments to 10 CFR Part 61 that are detailed in the ILLRWD rule.

This guidance document is intended to support the implementation of the requirements for *technical analyses* and waste acceptance to demonstrate compliance with 10 CFR Part 61 performance objectives. This guidance supplements, rather than replaces, previous guidance (e.g., NUREG-1200, Revision 3, "Standard Review Plan for the Review of a License Application for a Low-Level Radioactive Waste Disposal Facility," issued April 1994 (NRC, 1994); NUREG-1573, "A Performance Assessment Methodology for Low-Level Radioactive Waste Disposal Facilities: Recommendations of NRC's Performance Assessment Working Group," issued October 2000 (NRC, 2000a)). This document applies to all *waste streams* disposed at a 10 CFR Part 61 disposal facility, while providing specific considerations for *long-lived wastes*, GTCC waste, and blended wastes (see Section 1.2).

1.1 Background

An integrated systems approach is emphasized in 10 CFR Part 61 for the disposal of commercial LLW, including site selection, disposal facility design and operation, minimum *wasteform* requirements, and disposal facility closure. To lessen the burden on society over the long periods of time contemplated for the control of radioactive material and thus lessen reliance on *institutional controls*, 10 CFR Part 61 emphasizes passive, rather than active, systems to reduce and slow radioactive releases to the environment (i.e., outside the *disposal units*).

To grant a license, the NRC must conclude that there is reasonable assurance that the performance objectives of 10 CFR Part 61, Subpart C, "Performance Objectives," will be met. To demonstrate that they will meet the performance objectives, 10 CFR Part 61 license applicants need to prepare technical analyses. The technical analyses required for *licensees*[2] to demonstrate that the performance objectives will be met are specified in 10 CFR 61.13, "Technical analyses," and include *performance assessment*, *inadvertent intruder assessment*, *site stability assessment*, *operational safety assessment*, and, in some cases *performance period analyses*.

Licensees must meet specific technical requirements to ensure the safe disposal of LLW. These requirements are specified in 10 CFR Part 61, Subpart D, "Technical Requirements for Land Disposal Facilities," and include, among others, requirements for waste acceptance that are specified in 10 CFR 61.58, "Waste acceptance." The waste acceptance requirements, coupled with the *disposal site* and facility design, are intended to provide reasonable assurance that the performance objectives of Subpart C will be met.

The regulatory requirements in 10 CFR Part 61, if met, ensure adequate protection of public health and safety during the operation of a commercial LLW disposal facility. The regulations in 10 CFR Part 61 are performance-based, and the technical criteria are written in relatively general terms. This allows license applicants flexibility to demonstrate how their proposals meet the respective performance objectives for the specific land disposal method selected.

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[1] Previous regulatory practice was to address the disposal of these waste streams by license conditions implemented by Agreement State programs for existing disposal facilities.

[2] This document uses the term "licensees" to refer to licensees or license applicants, as applicable. In instances where a regulatory requirement is being referenced, the reader should refer to the regulation to determine whether the guidance applies in a particular case.

1.1.1 Low-Level Waste

LLW includes items that have become contaminated with radioactive material or have become radioactive through exposure to neutron radiation. This waste typically consists of contaminated protective shoe covers and clothing, rags, mops, filters, reactor water treatment residues, equipment and tools, luminous dials, medical tubes, swabs, injection needles, syringes, and laboratory animal carcasses and tissues, among other items. The radioactivity levels of LLW have a large range, from just above background levels found in nature to very high levels found in nonfuel components from inside the reactor vessel at a nuclear power plant. LLW is defined in 10 CFR 61.2, "Definitions," as follows:

> Radioactive waste not classified as high-level radioactive waste (HLW), spent nuclear fuel, or byproduct material as defined in paragraphs (2), (3), and (4) of the definition of *byproduct material* set forth in 20.1003 of this chapter.

LLW is defined by how it was generated (origin based) rather than technical characteristics (i.e., not concentration based). Previously, this definition included a prohibition on transuranic waste (TRU), which is a term used to describe waste with alpha-emitting transuranic radionuclides with half-lives greater than 5 years, in concentrations greater than 3.7 kilobecquerels (kBq) per gram (g) (100 nanocuries (nCi)/g).[3] The original regulations allowed only for the disposal as LLW of materials containing less than 3.7 kBq/g (100 nCi/g) of transuranic radionuclides. However, in SRM-SECY-15-0094, "Staff Requirements—SECY-15-0094—Historical and Current Issues Related to Disposal of Greater-Than-Class C Low-Level Radioactive Waste," dated December 22, 2015 (NRC, 2015e), the Commission approved the staff recommendation to revise the definition of LLW to no longer exclude TRU. Therefore, the ILLRWD rule includes the revised definition of LLW above (NRC, 2026).

1.1.2 Performance Objectives

The performance objectives for a LLW disposal facility are contained in 10 CFR Part 61, Subpart C. As stated in 10 CFR 61.40, "General requirement," land disposal facilities must be sited, designed, operated, closed, and controlled after closure to provide reasonable assurance that exposures to humans are within the limits established in the performance objectives in 10 CFR 61.41, "Protection of the general population from releases of radioactivity," through 10 CFR 61.44, "Stability of the disposal site after closure." The general requirement covers all timeframes in the life cycle of waste disposal (e.g., during operations and postclosure).

[3] The use of 100 nCi/g is mixed units (metric/English). The conversion factors are 1 nCi = 37 Bq and 1 kg = 2.205 lb.

Several of the performance objectives were revised through the ILLRWD rule to preserve the original 10 CFR Part 61 rule language for disposal facility licensees that meet the criteria in 10 CFR 61.1(b), "Purpose and Scope," as follows:

(1) Licensees need not comply with the requirements in §§ 61.10(c), 61.13(a) through 61.13(e), 61.24(l), 61.41(a) and (b), 61.42(a) and (b), 61.50 (a) and (b), and 61.58, if the following criteria are met:

(i) the land disposal facility license was originally issued before [30 DAYS AFTER DATE OF PUBLICATION OF THE FINAL RULE IN THE *FEDERAL REGISTER*]; and

(ii) the licensee does not accept Greater-Than-Class-C waste or a *significant quantity* of long-lived radionuclides. For purposes of this paragraph, an amount greater than or equal to 10 metric tons of depleted uranium is considered a significant quantity of long-lived radionuclides.

(2) Licensees who meet the criteria of § 61.1(b)(1)(i) and (ii) and who choose not to comply with the requirements in §§ 61.10(c), 61.13(a) through (e), 61.24(l), 61.41(a) and (b), 61.42(a) and (b), 61.50(a) and (b), and 61.58 must instead comply with §§ 61.13(f), 61.41(c), 61.42(c), and 61.50(c).

Section 1.4 provides additional details on the requirements for licensees that meet the criteria in 10 CFR 61.1(b). After facility operations, the performance objective in 10 CFR 61.41 requires that the general population be protected from releases of radioactivity:

- The performance objective in 10 CFR 61.41(a) provides an annual *dose* limit that refers to the radiological dose criteria for unrestricted use in 10 CFR 20.1402 to any member of the public for the *compliance period*. Licensees must manage releases of radioactivity to the general environment consistent with 10 CFR 20.1101(b).

- The performance objective in 10 CFR 61.41(b) requires that releases of radioactivity from the disposal site must be effectively managed during the *performance period*.

- The performance objective in 10 CFR 61.41(c) provides the original 10 CFR Part 61 rule language (NRC, 1982a) for existing licensees that meet the criteria in 10 CFR 61.1(b). This performance objective provides an annual dose limit of 0.25 millisievert (mSv) (25 millirem (mrem)) to the whole body, 75 mrem to the thyroid, and 25 mrem to any other organ of any *member of the public*.

The performance objective in 10 CFR 61.42, "Protection of individuals from inadvertent intrusion," requires that an *inadvertent intruder* must be protected during the compliance period after active institutional controls are removed:

- The performance objective in 10 CFR 61.42(a)(1) provides an annual dose limit of 5 mSv (500 mrem) for the compliance period for near-surface disposal. Section 10 CFR 61.42(a)(2) provides that for waste in a specialized land disposal facility, the annual dose to any inadvertent intruder within the compliance period from any onsite

releases (e.g., radon emanation, groundwater contamination) must not exceed 5 mSv (500 mrem).

- The performance objective in 10 CFR 61.42(b) requires exposures to an inadvertent intruder during the performance period to be effectively managed.

- The performance objective in 10 CFR 61.42(c) provides the original 10 CFR Part 61 language (NRC, 1982a) for existing licensees that meet the criteria in 10 CFR 61.1(b). This performance objective does not include the 500 mrem dose limit.

The regulations at 10 CFR 61.41 and 10 CFR 61.42 require a demonstration of protection beyond the time of closure of the disposal facility. The length of time specified in these requirements is defined in 10 CFR 61.2 as follows:

> *Compliance period* means the time from the completion of site closure to 1,000 years after site closure for disposal sites that do not contain significant quantities of long-lived radionuclides. For disposal sites that contain significant quantities of long-lived radionuclides, the compliance period ends 10,000 years after closure of the disposal site.
>
> *Performance period* is the timeframe after the compliance period during which waste that contains significant quantities of long-lived radionuclides remains a radiological hazard.
>
> *Significant quantities* of long-lived radionuclides means an amount (volume or mass) and concentration accepted for disposal [after the effective date of the final ILLRWD Rule] that could, if released, result in the performance objectives of subpart C of this part not being met.

The performance objective in 10 CFR 61.43, “Protection of individuals during operations,” requires protection of individuals during operations. Compliance with 10 CFR 61.43 is determined largely through compliance with the standards for radiation protection set forth in 10 CFR Part 20, “Standards for Protection Against Radiation.” Some wastes, such as GTCC waste, may pose greater operational *hazards* than Class A, B, or C waste. Appendix G to this document discusses specific considerations for GTCC waste disposal in both near-surface and specialized land disposal facilities. Section G.6.1 discusses the information licensees should consider, and *reviewers* should evaluate, for performing operational safety assessments for GTCC waste disposal.

The performance objective set forth in 10 CFR 61.44 requires that the licensee demonstrate that the land disposal facility will be sited, designed, used, operated, and closed to achieve long-term *stability* of the disposal site. Stability during the institutional control period eliminates, to the extent practicable, the need for ongoing active *maintenance* of the disposal site following closure so that only *surveillance*, *monitoring*, or minor custodial care is required. No maintenance is allowed to be credited after license termination, though maintenance may be necessary by the site custodian if unforeseen changes were to occur. The regulations in 10 CFR 61.44 do not specifically mention the demonstration of long-term stability throughout the compliance period (which could be up to 10,000 years), but the analyses that may be necessary are performed in the context of demonstration of meeting the performance objectives detailed in 10 CFR 61.41 and 10 CFR 61.42.

Figure 1-1 depicts the timeframes discussed in 10 CFR Part 61 and this document that licensees that do not meet the criteria in 10 CFR 61.1(b) should consider.

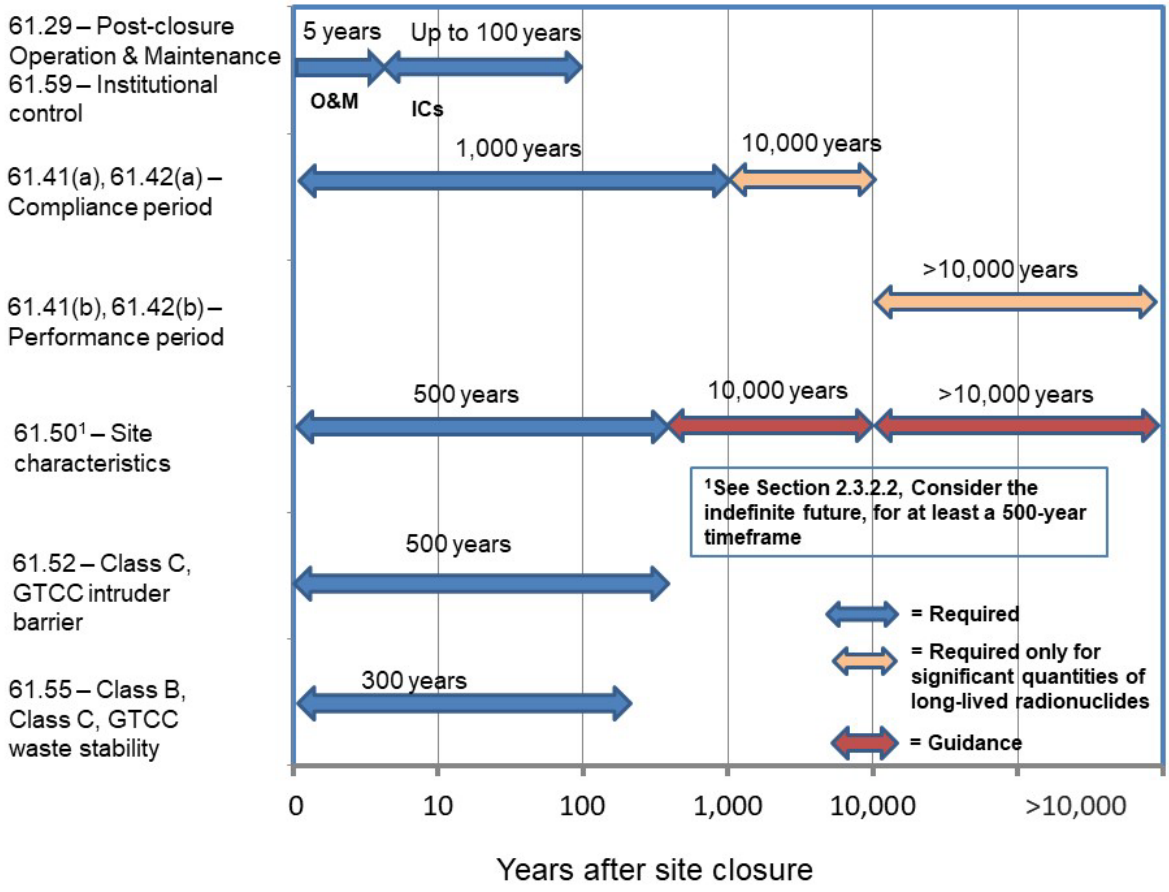


Figure 1-1 Timeframes to be Considered for 10 CFR Part 61

1.1.3 Safety Case

In Section 10 CFR 61.2, the NRC defines a *safety case* as a high-level evaluation of the information and analyses that support the licensee's demonstration that the land disposal facility will be constructed and operated safely. The safety case summarizes the basis that the disposal site will be capable of isolating waste and limiting releases to the environment. The safety case is a component of a license application that describes the safety relevant aspects of the site; the facility design; and the managerial, engineering, regulatory, and institutional controls. This includes the technical analyses discussed in Section 1.1.5, as well as information on *defense-in-depth* and supporting evidence and reasoning on the strength and reliability of the technical analyses and the assumptions made therein.

As required in 10 CFR 61.10(c), "Content of application":

> The application must include the safety case, which demonstrates that the land disposal facility will be constructed and operated safely, provides a summary of the safety basis

that the disposal site will meet the performance objectives in subpart C of this part, and describes the defense-in-depth protections that enhance the resiliency of the facility in complying with the performance objectives specified at 61.41 and 61.43.

Much of the information provided in a license application comprises the content of a safety case. Information included in a license application must include the general and technical information required by 10 CFR 61.11, "General information," and 10 CFR 61.12, "Specific technical information"; the technical analyses required in 10 CFR 61.13, "Technical analyses"; and institutional, financial, and other information required in 10 CFR 61.14, "Institutional information," through 10 CFR 61.16, "Other information." Thus, a safety case for a land disposal facility covers the suitability of the site and the design, construction, and operation of the facility; the assessment of radiation *risks*; and assurance of the adequacy and quality of the safety-related work associated with the disposal facility. The NRC staff provides detailed guidance on the contents of a license application in NUREG-1200. The guidance in NUREG-1200 is supplemented by guidance on conducting the technical analyses in NUREG-1573 and is updated in Chapters 2.0–6.0 of this document. Chapters 7.0 and 8.0 provide guidance on defense-in-depth protections and developing *waste acceptance criteria*, respectively.

The purpose of a safety case is to inform the decision whether to grant a license for a land disposal facility and provide a summary of the safety basis that the land disposal facility will be designed, constructed, operated, and closed safely. The safety case provides assurance to the public that the facility will be designed, constructed, operated, and closed safely (IAEA, 2012).

Licensing decisions are based on whether there is reasonable assurance that the performance objectives can be met. Defense-in-depth protections, such as siting, wasteforms, radiological source-term, engineered *features*, and natural features of the disposal site, combined with technical analyses and scientific judgment support the safety case for licensing an LLW disposal facility. The insights derived from technical analyses include supporting evidence and reasoning on the strength and reliability of the layers of defense relied upon in the safety case. These insights provide input for making regulatory decisions. The safety case must conclude that public health and safety will be adequately protected from exposures resulting from the disposal of LLW (including long-lived LLW). The safety case also serves to enhance communication among stakeholders. The NRC staff recommends that licensees include a plain language description of the following in their safety cases:

- Strategy for achieving safe disposal of radioactive waste—The overall management strategy for the various activities required in the planning, operation, and closure of a land disposal facility, including siting and characterization, facility and disposal site design, development of the technical analyses, operations, waste acceptance, environmental monitoring, and institutional control. The strategy should describe approaches for managing significant uncertainties to ensure the protection of public health and safety.

- The disposal site and facility—The description of the disposal site and facility should include the relevant information and knowledge about the disposal system and should provide the basis for the technical analyses. The description should summarize the information required in 10 CFR 61.12.

- The technical analyses demonstrating performance objectives will be met—The description of the technical analyses should summarize the performance assessment,

inadvertent intruder assessment, site stability assessment, operational safety assessment, and, where appropriate performance period analyses. These analyses are required by 10 CFR 61.13 to demonstrate compliance with the performance objectives. The description should include key findings of each of the technical analyses, including significant associated uncertainties.

- Strategy for institutional control of the disposal site—The institutional control strategy should summarize the institutional information required by 10 CFR 61.14.

- Financial qualifications of the licensee—The description should summarize the financial information required by 10 CFR 61.15, "Financial information."

- Other information—Depending upon the types of waste to be disposed of and the design and proposed operation of the land disposal facility, the description may need to summarize other information required by 10 CFR 61.16.

- Safety basis—The safety basis should draw together the key findings from the technical analyses to highlight the main evidence, analyses, and information that quantify and support the claim that the land disposal facility will ensure protection of public health and safety. The safety arguments should include an evaluation of uncertainties and unresolved issues and discuss planned steps to resolve them. The safety arguments should also describe additional evidence and arguments that may not have been evaluated in the technical analyses but complement their findings.

- Defense-in-depth—The description should include information on the defense-in-depth protections that enhance the resiliency of the facility in complying with the performance objectives specified in 10 CFR 61.41 and 10 CFR 61.43.

The safety case for a land disposal facility will evolve over time as new information is gained during the various phases[4] of the facility's development and operation. Therefore, as new information that could significantly impact the safety of the facility is learned, the safety case should be updated. In 10 CFR 61.27, "Application for renewal or closure," applications for renewal of a license must be filed in accordance with 10 CFR 61.10, "Content of application," through 10 CFR 61.16 and 10 CFR 61.20, "Filing and distribution of application.". Therefore, licensees renewing their licenses should update their safety case in accordance with 10 CFR 61.10 and 10 CFR 61.13(a) through (e). Licensees that meet the criteria in 10 CFR 61.1(b) need not comply with 10 CFR 61.10(c) and are, therefore, not required to develop a safety case.

According to 10 CFR 61.28(a), the application for closure of a licensed land disposal facility must include a final revision to the safety case, if applicable, and a final disposal site closure plan that includes (among other items) revisions to the technical analyses. Specifically, it states the following:

[4] A low-level waste disposal facility will be operated and closed in phases, or distinct periods of time. The operational period is generally 30 to 70 years. After active disposal operations are completed, the facility will be prepared for closure and observed for a period of time (approximately 5 years). If the closure system is observed to be stable, then the facility will enter an institutional control period during which the facility will be monitored and observed but maintenance operations are not anticipated. After the institutional control period, the facility will enter the postclosure period.

(3) Any revisions to the technical analyses necessary to support site closure; such revisions must consider the waste disposed during operations and reflect significant changes to the human activities occurring in and around the site.

The extent of the final revisions to the safety case may vary depending on the licensee's operation and closure of the land disposal facility and the amount of new information that is developed that could significantly impact the demonstration of the facility safety.

1.1.4 Defense-in-Depth Protections

In 10 CFR 61.2, defense-in-depth is defined as the use of multiple independent and, where possible, redundant layers of defense against release of radioactive material such that no single layer, no matter how robust, is exclusively relied upon. Defense-in-depth includes, but is not limited to, the use of siting, wasteforms and radionuclide content, engineered features, geologic features of the land disposal facility, and management and operational controls to enhance the resiliency of the disposal site to meet performance objectives. Defense-in-depth may be used to mitigate the effects of uncertainties identified in the technical analyses.

Licensees applying defense-in-depth protections for land disposal facilities should recognize differences between operating facilities and closed land disposal facilities. While waste is being disposed and before a land disposal facility is closed, defense-in-depth protections can be provided by active or passive safety systems commensurate with the hazard and complexity of the activities, as is the case in other operating nuclear facilities. A closed disposal site is a passive system that does not rely on active safety systems, and assessment of its safety over long timeframes is best evaluated through consideration of the relative likelihood of threats to its integrity and long-term performance. With respect to the long-term performance of the disposal site, and in particular for the disposal of long-lived radionuclides, defense-in-depth is provided through the diversity and capabilities of the components and attributes of the disposal site (e.g., wasteform, container, engineered features, depth of the disposal unit below the land surface, hydrologic and geochemical characteristics).

Chapter 7.0 describes the information that a licensee should consider to ensure that a land disposal facility incorporates defense-in-depth protections during the operational and postclosure phases of the land disposal facility life cycle. The regulation in 10 CFR 61.10(c) requires licensees to describe the defense-in-depth protections that enhance the resiliency of the facility in complying with the performance objectives specified in 10 CFR 61.41 and 10 CFR 61.43.

1.1.5 Technical Analyses

The technical analyses needed to demonstrate that the performance objectives of 10 CFR Part 61, Subpart C, are met appear in 10 CFR 61.13(a) through (e). Licensees that meet the criteria in 10 CFR 61.1(b) may instead comply with 10 CFR 61.13(f). A licensee uses technical analyses to assess the impact of site-specific factors on the performance of the disposal facility and the site environment both (1) during the operational period, as in the operational safety assessment for protection of individuals during operations, and (2) for disposal of radioactive waste over the longer term, as in the analyses for protection of the general population from releases of radioactivity, protection of inadvertent intruders, stability of the disposal site after closure, and assessment of long-term impacts from LLW disposal over the performance period.

In this document, consistent with 10 CFR 61.13, the term "technical analyses" comprises the performance assessment, inadvertent intruder assessment, operational safety assessment, site stability assessment, and performance period analyses.[5] The sections below describe each of the analyses.

1.1.5.1 Performance Assessment

A performance assessment is a type of risk analysis that addresses (1) what can happen, (2) how likely it is to happen, and (3) what are the resulting impacts (Eisenberg et al., 1999). Disposal system behavior is characterized by the facility design, the characteristics of the waste, the geologic and environmental characteristics of the site, and the *processes* and *events* that influence the aforementioned features. The performance assessment identifies the specific characteristics of the disposal site (e.g., hydrology, meteorology, geochemistry, biology, geomorphology); *degradation*, deterioration, or alteration processes of the *engineered barriers* (including the wasteform and container); and interactions between the site characteristics and engineered barriers that might affect the performance of the disposal facility. The performance assessment examines the effects of these processes and interactions on the ability of the disposal facility to limit releases of radioactivity to the environment that could cause a dose to a member of the public.

To obtain a license to receive, possess, and dispose of LLW, applicants should use the performance assessment to demonstrate, with reasonable assurance, that the 10 CFR 61.41 performance objective for protection of the general public will be met. The requirements for a performance assessment are set forth in 10 CFR 61.13(a), which states that a performance assessment shall do the following:

> (1) Identify the natural characteristics of the disposal site (e.g., geomorphology, meteorology, hydrology, hydrogeology, geochemistry, biology); the characteristics of any engineered barriers; and the interactions between the disposal site and any engineered barrier's characteristics that might affect performance of the disposal site. Identify radionuclide transport characteristics of the waste (i.e., the ease with which radionuclides can escape containment). A performance assessment must further identify and examine the effects of the eventual degradation, deterioration, or alteration of any engineered barriers (including the wasteform and container) together with the disposal site characteristics to evaluate the ability of the disposal site to limit waste releases and to provide an estimate of the annual dose to a member of the public for comparison with the appropriate subpart C performance objective.
>
> (2) Consider features, events, and processes that might affect demonstrating compliance with § 61.41. The features, events, and processes considered must represent a range of phenomena, including those that may have beneficial effects, adverse effects, or both beneficial and adverse effects on performance, and must consider the specific technical information required in § 61.12(a)

[5] The terms "assessment" and "analysis" are used throughout the regulation and this guidance document. For the purposes of complying with the regulatory requirements of 10 CFR Part 61, there is no practical difference between these terms, and they are used interchangeably.

through (i). A technical basis for either the inclusion or exclusion of specific features, events, or processes must be provided.

(3) Consider the probability of disruptive features, events, or processes when estimating the consequences for comparison with the dose limits set forth in § 61.41(a).

(4) Provide *model support* for the *models* used in the performance assessment.

(5) Evaluate contaminant transport *pathways* and processes in environmental media (e.g., air, soil, groundwater, surface water) including but not limited to advection, diffusion, plant uptake, and exhumation by burrowing animals.

(6) Account for uncertainties and variability in the projected performance of the disposal site and surrounding environment.

(7) Assume the receptor is either a resident farmer or resident gardener based on reasonably foreseeable pursuits that are consistent with the activities occurring at and around the receptor location or provide justification for an alternative receptor scenario.

(8) Identify and differentiate between the roles performed by the natural characteristics and the design features of the disposal site in limiting radiological releases to the general population.

Licensees must update the performance assessment (and the other technical analyses) before closure, in accordance with 10 CFR 61.28(a)(3), to consider the waste disposed during operations.

1.1.5.2 Inadvertent Intruder Assessment

The regulations in 10 CFR Part 61 envision a period of active institutional controls for up to 100 years following closure of the LLW disposal facility. Longer or shorter periods of institutional control may be used, but for the purposes of technical analyses only periods up to 100 years may be credited. During that time, the disposal site and its contents are protected from disturbance by potential intruders through a series of measures (e.g., site access controls). At the end of active institutional controls, 10 CFR Part 61 requires licensees to assume there will be no active caretaking of the disposal site and it is possible for inadvertent intruders to gain access to the site. An inadvertent intruder is defined in 10 CFR 61.2 as—

> a person who occupies the disposal site after closure and might engage in agricultural or residential activities and other reasonably foreseeable pursuits that could unknowingly expose the person to radiation emitted or released from the waste in the disposal units.

Licensees should demonstrate that potential inadvertent intruders, who might occupy the site at any time after institutional controls over the disposal site are removed, will be protected.

An inadvertent intruder assessment (also referred to as an “intruder assessment”) is a site-specific, prospective modeling evaluation of potential radiological consequences as a result

of reasonably foreseeable human activities that might occur should an individual occupy a LLW land disposal facility after the lapse of active institutional controls. The intruder assessment is used to evaluate how these impacts compare to the appropriate performance objective in 10 CFR 61.42.

Because there is no scientific basis for quantitatively predicting the probability of a future disruptive human activity over long timeframes, an inadvertent intruder assessment does not consider the probability of inadvertent intrusion occurring. Rather, the assessment assumes that reasonably bounding *inadvertent intruder scenarios* occur and evaluates the radiological consequences that could be experienced by inadvertent intruders after active institutional controls lapse (NCRP, 2005).

As stated in 10 CFR 61.13(b), an inadvertent intruder assessment must demonstrate that an inadvertent intruder will not be exposed to radioactivity in amounts that result in doses that exceed the limits set forth in 10 CFR 61.42(a). An inadvertent intruder assessment must:

> (1) Assume that an inadvertent intruder occupies the disposal site and engages in agricultural and residential activities and other reasonably foreseeable pursuits that are consistent with activities occurring in and around the site at the time of of the inadvertent intruder assessment.
>
> (2) Account for uncertainties and variability in the projected performance of the disposal site and surrounding environment.
>
> (3) For near-surface disposal, identify barriers to inadvertent intrusion that inhibit contact with the waste or limit radiation exposure from the waste and provide a basis for their degree of effectiveness and the time period over which barriers are effective.
>
> (4) For specialized land disposal, demonstrate that the engineered barriers and natural features ensure that an inadvertent intruder will not disrupt or contact emplaced waste during any part of the compliance period that the waste remains a radiological hazard.

Because the 10 CFR 61.42(a) requirements related to barriers to intrusion differ for near-surface and specialized land disposal facilities, the analyses of potential radiological consequences also differ. For a near-surface disposal facility, the inadvertent intruder analysis would generally involve contributions from multiple exposure pathways, including direct contact with the waste. For a specialized land disposal facility, the assessment would include exposure to any radiological releases from the disposal units that an intruder could be exposed to onsite without directly contacting the waste (e.g., consumption of contaminated groundwater pumped onsite). Section G.5 of Appendix G provides additional guidance on intrusion assessments for specialized land disposal facilities.

Licensees must update the inadvertent intruder assessment before closure, in accordance with 10 CFR 61.28(a)(3), to reflect any significant changes to the activities and pursuits occurring in and around the site.

1.1.5.3 Operational Safety Assessment

The disposal of LLW requires an analysis of the protection of individuals during operations, pursuant to 10 CFR 61.43:

> Operations at the land disposal facility must be conducted in compliance with the standards for radiation protection set out in part 20 of this chapter, except that the annual dose to a member of the public from releases of radioactivity from the land disposal facility must not exceed an annual total effective dose or total effective dose equivalent, as applicable, of 0.25 milliSievert (25 millirems). Compliance with this section must be demonstrated through the operational safety assessment.

The specific requirements for the analyses are set forth in 10 CFR 61.13(c):

> An operational safety assessment of the protection of individuals during operations. The assessment must include analyses of expected exposures due to routine operations and likely accidents during handling, storage, and disposal of waste. The assessment must demonstrate that exposures will be controlled to meet the requirements of part 20 of this chapter, thereby meeting the performance objective set forth in § 61.43. These analyses can be qualitative and credit administrative controls and procedures. Operational safety assessments involving Greater-Than-Class C waste must also include quantitative analyses of expected exposures due to unlikely accidents (including fire, handling events, and other credible accidents) and the identification of safety features to prevent and mitigate accidents.

Licensees of LLW disposal facilities handling and disposing of Class A, B, and C wastes complete operational safety assessments through mostly qualitative analyses using management controls such as operational procedures, training and qualifications, radiological protection systems, monitoring, and inspection. Existing LLW disposal facilities have been very successful using management controls to provide a high level of protection to workers and the public during operations. Licensees of disposal facilities handling and disposing of GTCC wastes will likely use similar approaches to satisfying operational safety requirements but, for some types of GTCC wastes and for specialized land disposal facilities, a complete quantitative operational safety assessment comparable to that done for nuclear facilities handling dangerous amounts of nuclear materials is needed. An operational safety assessment should be more detailed and comprehensive as the level of hazard posed by the waste increases. Section G.6.1 of Appendix G contains a detailed discussion of operational safety assessment.

1.1.5.4 Site Stability Assessment

Stability is the capability of the wasteform, disposal containers, disposal site, and disposal facility to maintain their shape and properties to an extent that the disposal action will meet the 10 CFR 61.41 and 10 CFR 61.42 performance objectives. Stability is defined in 10 CFR 61.2 as follows:

> the capability of the disposal site (e.g., wasteform, disposal containers, and disposal units) to maintain its shape and properties to an extent that will not prohibit the demonstration that the disposal site will meet the performance

> objectives in §§ 61.41 and 61.42 of this part and will, to the extent practicable, eliminate the need for active maintenance after site closure and for maintenance in any form after license termination.

Stability is important to do the following:

- Minimize changes to the disposal system that may result in increased releases of radioactivity to the environment.
- Limit erosion and similar processes (10 CFR 61.50, "Disposal site suitability requirements for land disposal").
- Ensure waste is recognizable to an inadvertent intruder who contacts or disrupts the waste.
- Protect an inadvertent intruder by maintaining an appropriate overburden over the waste.

The NRC strategy for waste disposal is to "concentrate and contain" the LLW for as long as it remains hazardous. A key feature of that regulatory strategy is that components of the LLW disposal system must maintain stability. Stability ensures that once waste is emplaced and covered, access to the waste by water, biota, or humans is minimized, reducing the potential exposure to the public.

The stability of the waste and wasteform is closely tied to 10 CFR 61.56(b) and prevents waste from structurally degrading and affecting the overall stability of the site. The structural stability of the waste ensures that the waste is recognizable and nondispersible, thereby limiting exposure to an inadvertent intruder who contacts or disrupts the waste. Requirements to ensure structural stability of the waste are more prescriptive in nature while evaluations of disposal site stability are more performance-based.

The site stability assessment evaluates the long-term stability of the disposal site and can be used by a licensee to determine compliance with 10 CFR 61.44 (see Chapter 5.0). The specific requirements for the assessment are set forth in 10 CFR 61.13(d):

> An assessment of the stability of the disposal site and the need for ongoing active maintenance after site closure. The assessment must demonstrate that long-term stability of the disposal site can be ensured and that there will not be a need for ongoing active maintenance following site closure, thereby meeting the performance objective set forth in § 61.44.

1.1.5.5 Performance Period Analyses

The primary purpose of the performance period analyses is to provide information that demonstrates that releases of long-lived waste from a disposal facility are effectively managed. The regulations define the performance period to be the timeframe established for those sites that contain significant quantities of long-lived radionuclides for considering waste and disposal site characteristics to evaluate the anticipated performance of the disposal site after the compliance period. Performance period analyses are required only if the disposal facility is accepting significant quantities of long-lived radionuclides, and the licensee has used a 10,000-

year or longer compliance period. Performance period analyses are not required if the disposal facility is not accepting significant quantities of long-lived waste (and is able to justify the use of a 1,000-year compliance period). The analyses for the performance period may be similar to the analyses performed for the compliance period, but they are not required to be the same.

The disposal system consists of the disposal units, disposal site, land disposal facility, and surrounding environment. The performance period analyses should evaluate natural and engineered characteristics of the disposal system and describe how those characteristics will reduce long-term impacts. Over the long timeframes of regulatory concern, performance of the disposal system is likely to be driven by the features of the natural system rather than by man-made engineered barriers.

The regulation in 10 CFR 61.13(e) states the following:

> Analyses of how the disposal site limits the potential long-term radiological impacts during the performance period. The performance period analyses must be consistent with available data and current scientific understanding. The analyses must identify and describe disposal site design features and natural characteristics relied on to demonstrate compliance with the applicable performance objectives set forth in §§ 61.41(b) and 61.42(b).[6] In addition, the analyses must evaluate those processes likely to occur during the performance period, including degradation, deterioration, and alteration processes that affect performance.

Figure 1-2 shows the relationship between the major components of the technical analyses and their *analysis timeframes*. For example, the *Assessment Context* and *Scenario Development* components apply to the performance assessment, inadvertent intruder assessment, and site stability assessment (see Sections 2.3 and 2.5). As shown in Figure 1-2, both the performance assessment and inadvertent intruder assessment are evaluated over the compliance period. If significant quantities of long-lived radionuclides are disposed, and a 10,000-year compliance period is used, then performance period analyses are required. However, a licensee may meet the requirements for the performance period by extending the performance assessment and intruder assessment for the compliance period to the performance period. As described in Chapter 5.0, site stability assessments are only applied for the compliance period because site stability is an evaluation interpreted in the context of the potential impacts to a member of the public evaluated for 10 CFR 61.41 and 10 CFR 61.42. In other words, some instability may be tolerable, the question is how much. Site stability should be demonstrated for the compliance period, but that demonstration may transition from justifying that adequate dimensional stability will be achieved early in the compliance period to demonstrating that expected instability later in the compliance period will not compromise compliance with 10 CFR 61.41 and 10 CFR 61.42. Defense-in-depth is applicable over all time periods and all analysis types. The information gained from all these components contributes to the demonstration that the performance objectives in 10 CFR Part 61, Subpart C, are met. Chapter 6.0 contains additional discussion on the performance period analyses, including the duration of the analyses.

[6] The performance metric in §§ 61.41(b) and 61.42(b) is a qualitative measure that releases of radioactivity from the disposal site and exposures to an inadvertent intruder "must be effectively managed" during the performance period. This metric does not require meeting the dose limits in §§ 61.41(a) and 61.42(a).

1.1.5.6 Updating Technical Analyses

In accordance with 10 CFR 61.10, an initial license application for an LLW disposal facility must include general information, specific technical information, technical analyses, institutional information, and financial information, as set forth in 10 CFR 61.11 through 10 CFR 61.16. Therefore, initial license applications for LLW disposal facilities must include the technical analyses described in Sections 1.1.5.1 through 1.1.5.5 (included in 10 CFR 61.13). In addition, applications for license renewal according to 10 CFR 61.27(b) must be filed in accordance with 10 CFR 61.10 through 10 CFR 61.16 and 10 CFR 61.20, and the regulations in 10 CFR 61.28(a)(3) require a similar update to technical analyses upon application for closure:

> (3) Any revisions to the technical analyses necessary to support site closure; such revisions must consider the waste disposed during operations and reflect significant changes to the human activities occurring in and around the site.

The regulations in 10 CFR 61.25(a) indicate that except as provided for in specific license conditions, the licensee must not make changes in the land disposal facility or procedures described in their license application unless approved by the regulatory agency with oversight of the facility. Changes to the technical analyses should not be made without prior Commission notice and approval (or the Agreement State regulator).

Finally, in 10 CFR 61.24, “Conditions of licenses,” paragraph l requires that the licensee shall not operate the land disposal facility in a manner that would be inconsistent with the technical analyses and that the licensee shall evaluate whether updates to the technical analyses are warranted in accordance with the following conditions:

> (i) if significant changes have occurred at the site, or
>
> (ii) before receiving new waste streams not analyzed in the most recent approved technical analyses.

Licensees should evaluate whether updates to their most recent technical analyses are warranted based on the actual inventory disposed. If the licensee determines that the assumptions included in their technical analyses remain valid, an update is not required though the basis for that decision should be documented and retained as a record. The licensee should also evaluate whether an update of the technical analyses is needed if significant changes occur at the site, such as unexpected environmental or climatic conditions, new types or forms of waste are disposed, or an event such as a flood occurs, leading to erosion. Licensees should also update their technical analyses before receiving new waste streams such as GTCC waste that were not included in their most recent analyses.

Licensees that meet the criteria in 10 CFR 61.1(b) need not comply with 10 CFR 61.24(l).

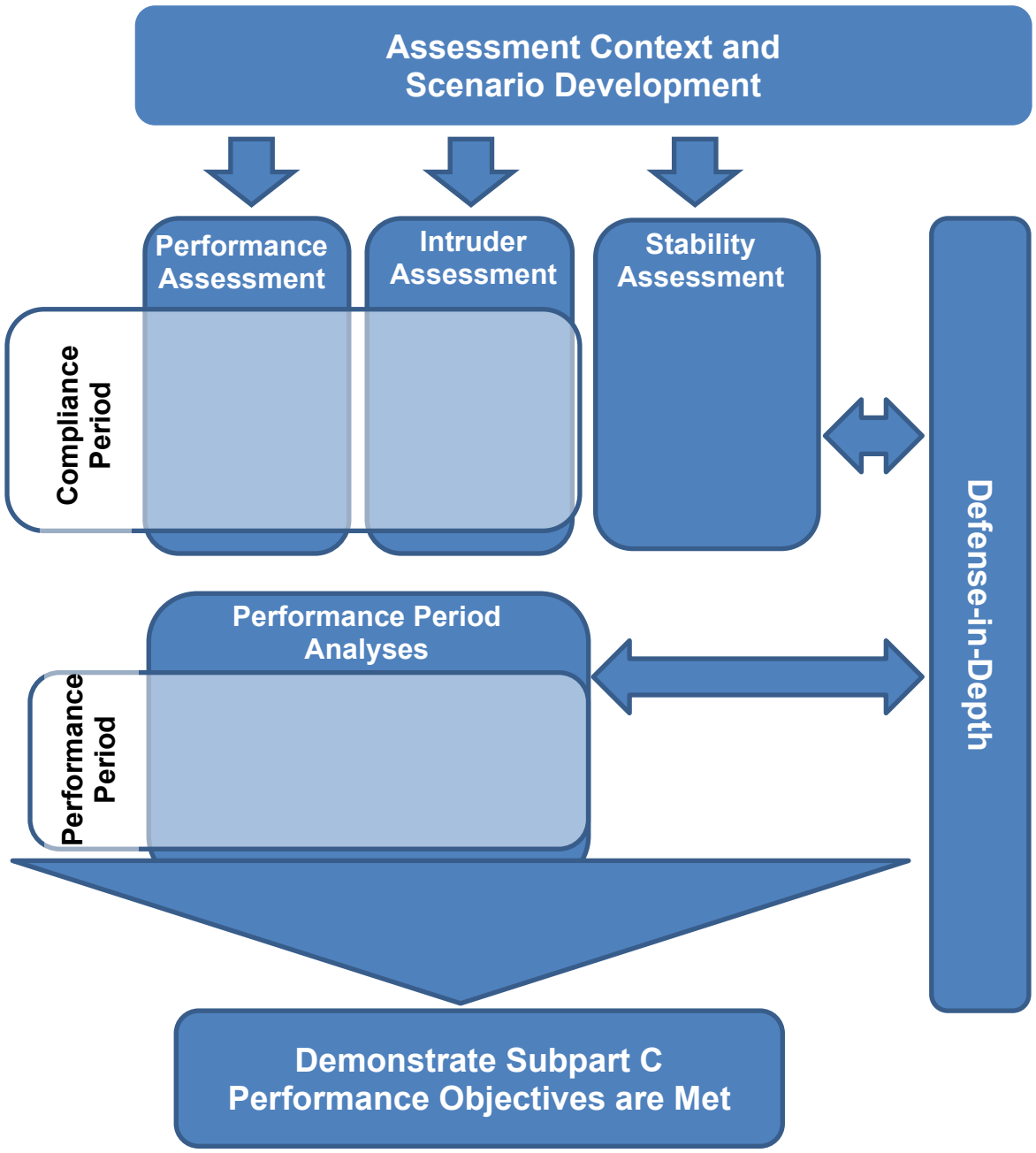


Figure 1-2 **Technical Analyses and Defense-in-Depth Relationship to Analyses Timeframes and Performance Objectives**

1.1.6 Waste Acceptance Requirements

Requirements for waste acceptance are specified in 10 CFR 61.58. The regulations require licensees to identify: (1) criteria for the acceptance of waste for disposal (i.e., waste acceptance criteria), (2) acceptable methods for characterizing the waste, and (3) a program to certify that waste meets the acceptance criteria before transfer to the disposal facility. The waste acceptance requirements are intended to provide reasonable assurance that the performance objectives of 10 CFR Part 61, Subpart C, will be met. The waste acceptance criteria, as specified in 10 CFR 61.58(a), may be generic (i.e., based on the criteria set forth in 10 CFR 61.55, "Waste classification," and 10 CFR 61.56, "Waste characteristics"), or they may be site-specific (i.e., based on the technical analyses discussed in Section 1.1.5). The waste acceptance criteria must identify allowable activities and concentrations of specific radionuclides, acceptable wasteform characteristics and container specifications, and restrictions or prohibitions on waste, materials, or containers that might affect the ability to meet the performance objectives. Licensees that meet the criteria in 10 CFR 61.1(b) need not comply with 10 CFR 61.58 and should continue with their waste acceptance programs currently approved by their regulators.

Licensees must also identify acceptable methods for characterizing the waste for acceptance. The methods shall identify the characterization parameters and acceptable uncertainty in the characterization data. The regulations in 10 CFR 61.58(b) specify the minimum information that the acceptable methods must include to adequately characterize waste for acceptance. The requirements are intended to ensure that knowledge of the waste's characteristics is commensurate with the assumptions and approaches employed in the technical analyses used to develop the site-specific waste acceptance criteria and is, thus, sufficient to demonstrate that the waste acceptance criteria are met. For generic waste acceptance criteria, adequate characterization is needed to demonstrate compliance with the criteria set forth in 10 CFR 61.56.

According to 10 CFR 61.58(c), a program is required to certify that waste meets the acceptance criteria before shipment to the disposal facility. Certification of waste provides assurance that a disposal facility operates within the limits established to demonstrate compliance with the performance objectives of 10 CFR Part 61, Subpart C.

The certification program must do the following:

- Provide procedures for designating authority to certify and receive waste for disposal at the land disposal facility.
- Provide procedures for certifying that the waste to be received at the land disposal facility meets the waste acceptance criteria.
- Specify documentation required for waste acceptance, including waste characterization, shipment information (including the information described in Appendix G, "Requirements for Transfers of Low-Level Radioactive Waste Intended for Disposal at Licensed Land Disposal Facilities and Manifests," to 10 CFR Part 20), and certification.
- Identify records, reports, tests, and inspections that are necessary to comply with the requirements in 10 CFR 61.80, "Maintenance of records, reports, and transfers."

- Provide approaches for managing waste that has been certified as meeting the waste acceptance criteria in a manner that maintain its certification status.

Lastly, 10 CFR 61.58(d) requires the approved waste acceptance criteria to be incorporated into the facility license and 10 CFR 61.58(e) requires that each licensee shall annually review the content and implementation of the waste acceptance criteria, waste characterization methods, and certification program. Chapter 8.0 discusses waste acceptance and waste classification.

1.1.7 Waste Classification

Waste classification is used to broadly describe waste with similar characteristics within the framework of LLW. Wastes with similar characteristics can be managed consistently with appropriate requirements applied to each waste class. The waste classes in the NRC requirements are distinguished by different concentrations of radionuclides rather than by the longevity (half-lives) of the radionuclides. The requirements found in 10 CFR 61.55 identify four classes of LLW: A, B, C, and GTCC.[7] Each class may be a mixture of short- and long-lived radionuclides. Additional restrictions are placed on each class of waste as the concentration of radioactivity increases from class to class. Class A waste has the least restrictions and GTCC waste has the most restrictions. To facilitate consistent classification of waste, the NRC developed Tables 1 through 3 of 10 CFR 61.55, which contain numerical limits for concentrations of select radionuclides.

If LLW contains only radionuclides not listed in Tables 1 and 2, it is Class A waste according to 10 CFR 61.55(a)(6) irrespective of the comparison of the radionuclide concentrations in the waste to the values in Table 3. However, the safety of the disposal of LLW must be demonstrated with the technical analyses required to demonstrate that the performance objectives (10 CFR 61.41 through 10 CFR 61.44) will be met, irrespective of the waste class.

Waste with radionuclide concentrations that exceed the Class C limits under § 61.55(a) is GTCC waste if the radionuclide concentrations in the waste do not exceed any of the values in Table 3. As stated in 10 CFR 61.55(b)(2)(iii) waste that exceeds one or more of the values in Table 3 is not generally acceptable for land disposal (i.e., neither near-surface no specialized disposal) unless proposals for disposal of such waste in a disposal site licensed pursuant to this part are approved by the Commission.

The main requirements for each waste class include the following:

- Unstable Class A waste must be segregated from stable wastes.

- Class B and Class C waste must meet waste stability requirements (e.g., remain stable for over 300 years).

- Class C waste must be disposed so that the top of the waste is a minimum of 5 meters (m) (16 feet (ft)) below the top surface of the cover or must be disposed with

[7] The U.S. Department of Energy (DOE) generates waste similar in technical characteristics to GTCC, which it calls GTCC-like waste. The NRC acknowledges that its guidance and regulatory requirements would be similar for GTCC-like waste compared to GTCC waste; however, the DOE is not required to comply with NRC regulations when disposing of LLW in DOE disposal facilities.

intruder barriers designed to protect against an inadvertent intrusion for at least 500 years.

- GTCC waste—
  - with concentrations of alpha-emitting transuranic radionuclides with half-lives greater than 5 years that are equal to or less than 370 kBq/g (10,000 nCi/g) may be disposed in the near surface provided that the top of the waste is a minimum of 5 m (16 ft) below the land surface **and** it must be disposed with intruder barriers that are designed to protect against an inadvertent intrusion for at least 500 years.
  - with concentrations of alpha-emitting transuranic radionuclides with half-lives greater than 5 years that exceed 370 kBq/g (10,000 nCi/g) may be disposed in a specialized land disposal facility in accordance with 10 CFR 61.52(b). This GTCC waste is not generally acceptable for near-surface disposal unless proposals for disposal of such waste (e.g., specialized design and disposal methods) in a disposal site licensed pursuant to this part are approved by the Commission.
  - must meet additional waste characteristic requirements in 10 CFR 61.56(c).
- Significant quantities of uranium must be disposed so that the top of the waste is a minimum of 5 m (16 ft) below the land surface.

The original 10 CFR Part 61 regulations allowed for the disposal as LLW of materials containing less than 3.7 kBq/g (100 nCi/g) of transuranic radionuclides. In the ILLRWD rule (NRC, 2026), the NRC revised 10 CFR Part 61 regulations to place greater emphasis on risk-informed regulation using site-specific technical analyses. As such, the limitation on the disposal of waste containing greater than 3.7 kBq/g (100 nCi/g) of alpha-emitting transuranic radionuclides with half-lives greater than 5 years has been removed. The revised requirements provide that disposal of GTCC waste with concentrations of alpha-emitting transuranic radionuclides with half-lives greater than 5 years that are equal to or less than 370 kBq/g (10,000 nanocuries per gram) may be suitable for near-surface disposal and that GTCC waste with concentrations of alpha-emitting transuranic radionuclides with half-lives greater than 5 years that exceed 370 kBq/g (10,000 nanocuries per gram) is not generally acceptable for near-surface disposal but may be disposed in a specialized land disposal facility. A licensee may seek approval to dispose of such GTCC waste by detailing to the Commission the technical characteristics of its disposal site, engineered barriers, specialized design and disposal methods (see Section 1.1.10).

The original language in 10 CFR 61.58 was moved in the final ILLRWD rule to 10 CFR 61.55 (c). This section states that the Commission may authorize other provisions for the classification and characteristics of waste on a specific basis, if, after evaluation of the specific characteristics of the waste, disposal site, and method of disposal, it finds reasonable assurance of compliance with the performance objectives in Subpart C.

Appendix G to 10 CFR Part 20 gives the requirements for transfers of LLW intended for disposal at licensed land disposal facilities and for waste manifests. Instructions for completing the NRC's Uniform Low-Level Radioactive Waste Manifest (UWM) appear in NUREG/BR-0204, Revision 3, "Instructions for Completing the Nuclear Regulatory Commission's Uniform Low-Level Radioactive Waste Manifest," issued June 2020 (NRC, 2020a). . A waste generator,

collector, or processor who transports, or offers for transportation, LLW intended for ultimate disposal at a licensed LLW land disposal facility must prepare a manifest reflecting information requested on NRC Forms 540, 541, and 542, as applicable. The licensee does not need to use these NRC Forms themselves as long as the manifest reflects the information requested on the applicable NRC Forms.

When waste is generated and prepared for shipment for disposal, the waste must be classified according to 10 CFR 61.55 even if the disposal facility operator intends to accept the waste for disposal based on its site-specific waste acceptance criteria derived from its 10 CFR 61.13 technical analyses. Requiring the UWM form to continue to reflect the waste classification according to 10 CFR 61.55 will ensure that existing transportation requirements are met, and other prohibitions associated with waste class are identified and implemented. Additional information on the site-specific waste acceptance criteria and waste characteristics should be included in pages attached to the Form 541. As required by 10 CFR 61.57, "Labeling"—

> Each package of waste must be clearly labeled to indicate the waste class in accordance with § 61.55 when shipped for disposal in a land disposal facility. Each package of waste also must be clearly labeled to identify any additional information required by the land disposal facility's criteria for waste acceptance developed in accordance with § 61.58.

Section 8.1 of this document contains guidance on waste classification.

1.1.8 Blended Waste

The NRC staff discusses the concept of "blending" in SECY-10-0043, "Blending of Low-Level Radioactive Waste," dated April 7, 2010 (NRC, 2010b). Blending is the mixing of LLW with different concentrations of radionuclides, which results in a relatively homogeneous mixture that may be appropriate for disposal in a licensed land disposal facility. The concentration of the resulting mixture is the total radioactivity in the mixture divided by its volume or weight. The types of waste may include those that are physically and chemically similar (such as ion-exchange resins from nuclear power plant systems) but they could also include different *waste types* that can be made into a relatively homogeneous final mixture, such as soil and ash. Blending does not include placement of discrete wastes of varying concentrations into a disposal container, or the averaging of concentrations of radioactivity of a discrete component over its volume (i.e., concentration averaging).

In SECY-10-0043, the NRC staff recommends addressing large quantities of blended waste through a combination of guidance and incorporation into the 10 CFR Part 61 rulemaking (later evolved into the ILLRWD rule). This approach allowed the agency's position on blending of waste streams to be risk-informed and performance-based. The staff updated the "Concentration Averaging and Encapsulation Branch Technical Position, Revision 1" (BTP CA), in February 2015 (NRC, 2015a) to address both blending and concentration averaging. The ILLRWD rule (NRC, 2026) will ensure continued safety by requiring that disposal of large-scale blended waste is subjected to a site-specific intruder assessment (see Section 1.1.5.2) as part of the technical analyses for a disposal facility to demonstrate meeting the performance objectives of 10 CFR Part 61. The principal consideration is whether a final blended wasteform could be safely disposed, therefore, licensees must perform a site-specific intruder analysis to determine whether an intruder could be protected and identify the conditions necessary for such protection.

Section 3.8 of the BTP CA provides guidance on performing site-specific or waste-specific intruder analyses as part of the technical basis for supporting an alternative approach to concentration averaging, which can be used for blended wastes.

1.1.9 Near-Surface Disposal and Burial Depth

Near-surface disposal is defined in 10 CFR Part 61 as a land disposal facility in which radioactive waste is disposed generally within the upper 30 m (100 ft) of the earth's surface. The purpose of this definition is to provide constraints for the technical requirements for this depth of disposal provided in the regulation. The requirements for disposal in a deep mine would be different than those for a trench. At the time 10 CFR Part 61 was originally developed, essentially all LLW disposal was occurring within 30 m (100 ft) of the land surface and most disposal designs used simple excavated trenches. The technical requirements that were developed, such as the waste classification concentrations in Tables 1 and Table 2 of 10 CFR 61.55, were consistent with the disposal designs at that time. However, current 10 CFR Part 61 requirements allow for the site-specific technical analysis of inadvertent intruders, so the constraints around near-surface disposal can be broadened to account for deeper excavated near-surface designs (deeper than 30 m (100 ft)) and specialized designs used for wastes such as sealed sources (e.g., near-surface boreholes).

Class A and B wastes do not have a minimum burial depth specified in the regulations. Class C wastes are required to be disposed so that the top of the waste is a minimum of 5 m (16 ft) below the top surface of the cover or must be disposed with intruder barriers that are designed to protect against an inadvertent intrusion for at least 500 years. However, GTCC waste and significant quantities of uranium (e.g., DU) may contain significant quantities of long-lived radionuclides. The hazards from these wastes may persist well after the performance of engineered covers built above the land surface can be demonstrated. To reduce the uncertainties associated with estimating the impacts of natural and anthropogenic surface processes and events on engineered surface covers, the burial depth required for GTCC waste and significant quantities of DU is to be determined from (1) the natural surface of the land immediately before development of the site (i.e., licensees should not include the thickness of the surface cover), or (2) the surface of the land after development of the site, whichever is lower. This design-based requirement will help to mitigate limitations in the long-term modeling of engineered cover performance. Figure 1-3 provides an illustration of the burial depth requirement for near-surface disposal of different wastes. Table 1-1 lists the burial depth and intruder barrier requirements for different types of waste.

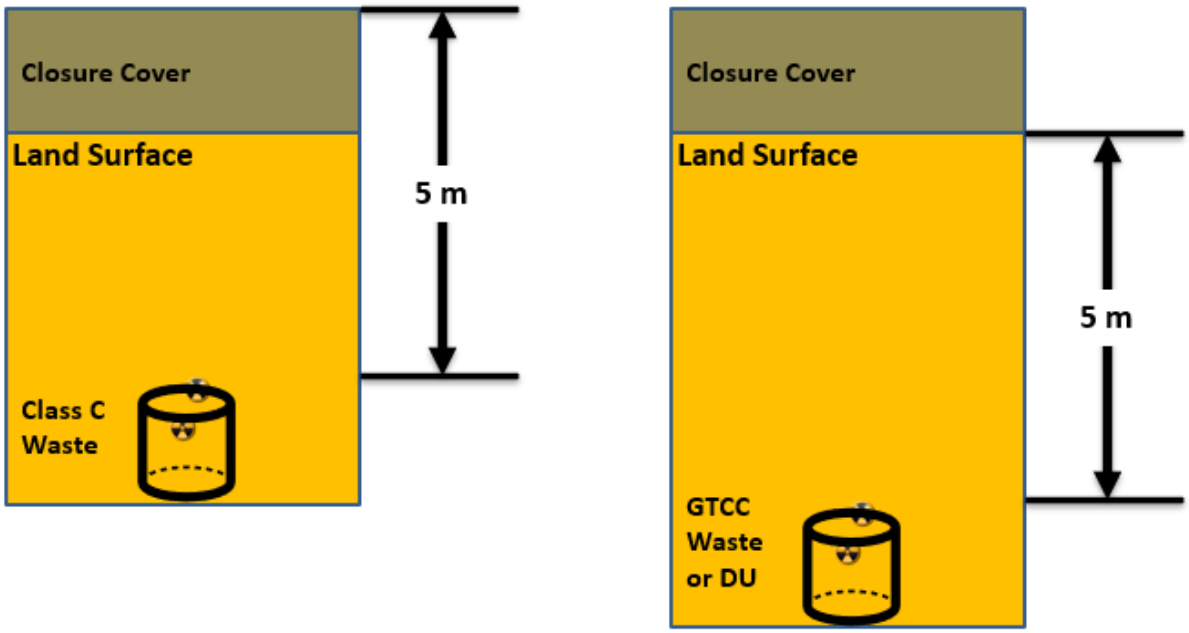


Figure 1-3 Illustration of Near-Surface Burial Depth Requirements for Different Waste Types

While GTCC waste must be disposed with intruder barriers that are designed to protect against an inadvertent intrusion for at least 500 years, intruder barriers are not required for significant quantities of DU because the uranium persists long after an intruder barrier is likely to deter an inadvertent intruder from contacting the waste. Certain types of barriers may have long-term benefits for inhibiting contact of the waste by biota (e.g., burrowing animals). If the DU is not pure and it contains short-lived radionuclides other than from uranium and its progeny, then an intruder barrier may be necessary to prevent contact with the waste while the short-lived radionuclides decay.

Table 1-1 Depth and Barrier Requirements for Different Waste Types

Waste Type	Depth Requirement	Intruder Barrier Requirement
Class A and B	No requirement	No requirement
Class C	If no intruder barrier, disposal 5 m (16 ft) below cover surface	If disposal less than 5 m below cover surface, intruder barrier is required
Significant Quantities of Uranium	Disposal 5 m below land surface	No requirement
GTCC	Disposal 5 m below land surface	Intruder barrier required

1.1.10 Specialized Land Disposal

A specialized land disposal facility is defined in 10 CFR Part 61 as a land disposal facility, other than a near-surface facility, that is designed to prevent an inadvertent intruder from disrupting or contacting waste during any part of the compliance period in which the waste remains a radiological hazard. The purpose of this definition is to establish a distinct category of disposal that provides enhanced intruder protection beyond what is possible for near-surface disposal. Because the term near-surface disposal means disposal that is generally within the upper 30 m (100 ft) of the land surface, specialized disposal typically occurs at depths greater than 30 m. However, designs can be significantly deeper, including boreholes extending hundreds of meters to kilometers below ground. Any class of LLW may be disposed in a specialized land disposal facility. However, the more rigorous requirements for a specialized facility are designed to ensure the safe disposal of GTCC waste with a concentration of alpha-emitting transuranic radionuclides with a half-life greater than 5 years exceeding 10,000 nCi/g, which cannot be disposed of in a near-surface facility without Commission approval (10 CFR 61.55(a)(1)(iii)).

As discussed in Section 1.1.9, it is difficult to demonstrate that engineered barriers in the near surface will continue to effectively deter inadvertent intrusion for more than 500 years. Therefore, because of the longer-lived hazard of GTCC waste permitted for disposal in specialized land disposal facilities as compared to near-surface facilities, the technical requirements for intrusion analyses in 10 CFR 61.13(b)(4) require licensees to demonstrate that a combination of engineered barriers and natural features will be used to prevent an inadvertent intruder from contacting the waste. An example of using a natural feature to prevent inadvertent intrusion is for GTCC waste to be emplaced at depths significantly below any attractive natural resource, such as beneath the deepest potable water aquifer[8].

Engineered barriers for specialized disposal may include more robust intrusion resistant designs. For example, a specialized facility could rely on multiple engineered barriers such as thick reinforced concrete slabs and deflection plates made from corrosion resistant metals (e.g., corrosion-resistant stainless steel, titanium). Although such barriers may be cost prohibitive or physically impractical for traditional near-surface facilities, they may be practical for disposal units with small areal footprints, such as boreholes. Sections G.5.5 and G.7 of Appendix G provide additional guidance on intrusion analyses for specialized land disposal facilities.

1.1.11 Agreement State Roles and Coordination

Section 274(b) of the Atomic Energy Act of 1954 as amended, authorizes the NRC to enter into an agreement with a State where the NRC relinquishes, and the State assumes, regulatory authority over byproduct, source, and small quantities of special nuclear material at academic, commercial, and medical facilities. Under an Agreement with the NRC, the Agreement State can regulate an LLW disposal facility authorized to dispose of Class A, B, or C LLW, including its licensing, oversight during operation, and closure, using requirements compatible with 10 CFR Part 61.

[8] In general, LLW disposal sites should be located away from natural resources that could increase the chance of inadvertent intrusion into the waste. As discussed further in Appendix G Section G.5, the NRC gave particular consideration to water well drilling because large subsurface industrial projects, such as in-situ recovery of uranium or oil well drilling is very expensive, and the NRC expects companies would detect the waste site during extensive geophysical surveys that are typically done before drilling starts.

An Agreement State is required to maintain a program that is adequate and compatible with NRC requirements to protect public health and safety and is subject to periodic NRC reviews to ensure the State maintains that level of performance. The Agreement State radiation control program is adequate to protect public health and safety if administration of the program provides reasonable assurance of the protection of public health and safety in regulating the handling, use, and storage of Agreement material. The NRC presumes that the implementation of the NRC's materials regulatory program elements affords a level of protection that provides a reasonable assurance of adequate protection of public health and safety. States use legally binding requirements to implement certain program elements such as regulations, license conditions, and orders (NRC, 2017b; NRC, 2018).

An Agreement State is compatible with the NRC's regulatory program when the State program does not create conflicts, duplications, gaps, or other conditions that jeopardize an orderly pattern in the regulation of byproduct, source, and small quantities of special nuclear material on a nationwide basis. Based on the Agreement State Program Policy Statement (NRC, 2017b), program elements (including regulations and guidance) can be placed into six categories (A, B, C, D, NRC, and health and safety (H&S)) to form the basis for evaluating and classifying the program elements. Under categories A and B, the program elements adopted by an Agreement State should be essentially identical to those of the NRC to provide uniformity in the regulation of Agreement material on a nationwide basis. For category C, an Agreement State shall embody the essential objectives of these NRC program elements; however, the Agreement State program elements may be more restrictive than the NRC program elements provided that the essential objective is met, and the State requirements do not jeopardize an orderly pattern of regulation of Agreement material on a nationwide basis. For category H&S, an Agreement State must adopt program elements in this category that embody the basic health and safety aspects of the NRC's program elements because of particular health and safety considerations, but these elements are not required for compatibility. Agreement States are not required to adopt category D program elements and category NRC includes those program elements that address areas of regulation that cannot be discontinued when a State enters into an Agreement (NRC, 2018).

> As explained in the final ILLRWD rule, Section 3(b)(2) of the Low-Level Radioactive Waste Policy Amendments Acts of 1985 provides that all radioactive waste designated a federal responsibility pursuant to (b)(1)(D) (a section which pertains to GTCC waste) that results from activities licensed by the NRC under the Atomic Energy Act of 1954, as amended, shall be disposed of in a facility licensed by the NRC. As a result, disposal of GTCC waste remains within the NRC's exclusive jurisdiction. For this reason, all provisions in the final ILLRWD rule that pertain to GTCC disposal are category "NRC."

Currently (as of 2026), all operating commercial LLW disposal facilities are in Agreement States. Based on the categories for individual program elements for the final ILLRWD rule (NRC, 2026), the Agreement States have the option to be more restrictive than NRC regulations (e.g., in terms of establishing a compliance period). A licensee must implement the requirements in the State where the facility is located.

Agreement State regulators may request technical assistance from the NRC staff to review the technical analyses provided by the licensees and to assist in interpreting the guidance in this document. The NRC staff handles technical assistance provided to the Agreement States in one of three ways (NRC, 2020b). Routine technical assistance is the most common way and is

provided as part of the NRC's daily interaction with Agreement States. This assistance may include, but is not limited to, the discussion of technical issues regarding licensing, compliance, and security. Examples of routine technical assistance include requests for and the sharing of information on licensing, inspection, security, and enforcement activities. The NRC staff may perform confirmatory reviews of portions of completed Agreement State technical assessments when resources are available. Special technical assistance is for specific actions or assignments for which the NRC staff or consultants conduct the regulatory review and make the regulatory decision on behalf of the Agreement State. Such assistance is rare and will be based on the availability of staff resources and will be cost-reimbursable. The third type of assistance is programmatic and is reserved for situations when the Agreement State is challenged to maintain an adequate and compatible program. Programmatic issues are addressed as part of the Integrated Materials Performance Evaluation Program process. Formal requests by Agreement States for technical assistance should first be discussed with the Regional State Agreement Officer and the Agreement State program in the Office of Nuclear Material Safety and Safeguards.

1.1.12 Licensing Process—Documentation and Transparency

Communication, transparency, and clarity are essential to engage the public in licensing decisions. The use of complex, site-specific models can be a challenge to public involvement; however, steps can be taken (e.g., clear and concise documentation) to alleviate this burden. Regulatory decisions should be as transparent as possible, with a clear description of the process and basis for the decisions. The regulator should produce documentation of what was reviewed, what questions (if any) were asked of the licensee, the resolution of those questions, and the basis for acceptance or rejection of the licensee's proposal. The licensing review documentation should be publicly available. The requirement to develop a safety case should help ensure that a plain-language description of the basis for the safety decision is available to the public.

1.2 Purpose of This Guidance Document

This guidance document is intended to support the implementation of the final ILLRWD rule (NRC, 2026) requirements for technical analyses to demonstrate compliance with 10 CFR Part 61 performance objectives, waste acceptance and classification, GTCC waste disposal requirements, development of a safety case, and inclusion of defense-in-depth protections at land disposal facilities.

1.2.1 Relationship to Other NRC Guidance

The NRC staff has issued several guidance documents to assist in the implementation of the requirements of 10 CFR Part 61. Additionally, in certain cases, guidance for other NRC regulatory programs (e.g., NUREG-1757, "Consolidated Decommissioning Guidance") may be applicable to 10 CFR Part 61. Early guidance for the implementation of 10 CFR Part 61 (e.g., NUREG-1199, Revision 2, "Standard Format and Content of a License Application for a Low-Level Radioactive Waste Disposal Facility," issued January 1991 (NRC, 1991a), and NUREG-1200) was generally prescriptive. More recently, review of site-specific performance assessments has become more performance based, driven in part by the Commission's 1995 probabilistic risk assessment (PRA) policy statement (NRC, 1995a). More recent guidance (e.g., NUREG-1573 and NUREG-1854, "NRC Staff Guidance for Activities Related to U.S. Department of Energy Waste Determinations: Draft Final Report for Interim Use," issued

August 2007 (NRC, 2007a)) has been developed that might be helpful for completing or reviewing a performance assessment. The following documents, among others provided in Chapters 10.0 and 11.0, are available to licensees and reviewers for additional guidance:

(1) NUREG-1200, Revision 3, issued April 1994 (NRC, 1994), provides regulators with procedures and guidance for reviewing license applications for new disposal facilities (NRC, 1994). NUREG-1200 identifies areas of review and review procedures for evaluating technical analyses, including a performance assessment (NRC, 1994). NUREG-1199, Revision 2, issued January 1991 (NRC, 1991a), provides guidance on information applicants should submit in a license application to demonstrate compliance with 10 CFR Part 61.

(2) NUREG-1573 was issued in October 2000 (NRC, 2000a) to assist in the completion of performance assessments to comply with 10 CFR 61.41 and is referenced throughout this document. Appendices B and C to NUREG-1573 contain a bibliography of technical references applicable to LLW disposal (as of 2000). NUREG-1573 provides guidance on an acceptable approach for systematically integrating *site characterization*, facility design, and performance modeling into a single performance assessment process for purposes of demonstrating compliance with 10 CFR 61.41.

In some cases, NUREG-1573 provides information on topics (e.g., analyses timeframes, climate) that are also covered in this guidance document. For those topics, staff recommend licensees use the information provided in this guidance document. Areas where there may be inconsistent information include consideration of site characteristics, timeframe for the analyses, current land use, analysis of engineered barrier performance, climate change, and consideration of disruptive events. LLW disposal regulations have changed since NUREG-1573 was developed in 2000. In addition, different types of waste (e.g., larger quantities of long-lived waste) are being considered for disposal.

(3) NUREG-1854, issued August 2007 (NRC, 2007a), provides guidance specific to the NRC staff's review of technical analyses for DOE waste determinations. The DOE uses technical analyses that are documented in a "waste determination" to evaluate whether the waste at sites in the States of South Carolina, Idaho, Washington, and New York, is high-level waste (HLW) or *waste incidental to reprocessing* (incidental waste). A waste determination is the DOE's analysis of whether the waste will meet the applicable criteria to be classified as incidental waste. The DOE sites are operating under different requirements for waste determinations; however, they all include criteria that specify that the waste will be disposed of in compliance with, or with safety requirements comparable to, the performance objectives in 10 CFR Part 61, Subpart C. NUREG-1854 contains information that may provide guidance relevant to the technical analyses referred to in this document.

(4) Revision 1 of the BTP CA, issued February 2015 (NRC, 2015a), defines a subset of concentration averaging and encapsulation practices that the NRC staff finds acceptable in determining the concentrations of the radionuclides tabulated in 10 CFR 61.55. Its primary purpose is to provide acceptable methods that waste generators, processors, disposal facility operators, Agreement State regulators, and others can use to perform concentration averaging of specific wastes and mixtures of waste for the purpose of determining their waste class for disposal. It may prove useful to LLW disposal facility

operators in determining site-specific waste acceptance criteria (see Chapter 8.0). The BTP CA describes voluntary methods that the NRC staff considers acceptable for complying with 10 CFR 61.55(a)(8). Licensees may apply these methods or use other methods to achieve compliance with this provision in 10 CFR Part 61. Licensees may continue to use the averaging positions in the 1995 BTP CA (NRC, 1995b), if allowed by the appropriate regulatory authority (e.g., the Agreement State regulator of the disposal site).

(5) NUREG-1757, "Consolidated NMSS Decommissioning Guidance: Characterization, Survey, and Determination of Radiological Criteria, Volume 2, Revision 2," issued July 2022, provides guidance on the evaluation of engineered barriers used in site decommissioning (NRC, 2022c). If similar engineered barriers are used in land disposal of LLW, then this guidance might be useful to a licensee.

(6) NUREG/KM-009, "Historical Review and Observations of Defense-in-Depth," issued April 2016 (NRC, 2016a), develops principles (i.e., goals) to help guide the implementation of defense-in-depth to prevent and mitigate adverse events and accidents. NUREG/KM-009 states the approach used for achieving defense-in-depth is to incorporate multiple layers of defense into the design and operation of the regulated activities and to ensure that these multiple layers address both prevention and mitigation. The guidance contained in NUREG/KM-009 can be used to supplement the guidance provided in Chapter 7.0 of this document.

One of the purposes of this guidance document is to complement the aforementioned documents and provide guidance in areas not previously covered. The NRC staff has attempted to provide references to other documents that may be useful for specific topics. Chapter 10.0 of this document contains a road map directing the reader to individual sections of the documents above, as well as to other documents that might be useful to 10 CFR Part 61 licensees and reviewers.

1.2.2 What is New in This Document?

A primary purpose of this document is to update guidance on conducting technical analyses to demonstrate compliance with the performance objectives of Subpart C. New or revised topics discussed in this document include the following:

- identifying and screening of features, events, and processes (*FEPs)* to develop *site scenarios*
- analyzing the protection of the inadvertent intruder
- conducting site stability analyses that evaluate the long-term stability of the disposal site to determine compliance with 10 CFR 61.44
- developing waste acceptance criteria, waste characterization methods, and waste certification
- selecting analyses timeframes: the compliance period (1,000 years or 10,000 years, depending on the quantities of long-lived radionuclides) and the performance period (for long-lived waste beyond 10,000 years)

- identifying and evaluating defense-in-depth protections
- developing the safety case
- describing the site closure process and institutional controls
- implementing performance confirmation and conducting periodic reviews of technical analyses and waste acceptance criteria
- implementing disposal considerations related to GTCC waste, including considerations for specialized land disposal facilities
- determining amounts and concentrations of significant quantities of long-lived radionuclides

A disposal facility licensed under 10 CFR Part 61 must meet the performance objectives for waste disposed at the site. The technical analyses described in this document should be performed for the total inventory of waste at the site. As such, this document does not provide guidance specific to a waste stream (e.g., DU or GTCC). Rather, it provides guidance for the total waste inventory at each site. However, the NRC staff has attempted to provide examples in this guidance on the levels of effort appropriate for the analyses of long-lived waste versus that necessary for analyses of conventional short-lived waste (e.g., long-lived waste often requires more complex technical analyses).

1.3 Document Organization

This document provides guidance on conducting technical analyses to demonstrate compliance with the performance objectives of 10 CFR Part 61, along with other related topics, such as waste acceptance and safety case. This guidance document discusses the parameters and assumptions that can be used in conducting these technical analyses in a broad sense, rather than in a prescriptive manner, to allow flexibility to licensees. The NRC staff considers this flexibility important because the site-specific nature of LLW disposal can make specification of models or parameter values impractical. Table 1-2 presents the technical analyses and related topics, the relevant requirements in 10 CFR Part 61, and the individual sections of this document where licensees and regulators can find guidance related to the analyses.

Chapter 2.0 summarizes general considerations for conducting technical analyses, such as *model abstraction*, *model uncertainty*, and model support. Chapters 2.0 and 3.0 provide guidance for specific topics related to performance assessment modeling concerned with radiological protection of the general public, as required in 10 CFR 61.41. Chapter 4.0 provides guidance specific to the assessment required for radiological protection of the inadvertent intruder, as required in 10 CFR 61.42. Chapter 5.0 provides guidance on stability of the disposal site after closure, as required in 10 CFR 61.44. Chapter 6.0 provides guidance on the performance period analyses for 10 CFR 61.13(e) associated with the disposal of long-lived waste. Chapter 7.0 discusses the identification of defense-in-depth protections to enhance the resiliency of the facility in complying with the performance objectives specified at 10 CFR 61.41 and 10 CFR 61.43. Chapter 8.0 discusses waste classification and the process for waste acceptance, such as developing waste acceptance criteria and performing waste characterization and waste certification. Chapter 9.0 discusses the site closure process, including conducting performance confirmation to evaluate and verify the accuracy of information used to demonstrate compliance before site closure. Chapter 10.0 provides a

crosswalk by performance objective and by topic for the use of other NRC guidance documents. Chapter 11.0 provides the references cited in this document and Chapter 12.0 contains a glossary of technical terms used in this guidance. Appendix G provides guidance on conduct of an operational safety assessment, which may be more quantitative for certain types of GTCC waste, and inadvertent intrusion analyses for GTCC waste. Appendix H provides guidance on determining whether significant quantities of long-lived radionuclides are present in proposed waste streams at specific disposal sites, which can establish the length of the compliance period and the need for performance period analyses, as well as determine if a licensee meets the criteria in 10 CFR 61.1(b).

1.4 Licensees That Meet the Criteria in 10 CFR 61.1(b)

The ILLRWD affects existing licensees and future applicants to varying degrees. All future license applicants that are regulated by the NRC or by an Agreement State are required to meet the revised regulations. Any currently licensed LLW site that plans to accept significant quantities of long-lived radionuclides or plan to apply to the NRC to accept GTCC waste after the effective date of the final rule is required to meet the revised regulations.

Currently operating LLW facilities that do not plan to accept significant quantities of long-lived radionuclides or do not plan to apply to the NRC to accept GTCC waste after the effective date of the final rule (i.e., licensees that meet the criteria in 10 CFR 61.1(b)) are not required to comply with new requirements (e.g., for new proposed analyses such as an intruder assessment). These licensees may, instead, continue to comply with their existing regulatory requirements. These licensees may continue to:

- Meet the performance objective in 10 CFR 61.41(c), which provides an annual dose limit of 25 mrem to the whole body, 75 mrem to the thyroid, and 25 mrem to any other organ of any member of the public and does not require a specific compliance period

- Meet the performance objective in 10 CFR 61.42(c), which requires protection of any individual inadvertently intruding into the disposal site and occupying the site or contacting the waste at any time after active institutional controls over the disposal site are removed, but does not include a requirement to perform an intruder assessment, does not include a 500 mrem/year annual dose limit, and does not require a specific compliance period

- Meet the requirements for technical analyses in 10 CFR 61.13(f), “Technical analyses pursuant to criteria in § 61.1(b)”

- Use the timeframes in their analyses deemed acceptable by the appropriate regulator prior to final ILLRWD rule

- Implement their waste acceptance programs currently approved by their regulators, which do not need to comply with the revised 10 CFR 61.58

- Meet the performance objectives in 10 CFR 61.43 and 10 CFR 61.44

- Meet the site suitability requirements in 10 CFR 61.50(c), which are preserved from the original 10 CFRR 61.50 requirements

- Meet the remainder of 10 CFR Part 61 regulations except for those listed in 10 CFR 61.1(b)(1)

These licensees are not required to develop a safety case (10 CFR 61.10(c)) or update technical analyses as described in 10 CFR 61.24(l).

As another option, an existing licensee that meets the criteria in 10 CFR 61.1(b) may voluntarily choose to be regulated under all of the requirements in the final ILLRWD rule. These licensees would be required to meet all the revised regulations (except for those added to preserve the original 10 CFR Part 61 regulations—10 CFR 61.13(f), 10 CFR 61.41(c), 10 CFR 61.42(c), and 10 CFR 61.50(c)). Appendix A provides a comparison of applicable performance objectives for licensees that meet the criteria in 10 CFR 61.1(b) relative to the current 10 CFR Part 61 performance objectives as revised in the final ILLRWD rule.

1.5 Risk-Informed Approach

This document should be implemented in a risk-informed manner. NUREG-1614, Volume 5, "Strategic Plan: Fiscal Years 2008–2013," issued 2012 (NRC, 2012b), defines "risk-informed" as follows:

> An approach to decision making in which risk insights are considered along with other factors such as engineering judgment, safety limits, and redundant and diverse safety systems. Such an approach is used to establish requirements that better focus licensee and regulatory attention on design and operational issues commensurate with their importance to public health and safety [e.g., the risk to human health associated with exposure to ionizing radiation].

A reviewer should place relatively more emphasis on technical information associated with systems that prevent a risk from being realized or that significantly reduce the magnitude of a risk. Risk is a product of likelihood and consequence. Risk-informed review generally focuses on the probability-weighted consequences (i.e., the likelihood, the consequences, or both).

Licensees should perform sufficient evaluation and develop adequate bases to identify and emphasize the key areas of the performance assessment, intruder assessment, operational safety assessment, and site stability assessment that are expected to have the biggest impact on public health and safety. The type, quantity, and concentration of waste that a facility receives will drive the risk, which in turn will drive the complexity and level of detail of the assessments.

Table 1-2 Crosswalk between Technical Requirements and Document Chapters and Appendices

Technical Requirement	Rule Section	Chapter or Appendix															
		1	2	3	4	5	6	7	8	9	B	C	D	E	F	G	H
Performance Assessment	61.13(a), 61.41(a)	X	X	X							X	X	X				
Inadvertent Intruder Assessment	61.13(b), 61.42(a)	X	X		X						X	X	X		X	X	
Operational Safety Assessment	61.13(c), 61.43	X									X	X				X	
Site Stability Assessment	61.13(d), 61.44	X	X			X					X	X	X	X			
Performance Period Analyses	61.13(e), 61.41(b), 61.42(b)	X	X				X				X	X	X				
Defense-in-Depth	61.10(c)	X						X									
Waste Acceptance	61.58	X							X								
Site Closure	61.12(g), 61.28	X								X							
Safety Case	61.10(c), 61.23(m), 61.28(a)	X															
Waste Classification	61.55	X							X								
GTCC Waste	61.13(b)(4), 61.16(b)(3), 61.23(j), 61.52(a)(3)(4)(17)(18); 61.52 (b) 61.55(b), 61.56(c)	X							X		X	X	X		X	X	
Significant Quantities	61.52(a)(16)	X				X					X						X

Various sections of this document provide risk-informed guidance. The scope, technical bases, and level of detail in the assessments should be tailored to the waste characteristics. For example:

- The level of effort required for model support for the performance assessment to demonstrate compliance with 10 CFR 61.41 will most likely be higher for waste containing significant quantities of long-lived radionuclides compared to waste containing limited quantities of long-lived radionuclides. In addition, the model support will be more detailed if the results of the performance assessment calculations are close to the dose limits, compared to instances in which there is significant margin between the estimated results and the dose limits.

- The approach to the inadvertent intruder assessment described in Chapter 4.0 provides methods for risk-informing inadvertent intruder scenario selection by considering site environmental conditions and land use information. Use of higher disturbance and land usage scenarios is recommended to mitigate uncertainties; however, site-specific conditions may support the use of other scenarios that are reasonable and justifiable.

- The approach to operational safety assessment provides for a licensee to use management controls and other qualitative methods to provide protection of public health and safety during operations. However, for certain types of wastes, such as GTCC wastes that may pose a considerable hazard if impacted by unlikely events such as a fire, the licensee should use quantitative methods to evaluate operational safety.

- The analysis recommended for the site stability assessment in Chapter 5.0 will ensure that the complexity of the evaluation is commensurate with the hazard of the material that will be disposed.

- The depth of waste disposal must meet the minimum requirements specified for each waste class. However, as described in this document, a licensee or regulator may use the results of technical or other analyses to identify greater disposal depths for certain wastes that may be used to mitigate uncertainties and to provide defense-in-depth.

1

2.0 GENERAL TECHNICAL ANALYSES CONSIDERATIONS

The term "technical analyses" refers to the performance assessment, inadvertent intruder assessment, operational safety assessment, and site stability assessment, in addition to, under certain circumstances, performance period analyses,[1] needed to demonstrate compliance with the performance objectives in 10 CFR Part 61, Subpart C. This section is intended to provide general guidance for preparing these technical analyses, with emphasis on the performance assessment.

This chapter describes the information that a licensee should provide, and a reviewer should evaluate, with respect to the basic elements that typically make up a performance assessment. Many of the basic elements that apply to the performance assessment also apply to the inadvertent intruder assessment, site stability assessment, and performance period analyses. For example, quality assurance (QA), data adequacy, uncertainty, and model support are important with respect to all the technical analyses. To the extent possible, the NRC staff has noted in individual portions of the text where the material presented for preparation and review of a performance assessment also applies to the other technical analyses.

For efficiency purposes, the general information in Section 2.2 that is relevant to the other technical analyses discussed in later chapters of this document (e.g., Chapter 4.0, 5.0) is not replicated in those sections. Section 2.2 was written to address the preparation and review of performance assessments. Inadvertent intruder assessments are generally more constrained assessments using stylized inadvertent intruder scenarios that involve calculation but do not typically involve development of integrated *conceptual models* (see Chapter 4.0). The guidance in Section 2.2 could apply to the site stability assessment if a model-based approach is used but would be of limited applicability if a design-based approach is used (see Chapter 5.0). The performance period analyses discussed in Chapter 6.0 may not be required for disposal of certain short-lived wastes; therefore, the guidance in Section 2.2 has low applicability if only a screening analysis is performed, but it is applicable if a quantitative probabilistic assessment is developed.

A technical analysis, such as a performance assessment, can be a collection of other models (e.g., submodels or process models) of varying levels of complexity, or it can be an abstracted integrated model. A submodel is a representation of a specific process as part of the technical analysis, such as a model estimating the infiltration of water to the waste in a performance assessment. Technical elements that form the basic components of performance assessment modeling include *system description*, data adequacy, uncertainty identification and assessment, model support, and integration. These technical elements, though integral to performance assessment, may also apply to the other technical analyses required by 10 CFR Part 61.

[1] The term "performance period analyses" was used simply to avoid confusion with performance assessment if "performance period assessment" was used. The terms "analyses" and "assessment" are technically interchangeable in this document.

2.1 Assessment Process

Figure 2-1 shows the steps in developing a performance assessment that licensees may use. Development of the assessment context is the first step in the performance assessment methodology followed by the description of the system. After development of the system description, a licensee would complete scenario development. Scenario development includes the identification and categorization of features, events, and processes (FEPs), the screening of FEPs, and the representation of the screened-in FEPs in site scenarios. These steps will vary slightly if the licensee uses a top-down compared to a bottom-up approach to scenario development, as described in Section 2.5. Based on the site scenarios that result, a licensee can develop conceptual models that are implemented as *numerical models*. Licensees should account for uncertainty throughout the process. Future uncertainty is accounted for by developing and analyzing site scenarios, or alternative future system states, and model uncertainty is accounted for by developing and analyzing conceptual models. In summary, this approach is designed to ensure that the FEPs important to the assessment have been included.

2.1.1 Terms and Definitions

Features, events, processes, site scenarios, and other relevant terms used in this section are defined below.[2]

Feature is an object, structure, or characteristic that has a potential to affect the performance of the disposal system. Examples include rocks within an erosion layer of an engineered cover or a drainage layer of an engineered cover.

Event is a phenomenon or change that has the potential to affect the performance of the disposal system and that occurs during an interval that is short compared to the analysis timeframe. Examples of events that cause relatively rapid change are earthquakes, floods, storms, well drilling, and excavation.

Process is a phenomenon or change that has the potential to affect the performance of the disposal system and that occurs during all or a significant part of the analysis timeframe. Examples of processes that cause relatively gradual change are radionuclide transport, differential settlement, leaching, and erosion.

FEP categorization is the process of organizing individual FEPs into categories of similar properties to facilitate FEP screening. For example, FEPs related to natural, human, or waste phenomena may be grouped into separate categories.

FEP screening is the process of using regulatory, probability, and consequence criteria to eliminate FEPs from further consideration that will not significantly impact the performance of the disposal system or are otherwise excluded by regulation.

[2] Chapter 12.0 presents a glossary of terms for the whole document.

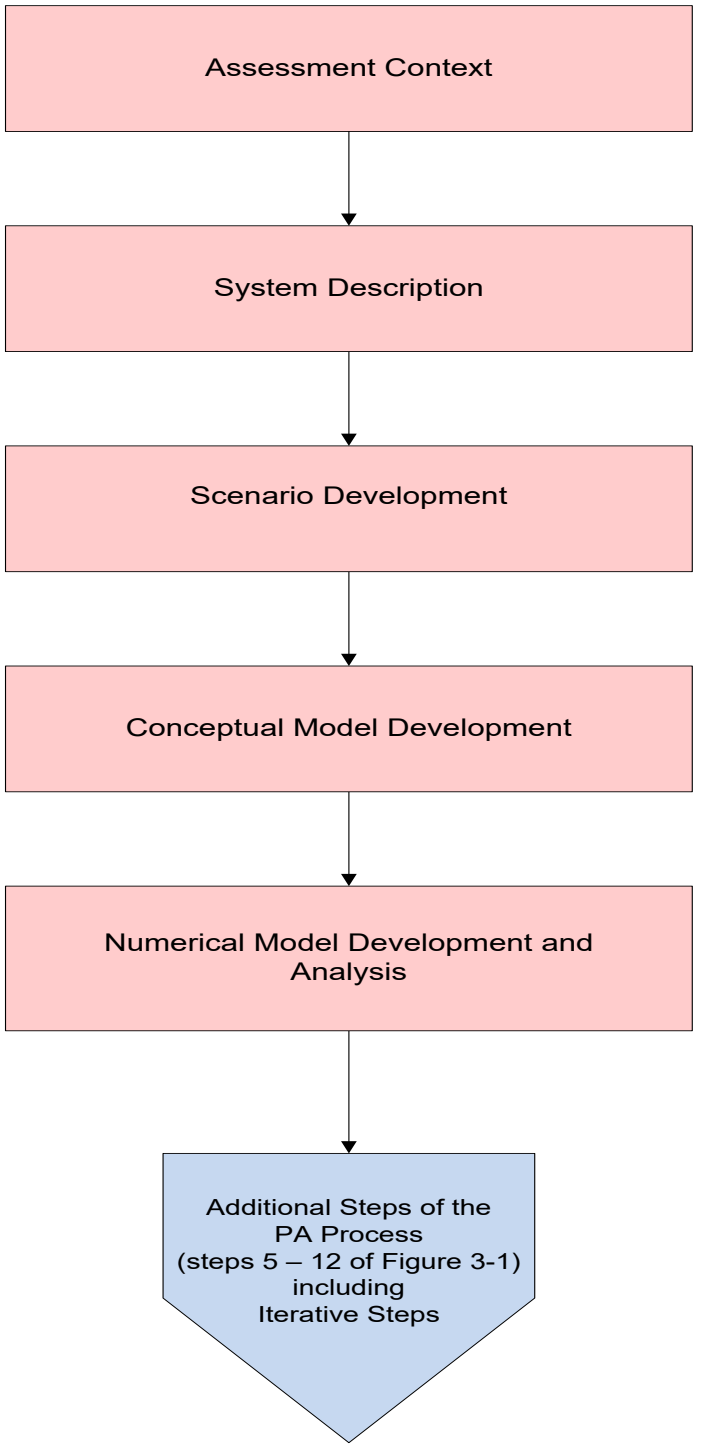


Figure 2-1 Initial Steps of the Performance Assessment Methodology

Scenario is a possible development of processes, sequence of events, or both, and their effect on features or components of the disposal system and, subsequently, on humans. This guidance document uses the following types of scenarios:

- *Site scenario* (for 10 CFR 61.41) is a type of postclosure scenario composed of FEPs that exist, or will exist, at the disposal site and their interactions with one another. The site scenario is used to identify the probable future evolution of the disposal site and its future state in which a receptor carries out activities. The central scenario is used to determine if the performance objective for 10 CFR 61.41 has been met unless dose assessment results for a receptor in a less likely, but plausible, alternative site scenario produce higher dose rates.
  - *Central scenario* (also referred to as a "central site scenario") is a site scenario that the licensee can best support as to the probable future evolution of the disposal site. As a result of the site selection process for low-level waste disposal, the central scenario generally will not include disruptive events but will include disruptive processes.
  - *Alternative site scenario* is a site scenario that is not as probable as the central scenario, but is still a plausible, future evolution of the disposal site and its future state in which a receptor could carry out activities. Alternative site scenarios may include disruptive events if those FEPs are relevant at the disposal site.
- *Intruder scenario* (for 10 CFR 61.42) is a type of stylized scenario composed of one or more exposure pathways and FEPs associated with the behaviors and activities of the humans who inadvertently intrude. The future state of the postclosure disposal site in which an inadvertent intrusion occurs is the central scenario. Intruder scenarios are used to determine if the performance objective for 10 CFR 61.42 has been met.
- *Accident scenario* (for 10 CFR 61.43) is a type of stylized scenario composed of accidents, disruptive events, and abnormal occurrences, involving the failure of the encapsulation media during disposal operations and resulting in release of radionuclides and exposure to onsite receptors. Accident scenarios are used to determine if the performance objective for 10 CFR 61.43 has been met. Accidents can include fires, disruptive events during operations, drops, construction or installation defects, earthquakes, tornadoes, etc.

Safety function is defined as a function through which a barrier component of the disposal system contributes to safety and achieves its safety objective throughout the analysis timeframe. Some types of layers, such as controls and personnel, can also act as safety functions before and during the institutional control period.

Conceptual model is a well-defined, connected sequence of phenomena describing the behavior of the system of concern.

Alternative conceptual model is an additional and different model of how the system might work that is consistent with available supporting information. For example, because of conceptual model uncertainty, a site scenario may have either a matrix flow conceptual model or an alternative fracture flow conceptual model.

Numerical model is the representation of a conceptual model representing a disposal site in the physical world with a mathematical model governed by equations in the form of computer code.

Model abstraction is the translation of a more complex process model to a simpler representation (e.g., lookup table, equation(s), parameter value(s)) that preserves the essential details and maintains validity.

Model simplification is the process of reducing the complexity of a numerical model while maintaining the validity of the simulation results.

Code is a set of software commands used to solve mathematical equations representing phenomena of the conceptual model.

Receptor activities and behaviors are the FEPs associated with the behaviors and activities of the people who may be exposed to radiation.

Long-lived radionuclides are radionuclides where (1) more than 10 percent of the initial activity of the radionuclide remains after 1,000 years; (2) the peak activity from progeny occurs after 1,000 years; or (3) more than 10 percent of the peak activity of the radionuclide (including progeny) that occurs within 1,000 years remains after 1,000 years.

Significant quantities are an amount (volume or mass) and concentration of long-lived radionuclides accepted for disposal after the effective date of the final ILLRWD rule that could, if released, result in the performance objectives of 10 CFR Part 61, Subpart C, not being met.

LLW contains radionuclides that are both short- and long-lived. From a specific activity standpoint, the short-lived isotopes make up the dominant fraction of the total activity. However, when a licensee demonstrates compliance with the 10 CFR 61.41(a) performance objective, the radiological risk is usually a result of the long-lived isotopes that remain in the inventory after the short-lived activity has decayed. LLW may contain significant quantities of DU or other long-lived waste. Some LLW may not decrease in risk after a few hundred years, although it will decrease in hazard. DU that contains significant quantities of long-lived radionuclides may pose a long-term risk to the public because of the ingrowth of progeny. LLW that contains radionuclide concentrations exceeding the limits for Class C waste are referred to as “Greater-Than-Class C” (GTCC) waste. Some GTCC waste streams also contain radionuclides that are categorized as “special nuclear material” (i.e., enriched uranium or plutonium). The regulation in 10 CFR Part 61 allows the near-surface disposal of GTCC waste with 370 kBq/g (10,000 nCi/g) of alpha-emitting transuranic radionuclides with half-lives greater than 5 years and specialized disposal of GTCC waste with greater than this concentration; Appendix G to this document discusses GTCC disposal.

When developing technical analyses, a licensee should use a level of effort commensurate with the risk to the public from the disposed waste. Specifically, the level of effort (i.e., the level of detail, comprehensiveness, completeness, and degree of iteration in the analysis development and refinement process) is commensurate to the longevity, concentrations of radionuclides, and quantity of the waste. The level of effort for the development of the technical analyses for the disposal of Class B LLW will generally be less than that for the disposal of GTCC waste, which in turn requires less effort than the disposal of HLW or spent nuclear fuel. As an example, the level of detail of an FEP analysis associated with disposal of LLW containing predominantly short-lived radionuclides will be considerably lower than that for a proposed LLW disposal site that will contain a significant quantity of DU. Complex modeling is generally not performed for disposal sites that contain only short-lived radionuclides and have low projected overall risk. Level of effort is not solely a function of the characteristics of the waste; it will also depend on the disposal site design and the natural characteristics of the disposal facility.

Historically, inadequate levels of effort while performing FEP analyses have resulted in inadequate conceptual models and the absence of *alternative conceptual models* creating situations in which the containment of the waste was compromised, and remediation and closure of the disposal facility was necessary (NRC, 2007b). FEP analysis and site scenario and conceptual model development are necessary for sites where primarily short-lived radionuclides are disposed. However, the identification and categorization of FEPs completed for the analysis of disposal of short-lived waste is expected to require less effort since the time period of the analysis is shorter. For analyses of longer time periods, there will be greater uncertainty, and a more comprehensive identification and screening of FEPs may be needed. The retained FEPs will need to be integrated into one or more conceptual models (i.e., greater level of effort). Section 2.5 discusses the importance of identifying FEPs relevant to the waste characteristics and disposal site, as well as methods to improve the completeness of FEPs lists and developed site scenarios.

2.2 General Review Considerations

This section provides general review considerations, as well as the main elements of QA associated with technical analyses. QA for technical analyses is a subset of QA applied to LLW disposal. Section 2.7.1 contains additional discussion of QA associated with numerical models.

2.2.1 Quality Assurance for Technical Analyses

QA, in the context of LLW disposal, comprises the planned and systematic actions necessary to provide adequate confidence that the applicable regulatory criteria will be met. An adequate QA program is essential to ensuring that the information relied on to make LLW disposal licensing decisions is correct and accurate. A licensee is required in 10 CFR 61.12(j) to provide a description of the QA program for the following:

(1) identification and selection of the disposal site, including any natural features relied upon to enhance the performance of the disposal site

(2) development of technical analyses

(3) design and construction of the land disposal facility

(4) operation of the land disposal facility, including the receipt, handling, and emplacement of waste

(5) site closure of the land disposal facility

This section provides updated guidance on QA for the development of the technical analyses required in 10 CFR 61.13. NUREG-1200, Revision 3 (NRC, 1994), provides guidance on items (1) and (3). Additional resources are available in Appendix I to NUREG-1757, Volume 2, Revision 2 (NRC, 2022c), in which regulatory guidance pertinent to QA is applied to decommissioning sites; NUREG-1293, Revision 1, "Quality Assurance Guidance for a Low-Level Radioactive Waste Disposal Facility," issued April 1991 (NRC, 1991c); and Sections 4.4 and 8.0 of NUREG-1854 (NRC, 2007a).

A QA program includes the following main elements:

- management systems
- assignments of responsibility within the organizational structure
- training and qualifications
- procedures
- document control
- audits and surveillance
- corrective actions

QA applied to technical analyses will be a subset of the overall QA program for the LLW disposal facility. The elements of QA for technical analyses may include the following:

- data quality
- development of scientific and engineering software
- software evaluation
- technical review
- transparency and traceability
- technical documentation

The review of QA for technical analyses should verify that the licensee has applied QA measures to the initial development of and updates to the technical analyses. A graded approach based on the hazard and risks associated with the waste may be used to review the QA information provided by a licensee. The adequacy of QA measures may be determined using a sample of analyses selected based on their risk significance, and conclusions may be based on the impacts of any identified deficiencies on performance. For example, a QA deficiency of a risk-significant element of the performance assessment may have a minor, moderate, or major impact on the results of the analysis. The reviewer should consider both the severity and the pervasiveness of deficiencies when evaluating the adequacy of a licensee's QA measures.

2.2.1.1 Data Quality

Technical analyses, such as performance assessments, may rely on a vast amount of information (e.g., hundreds of inputs). The information, termed "data inputs," can be obtained from a variety of sources, and the data may be unqualified or qualified. The rigor of the

qualification can differ for different sources of information. Ideally, the data inputs that licensees use in their technical analyses were obtained or qualified under an "acceptable" QA program, such as, but not limited to, an NRC-approved QA program developed to meet the requirements of 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix B, "Quality Assurance Criteria for Nuclear Power Plants." In some cases, not all the data used in a performance assessment have been qualified. At a minimum, data must be documented and identified in a manner that facilitates determination of the qualification status (qualified, unqualified, accepted) of the data. Regardless of the qualification status of the data, the validity of the data for use in the technical analysis should be established and documented. If data that are not qualified are used, the licensee should establish the validity of the data for use in the technical analyses.

The validity of qualified data is established through a qualification process, including the following:

- If data are not collected under an acceptable QA program, the data should be verified by one or more of the following processes: a QA program that is equivalent to an NRC-approved QA program (e.g., in accordance with 10 CFR Part 50, Appendix B), comparison to corroborating data, confirmatory testing, or peer review (NRC, 1988a).

- Data obtained from literature sources such as journal articles, NUREGs, and other similar technical reports should reference their original source.

- Modifications to the data should be described.

- The units of the data should be provided.

- The representativeness of the data for the site-specific application should be provided; temporal and spatial variability should be considered.

- The appropriateness of the data for the scale of application should be stated.

- The technical basis (e.g., statistical basis) should be provided to support the identification of data values as outliers.

Many data inputs will not have a significant impact on the results of the technical analyses. It is common that less than 10 parameters out of many hundred data inputs may explain the total variance in the results. Confidence should be high in the validity and representativeness of the most important parameters.

2.2.1.2 Development of Scientific and Engineering Software

Licensees should ensure that the software used conforms to the active standards of the Institute of Electrical and Electronics Engineers (IEEE) for software development, QA, testing, *verification*, and *validation* (e.g., IEEE Standard 730-2014, IEEE Standard for Software Quality Assurance Processes, issued June 2014, (IEEE, 2014)). Reviewers should be familiar with the NRC's software QA/quality control (QC) guidance (i.e., NUREG/BR-0167, "Software Quality Assurance Program and Guidelines," issued February 1993 (NRC, 1993c); and NUREG-0856, "Final Technical Position on Documentation of Computer Codes for High-Level Waste

Management," issued June 1983 (NRC, 1983f)) to ensure that the licensee's QA/QC procedures for selection and development of numerical models are comparable.

Software may need to be developed for the technical analyses or the software may be acquired from other sources.[3] The appropriate QA process for developing software may differ significantly from the QA process used when acquiring software for use. Acquired software may include commercially available products, as well as software obtained from external sources. It is the licensee's responsibility to determine that software is accepted for use. "Accepted" software has been demonstrated to perform intended functions.

2.2.1.2.1 Software Acceptance

Acquired software will need to be evaluated to determine that the software can be accepted and used. Requirements controlling software procurement should be established to ensure proper verification and validation support, software maintenance, configuration control, and performance of software audits, assessments, or surveys. Software acceptance will differ based on the maturity of the software and if the software is widely used. Mature and widely used software may be accepted as is after completion of installation testing. In some cases, the NRC may have endorsed the software. Further acceptance testing beyond installation testing is not necessary for NRC-endorsed software. The main elements of software acceptance may include the following:

- An installation test should be performed on each intended platform and operating system before its acceptance and release for use. Documentation of the installation testing should indicate who performed the testing and what tests were performed.

- A copy of the source *code* should be included with the QA documentation unless prohibited by copyright or licensing restrictions.

- A software validation plan should be developed that describes and justifies the approach taken for validation.

- Documentation should describe the limitations of the software.

- Requirements are provided for suppliers reporting software errors to the purchasers and, as appropriate, the purchasers reporting software errors to the supplier.

2.2.1.2.2 Software Development

Software development should be planned, controlled, and documented to establish that the software is validated for use. Planning for validation identifies the validation methods and validation criteria to be used. The licensee should demonstrate that the software adequately represents the intended processes and systems.

A licensee should apply controls to software to ensure that the software supporting the technical analyses is qualified for use and has been developed, tested, and controlled under suitable conditions. Specifically, the licensee should demonstrate that the following conditions are met:

[3] Models that are developed using commercially available modeling software should be managed similarly to developed software.

Planning and Development:

- The software development and maintenance process is completed in a planned, traceable, and orderly manner, using a defined software life-cycle methodology.
- A software configuration management system has been established.
- Software requirements documents, software test plans, and software validation plans are used.
- Training on software usage and QA requirements is provided.

Verification, Validation, and Testing:

- Software verification activities were planned, documented, and performed for each item of software.
- Software that was verified and was subsequently changed has undergone additional verification and validation, and documentation of the additional verification and validation has been developed.
- The software performs intended functions and provides correct solutions.
- A software test plan will demonstrate that the software performs intended functions. Tests should provide evidence of correct and successful implementation of underlying theory and algorithms and comparison to benchmarking or results from comparative testing with other software, if available. The test plan should identify acceptance criteria that specify what constitutes a passed or failed test.
- Software testing is completed by comparison to measurement results, research papers, laboratory experiments, existing test cases, or other published sources of data. In the absence of data, independent peer review may be used. Comparison to the results of other modules may provide an additional line of support.
- A test case (or cases) representative of the intended application should be performed.
- A test case typically describes the requirement being tested, the software or module being tested, the test conditions, the inputs, the outputs, and expected results.
- Test cases should be used to evaluate the stability of the numerical methods and the accuracy of the numerical methods to represent the *mathematical model*.
- When one unit of software or a module is used in combination with other units of software or modules, the testing should verify the correct transfer of information, that the measurement units in the components are consistent or have been converted properly, and that the scale of data (temporal and spatial) is consistent or has been converted properly.

Documentation:

- Software requirements documentation defines the objectives, inputs and constraints, operating systems and platforms, user interfaces, and overall structure and flow of information.
- Documentation should be developed that is sufficiently detailed to allow an informed reviewer to follow the logic of software development.
- A User's Guide should be developed that provides installation instructions, discussion of limitations, resource requirements, and installation test cases.
- Test cases and results should be described. The range of inputs included in the testing should be documented.
- A description of mathematical models used in the software should be provided, including governing equations, formulas, algorithms, and their scientific and mathematical bases.
- A copy of the source code should be included with the QA documentation unless prohibited by copyright or licensing restrictions.
- Errors identified should be documented, and their impacts should be assessed and documented.

2.2.1.3 Technical Review

Technical review is an essential element of the technical analysis process. The guidance that follows applies to internal technical review completed by the licensee before release of technical analyses to stakeholders and to regulators reviewing the licensee's technical analyses.

Technical analyses may use software and modules or other calculations (e.g., spreadsheets). The reviewer should assess the analysis to confirm that it was transparently documented, the objective of the analysis is described, accepted software was used, and the software or modules were not used outside of the range of their intended functions. The internal technical reviewers must be technically qualified to perform the review and be independent from the work being reviewed. Specifically, the reviewer should ensure the following:

- The objective (intended use) of the software has been provided.
- The description of the conceptual model implemented is clear.
- Software has been used in a manner that is consistent with the conceptual model implemented in the software.
- Inputs and their sources have been identified.
- The technical basis supporting the assumptions is provided.
- Software or models have not been used in a manner that exceeds known limitations.

- Associated software used, computer calculations performed, and inputs and outputs have been identified and are traceable.

- Initial and boundary conditions are discussed.

- Model limitations (e.g., data available for model development, valid ranges of model application, spatial and temporal scaling) are discussed.

- Results produced by the software are appropriate given the various sources of uncertainties (i.e., conceptual model, mathematical model, process model, system model, parameters).

- If a defect was identified in software that adversely affects the results of a previous application of the software, the condition adverse to quality was documented and controlled.

2.2.1.4 Transparency and Traceability

Transparency and traceability apply to data, software, and technical analyses, including the associated documentation for each. Traceability is the ability to trace the history, application, or location of an item (and like items) or activities through calculations and documentation. Acceptable traceability means items or data can be traced from the original source, or the originator of the data, to the final usage, through all calculations and data reductions. Transparency means the ability to easily access and work with data no matter where they are located or what application created them.

The licensee should ensure, and the reviewer should confirm the following:

- Data are identified in a manner that facilitates traceability from the associated documentation back to the source and the qualification status of the data are clearly identified (e.g., qualified, unqualified, or accepted, such as, for a physical constant).

- Documentation of data traceability and qualification is transparent and identifies the principal lines of investigation considered. A document is transparent if it is sufficiently detailed as to purpose, method, assumptions, inputs, conclusions, references, and units so that a person technically qualified in the subject can understand the document and ensure its adequacy without recourse to the originator.

- The data reduction process is described in detail sufficient to allow independent reproducibility by another qualified individual. Data reduction includes processes that change the form of expressions, quantity of data or values, or number of data items. Documentation of data reduction inputs, outputs, and computational methods should be verified.

- Changes that affect data identification (e.g., category or use/application) are made in a manner that preserves traceability.

2.2.1.5 Technical Documentation

The primary objective of a licensee's technical documentation is to provide a description in sufficient detail such that an independent reviewer can understand the document without recourse to the originator. The technical documentation provided by a licensee should do the following:

- Provide transparency and traceability of data, software, and technical analyses as discussed in Section 2.2.1.4.
- Describe the work in sufficient detail that a qualified independent reviewer can understand the work without recourse to the author.
- List assumptions and the basis for assumptions.
- List references cited.

2.2.2 Data Adequacy

It is important for a licensee to develop and use valid data. The data should be appropriately representative and complete, of adequate quality, and unbiased. There may be some overlap between data quality (see Section 2.2.1.1) and data adequacy. The objective of the data adequacy review is twofold. First, the reviewer should determine whether a licensee has provided sufficient data to support the performance assessment models. Second, the reviewer should determine whether the data have been used appropriately.

It is generally beneficial to have more data rather than less; however, some incompleteness in the data may be overcome by appropriately accounting for *parameter uncertainty*, as described in Section 2.2.3.1.3. The types of data to be considered by licensees may include, but are not limited to the following:

- site-specific data (e.g., laboratory and field-scale measurements or experiments)
- data from analogous sites
- data from generic sources
- output from detailed process-level models
- expert judgment

More objective sources of data are preferred over more subjective sources of data. The list above is ordered in increasing subjectivity. Licensees should review published literature even if they have made site-specific measurements. Review of published literature may help the licensee identify measurement errors or determine if the data are not representative. Licensees should use expert judgment only if necessary. When expert judgment is used to assign data values, appropriate uncertainty should be included, and the expert judgment should be completed with a formal expert elicitation process when the data are important to the performance assessment (NRC, 1996; NRC, 2022b). Because performance assessments are completed with an iterative process, initial data for scoping calculations may be from more subjective sources (e.g., informal expert judgment). If a licensee or regulator determines during independent analysis that certain data are risk significant, then the data should be replaced with data from more objective sources if possible, or appropriate uncertainties should be included.

Because performance assessment models can simulate processes and events over a wide range of temporal and spatial scales, licensees should ensure that the data are representative of the temporal and spatial scales evaluated in the model. *Upscaling* of data may be necessary to achieve representativeness. Upscaling is the modification of data for use at a different scale, most commonly to take data from fine-scale observations for use at a much coarser scale. If licensees use upscaling to modify data for use at different scales, they should ensure that the key features or structure of the data important to the performance assessment are preserved. For example, precipitation may occur as brief storm events (minutes to hours), whereas the timeframe for a performance assessment may be thousands of years. If a performance impact, such as erosion of a soil cover, is a non-linear function of runoff that itself is a function of precipitation intensity and duration, upscaling of the temporal precipitation data to develop the performance assessment calculations may result in the loss of important detail. Licensees should use caution when estimating whether measurements are outliers, particularly when there are few measurements. Licensees should not use subjective assessments of potential outliers. Additional data should be collected to provide a statistical basis that a measurement is an outlier.

2.2.3 Uncertainty

A performance assessment is not used to predict a future radiological dose. Performance assessment models and calculations are used to provide ranges of potential future outcomes from a disposal action considering many sources of uncertainties. In many cases, conservative biases may be introduced to account for uncertainties.

Sources of uncertainty inherent to waste disposal in the near surface include, but are not limited to, incomplete knowledge of the natural system, its evolution, and interactions. For the inadvertent intruder assessment, sources of uncertainty include performance of the intruder barrier in the face of a wide variety of drilling technologies and methods. Regulators expect that uncertainties that cannot be shown by a licensee to have a minimal effect on safety are avoided or reduced as much as possible (e.g., by means of site selection, site characterization, disposal design, and, if necessary, research). To some extent, uncertainties in the assessment results can be counterbalanced by using multiple lines of evidence, or model support. Licensees should characterize, eliminate (with justification), or bound uncertainties in their technical analyses, as well as document the impact of the uncertainties on performance.

The uncertainties in performance assessment can be classified as *site scenario uncertainty* (also referred to as “scenario uncertainty”), *model uncertainty* (which spans conceptual model uncertainty and mathematical model uncertainty), and *parameter uncertainty* (i.e., uncertainty in values used in the numerical model) (NRC, 1990a). Scenario uncertainty, defined as the consideration of uncertainty in the future evolution of the site, may result in several different conceptual models for the system, distinguished by the effects of phenomena on the system. Model uncertainty may be present in each of the different submodels comprising the overall system model. Licensees can evaluate these inherent uncertainties using *uncertainty analysis*, which is a way of formally assessing, reducing, or managing and documenting the inherent uncertainty of a system (Finkel, 1990). For example, an uncertainty analysis could provide information about where a licensee should focus model support activities, which in turn could reduce uncertainty.

2.2.3.1 General Structure of Uncertainty

Some radioactive waste will present a radiological risk to humans and the environment for a long time. The radiological risk from much of the activity in traditional commercial LLW is reduced substantially by radiological decay in hundreds of years. However, depending on the quantity of long-lived radionuclides, the radiological risk can persist for thousands of years, especially for DU and GTCC waste. To help assess this risk, licensees should use models that can describe the potential future behavior or performance of disposal systems. Building a *computational model* that combines and represents important FEPs at an appropriate level of detail is a complex process. Uncertainties are greater for long-lived waste (lasting thousands of years or longer) than they are for short-lived wastes (lasting tens or hundreds of years). Licensees must handle uncertainties within the performance assessment process. Scenario development is a commonly used technique to account for uncertainties about the future.

Figure 2-2 (from NRC, 1993a) shows a general structure of uncertainty analysis that involves separate treatments of scenario uncertainty (i.e., site scenario uncertainty), model uncertainty (i.e., conceptual model uncertainty), and parameter uncertainty (i.e., parameter value uncertainty). Within each site scenario (i.e., how the site will evolve in the long-term future), it is possible to postulate different or alternative conceptual models of the behavior of the disposal system, each of which leads to a particular mathematical model to describe that behavior. For each conceptual model, it is possible to postulate alternative sets of input parameter values. In the context of performance assessment for LLW disposal, the primary purpose of an uncertainty analysis is to support a decision about compliance of the disposal system with the regulatory requirements.

2.2.3.1.1 Scenario Uncertainty

Uncertainty about the future of the site is the result of inherent lack of knowledge about how the site will evolve over time although this uncertainty will vary from site to site. The future climatic, geologic, and population conditions that will prevail at a site are not known, but it is generally required in the performance assessment process that a licensee consider possible future conditions. Scenario uncertainties are evaluated in the technical analyses by including in the assessment the events or processes that may significantly influence projected doses to the receptor. For example, climatic variation may significantly change groundwater flow pathways over time, necessitating changes to the groundwater flow model or the introduction of new parameters. If analyses cannot exclude the possibility of either a site scenario in which the groundwater flow pathways remain unchanged (no major climate variation) or a scenario in which the groundwater flow pathways change (climate variation), then the site has two potential routes of evolutionary development (i.e., two site scenarios that need to be analyzed in this example). The longer the analysis timeframe, the greater the likelihood of significant changes to the flow path and properties. Section 2.5.4 provides additional information on site scenarios.

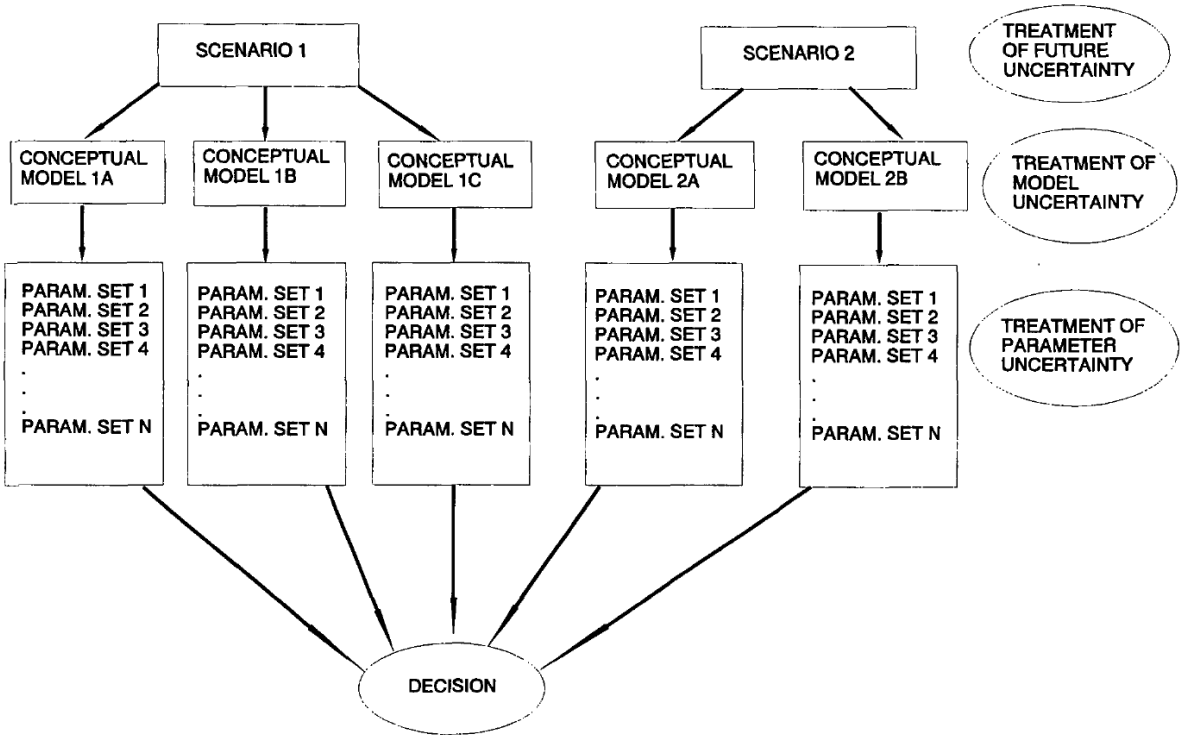


Figure 2-2 Overall Approach to Uncertainty Analysis for LLW Performance Assessment (NRC, 1993a)

2.2.3.1.2 Model Uncertainty

Model uncertainty encompasses the uncertainty in the conceptualization of the system, the uncertainty in its mathematical representation, and the uncertainty in the solution of the mathematical representation (Bonano and Cranwell, 1988). All models are a simplification of reality (model abstraction) and licensees will have to make a variety of choices about the level of detail to include in the model.

Conceptual model uncertainty is generally the dominant type of uncertainty in a performance assessment because of limitations in the available supporting data. The conceptual model should be as complete and as appropriate to the site scenario as possible. The conceptual model should be based on the available information and data and on previous experience with similar types of problems. If not all significant processes, features, or significant barriers have been considered, the conceptual model will have deficiencies and the ability of the licensee to bound or otherwise account for model uncertainty will be reduced (BIOMOVS II, 1996).

Licensees should adequately describe and document conceptual model uncertainties. The performance assessment documentation should provide the assumptions, limitations, and uncertainties of the models. Multiple representations of a model or different models may be consistent with the available data. In general, licensees should select the models that best represent available data. However, when data are sparse, multiple models may represent the available data. In this case, licensees should select the model that provides the most conservative result or collect additional data to determine which alternative conceptual model provides the best representation. Licensees do not need to evaluate all models, but they should

consider models that are reasonably consistent with available information. Regulators should perform an independent evaluation of model uncertainty and consider whether more than one conceptual model should be evaluated, especially for complex sites. Section 2.6 provides additional information on conceptual models.

Other types of model uncertainty include assumptions, decisions or judgments made during the development of a model, the mathematical form of the conceptual models, and the inexact implementation of mathematical models in numerical form in computer codes. Related computer model uncertainties can arise from errors in the computer code used to develop the model, input data errors, misapplication of the code (e.g., through application of the code to problems beyond the range for which the code was developed), and approximations in the solution of the mathematical model (e.g., due to inappropriate grid discretization of a domain or setting time steps that are too large).

The objective of the review of model uncertainty is to determine whether the licensee has considered and appropriately evaluated the impact of model uncertainty. Some uncertainties are inherent in the application of predictive models to (1) long periods of time for which direct validation is not possible, and (2) complex systems for which measurement and characterization may be limited. Ideally, a licensee can minimize the impact of model uncertainty by developing as much model support as practical (see Section 2.2.4). The licensee can better understand the impact of model uncertainty by (1) considering reasonable ranges in conditions and processes to test the robustness of the model, (2) using distributions of parameters to represent the likely ranges in conditions or processes, or (3) bounding the effects of model uncertainty by using conservative assumptions.

2.2.3.1.3 Parameter Uncertainty

Parameter uncertainty can be reducible or irreducible. Technical analyses will need to account for parameter uncertainty. Although evaluation of parameter uncertainty does not specifically address scenario uncertainty, parameter values are essential inputs to models. Spatial or temporal variations in parameter values are often needed to reflect the evolution of a site and to evaluate the impact of different representations of what might happen in the future. For example, the range of flow rates through a hydrogeological unit may increase over time in one plausible site scenario. In an alternative, drier site scenario, the same hydrogeological unit will start out with the same flow rate, but, in contrast with the previous site scenario, may decrease over time.

The objective of the review of parameter uncertainty is to determine whether the assessment by the licensee includes parameter uncertainty, which techniques were used to account for parameter uncertainty, and whether those techniques adequately accounted for parameter uncertainty. Selection of conservative values is one method a licensee may use to account for parameter uncertainty. This method works well when the number of uncertain parameters is limited, and conservative values can be clearly established. This method may not work well when there are many uncertain parameters, especially for complex models. Licensees or regulators should not interpret this to mean that conservative values should not be used for complex sites, but rather that licensees should demonstrate sufficient knowledge of their performance assessment so that conservative selections can be reliably made. Variations in the model responses, such as local minima and maxima in the outputs, can make the determination of conservative parameter values very challenging.

Another method to incorporate parameter uncertainty is the use of some form of *probabilistic analysis*. Parameter uncertainty can be propagated through the performance assessment by distributions of variables (e.g., hydraulic conductivity, porosity, *distribution coefficient*). When using probabilistic analysis, licensees should plot measured data on figures showing the probability distributions assigned to represent the measured data in the modeling to communicate: (1) the amount of data available to construct the distribution, and (2) how the selected distribution reflects the data. Licensees should preserve the correlation of data for related parameters (e.g., precipitation and irrigation rate) in probabilistic analysis, as necessary. Performance assessments that rely on a large amount of generic data (e.g., non-site-specific) will have comparatively more uncertainty, as discussed in Section 2.2.4. In a *deterministic analysis*, licensees can examine the impact of parameter uncertainty with sensitivity analyses. The licensee can represent uncertainty in the deterministic analysis by the selection of conservative values. Section 2.7.4 discusses some significant challenges with this approach, as well as the metrics to use for determining compliance (i.e., the peak of the mean).

2.2.3.1.4 Uncertainty Example—Transfer Factors

One acceptable quantitative approach for modeling the transport of radionuclides through the biosphere employs steady-state transfer factors (e.g., soil to plant, water to biota) and bioaccumulation factors. The literature contains many sources for these transfer factors (Staven et al., 2003; IAEA, 1994; NRC, 1992; Wang et al., 1993; Baes et al., 1984; NRC, 2007c; NRC, 2003a). Regulatory Guide 1.109, Revision 1, "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," issued October 1977 (NRC, 1977a), provides conservative values for a variety of these factors. Licensees should use site-specific parameter values or ranges of values, if available, and should document their usage. Site-specific parameter values are those measured at the site under consideration; true site-specific values are relatively rare. If site-specific transfer factors are not available, a licensee should document how its site conditions are comparable to the conditions under which the transfer factors that were used were collected. Licensees can infer transfer factors for a specific site from compilations in the literature; however, inferred values carry high uncertainty, which can be difficult to properly account for in a deterministic assessment.

Transfer factors can have very large variances and be non-uniformly distributed. These large variances may require careful treatment in the intruder assessment or performance assessment. If a transfer factor has few observations and the actual observations span many orders of magnitude, the appropriate statistic of the inferred distribution to use in the technical analysis may be difficult to determine. This is because the observations represent both inter-site and intra-site variability. The technical analysis should not benefit from inter-site variability. Use of the geometric mean, for example, may result in a high likelihood that data that are not representative of the specific site have been "credited" in the analysis. Even if intra-site values are used to extend the "conservative" end of the distribution (e.g., high values of transfer coefficients), the overall effect may not be conservative because overly broad uncertainty ranges can lead to risk dilution (Section 2.7.4). If no observations at a site are available and correlations between the specific site conditions and the conditions for which the observations were made are not available, then licensees should use conservative statistics for transfer factors in deterministic analyses. In probabilistic analyses, licensees can use the full non-truncated distribution (assuming the results are still physically reasonable) and use sensitivity analyses to determine if the uncertainty in the transfer factors is important. If the uncertainty is important, then licensees must provide a technical basis to explain why the distribution that

contains inter-site variability used for the technical analysis is protective of public health and safety, or the licensee may elect to collect additional site characterization data to better define the range and variability.

2.2.4 Model Support

Model support is one of the most essential technical elements of a licensee's analyses to demonstrate compliance with the performance objectives of 10 CFR Part 61, Subpart C. Performance assessments and site stability assessments are projections many years into the future, and therefore, cannot be validated in the traditional sense. Inadvertent intruder assessments rely on plausible but hypothetical site scenarios based on past, present, and future activities. Support for the calculations is essential for effective decision-making. Model support can help reduce uncertainty and provide a mechanism to determine when sufficient iteration in the development process has been performed. NUREG-1636, "Regulatory Perspectives on Model Validation in High-Level Radioactive Waste Management Programs: A Joint NRC/SKI White Paper," issued March 1999 (NRC, 1999a), provides regulatory perspectives on model validation for HLW disposal. Many of the concepts found in NUREG-1636 are generic and are applicable to LLW disposal as well.

Model support can be divided into verification-type activities (i.e., determining that the equations were solved correctly) and validation-type activities (i.e., determining that the correct equations were solved). Methods for verification of computational models are reasonably well-established.

The objective of the review of model support is to determine if the technical analyses have adequate support to justify the estimated system performance. In addition to reviewing a licensee's QA procedures for computational modeling, the reviewer may perform independent analyses. Independent analyses may be performed with the licensee's models, with independent models, or with simplified calculations. The objectives of independent analyses are to test, confirm, or refute the licensee's assumptions and analyses.

Because the primary output of a performance assessment is estimated dose to the public and those doses are in most cases not expected to occur until the distant future, the performance assessment cannot be supported by comparing the modeled dose to observed doses. However, a performance assessment typically includes many submodels that estimate the impact of processes such as infiltration, leaching of radionuclides from waste, and transport through groundwater. Licensees can and should support the output from those submodels using indirect methods if the output from the submodels is not observable in the real world. Model support that involves multiple sources and types of information is generally more robust. Types of model support may include laboratory or field tests, comparison to analogous systems, natural analogs, independent process modeling, formal independent peer review, and comparison to monitoring data.

Upon development, and through early iterations, the performance assessment model (or submodels) is likely to go through a formal or informal calibration process. Calibration is a comparison of calculated outputs (e.g., intermediate outputs such as hydraulic head in an aquifer) with measured or observed data resulting in changes to the numerical calculation to better represent the observed data. While calibration is important, it does not necessarily result in confidence in prediction. Confidence in prediction is derived through comparison of the model with independent data not used in the original development or calibration of the model.

Absolute proof cannot be achieved in performance assessment modeling; however, adequate confidence in the model and modeling results is essential.

The quantity and quality of model support information should be commensurate with the significance of the system or subsystem to achieving protection. For example, if the capabilities needed for an engineered barrier are consistent with experience under similar conditions and the barriers have similar design and QA, then the model support could be considerably less than for a barrier with projected capabilities that significantly exceed experience with similar barriers. When considering prior experience, licensees should ensure that the environmental conditions for the relevant degradation mechanisms are reasonably similar since many degradation mechanisms can be sensitive to the environmental conditions.

Performance assessment modeling may include the projection of performance of engineered barriers, such as engineered covers or intruder barriers, for long periods of time. Licensees should consider natural analogs for barriers desired to have very long-term capabilities (e.g., thousands of years). The greatest uncertainties in predicting future performance arise from extrapolating the results of short-term tests and observations to long-term performance. Standard approaches to development of natural analogs frequently implicitly assume that the initial conditions persist; however, the actual application of a barrier could more appropriately be viewed as an evolving component of a larger dynamic system (Waugh and Richardson, 1997). For some types of barriers, natural analogs might provide information about the possible long-term changes and can be thought of as a long-term, uncontrolled experiment.

Licensees should consider the capability of a barrier when developing model support based on analogs. For example, because of their longevity, Native American earthen mounds may provide a reasonable analog for the erosional stability of an engineered cover but may not be a reasonable analog for other capabilities (e.g., the hydraulic performance of a cap) (Shetrone, 2004). When evaluating analogs, it is important to note that the structures that have persisted are most likely the durable structures (i.e., survivor bias). That is why licensees should consider analogs that have persisted, as well as those that may have experienced damage or failure. An additional complicating factor is that the initial conditions and past exposure environment for the analogs are not typically known and may only be estimated. However, developing an understanding of analogs increases the likelihood that a barrier may be implemented with sustainable long-term capabilities. Natural analogs are only one element of the technical basis for the capabilities of engineered barriers. Analog information should not be envisioned as providing absolute proof of future barrier performance; rather, it provides confidence that the barrier is likely to perform as intended. Analogs can also be applied to other aspects of the performance assessment models (e.g., radionuclide transport, geochemistry).

2.2.4.1 Peer Review, Expert Judgment, and Expert Elicitation

Different types of assessments are used in the development, review, and approval of LLW technical analyses. Scientists and engineers routinely use professional judgment in solving problems. This section covers the use of experts for review and for developing information to support technical analyses. NUREG-1563, “Branch Technical Position on the Use of Expert Elicitation in the High-Level Radioactive Waste Program,” issued November 1996 (NRC, 1996), defines “peer review,” “expert judgment,” and “expert elicitation” as applied to HLW disposal. The NRC staff believes that those definitions and descriptions also can be applied to LLW disposal. The purpose of this section is to concisely summarize information pertaining to the use of experts and to remind licensees of existing sources of guidance. Methods of reviewing

or supplying information to technical analyses may have different degrees of effectiveness and reliability. Information from experts is necessary when other means of obtaining requisite data or information are not practical. The types of information that experts can supply include data (e.g., hydraulic conductivity of basalt) as well as input analogous to model support (e.g., independent peer review of a hydrogeology model).

Peer review is distinguished from expert judgment in that peer review seeks judgments from experts regarding the soundness and quality of an existing or proposed scientific stance or solution to a problem. Expert judgment is information provided by experts that gives rise to or contributes to the generation of a scientific stance or solution to a given problem. Expert elicitation is a formal, highly structured, and well-documented process whereby expert judgments are obtained (NRC, 1996).

For information obtained from experts, licensees should describe and document the information developed as well as the process used to obtain the information. Documentation entails a description in sufficient detail to allow independent interpretation and understanding of the information without recourse to the author. Licensees should establish and describe the independence of the experts and their qualifications.

If a licensee uses peer review as a form of model support, the experts should have independence from the activity for their input to be of most value. Otherwise, the input is an extension of the original analysis. The licensee should document how the experts were selected and their qualifications. Licensees should describe the information that the experts reviewed and document when the experts' review and input were received. Scope and schedule are important in establishing the thoroughness of the review. The experts' reviews should have sufficient detail for independent evaluation by the NRC. Independent peer review is more subjective than some other forms of model support (e.g., field experiments) but may be necessary to support the technical analyses for LLW disposal especially for long-term analyses of LLW disposal.

Expert elicitation and expert judgment are means to supply information such as uncertain values or uncertainty distributions when data are unavailable or inadequate and cannot be obtained with direct methods such as experiments. Expert elicitation and judgment characterize the state of knowledge about uncertain information. Expert elicitation and judgment should not be used instead of "hard" data, but rather in conjunction with those data. NUREG-1563 provides general guidelines on those circumstances that may warrant the use of a formal process for obtaining judgments of experts, as well as acceptable procedures for conducting expert elicitation (NRC, 1996). NUREG-2255, "Guidance for Conducting Expert Elicitation in Risk-Informed Decision-Making Activities," issued September 2022 (NRC, 2022b), provides guidance on the process of eliciting and integrating expert judgment as input to decision-making activities. This report includes an introduction to expert elicitation and its use withing the NRC, the basic principles of using expert judgment, guidance, and a recommended process for conducting a formal expert elicitation. It includes supplemental information based on lessons learned.

2.2.4.2 Model Support Example—Engineered Cover Performance

The following example illustrates the type of effort that may be necessary to develop model support for the long-term performance of an engineered system. The DOE, Office of Legacy Management, Environmental Sciences Laboratory, developed a multi-faceted strategy combining monitoring, modeling, and natural analog studies to provide support for the estimated

long-term performance of engineered covers (Waugh, 2004; Waugh 2006). The strategy used various independent sources of information to develop an understanding of the performance of engineered covers. Laboratory experiments (e.g., the hydraulic conductivity of clay), field measurements (e.g., the moisture content of the cover materials and tailings), in-situ field tests (e.g., lysimeter studies), natural analogs, and traditional monitoring data (e.g., radon fluxes, groundwater concentrations of radionuclides) provided information to compare to estimates of cover performance. There is uncertainty in any type of observational data. However, use of multiple sources of confirmatory information allows for a more complete understanding of the uncertainties and, therefore, a reduced chance of false positive or false negative confirmation of system performance.

As part of the model support strategy, the DOE obtained field measurements using lysimeters to test the performance of cover designs under the Alternative Cover Assessment Project (ACAP) (Albright et al., 2004). The DOE performed comprehensive lysimeter tests of prototype covers at landfill sites across the country in climates ranging from arid to humid and from hot to cold. Some of the cover sites were at locations of full-scale covers operated by the DOE. The DOE monitored conventional, low-permeability, and alternative evapotranspiration cover designs in side-by-side comparisons. The ACAP prototype tests were conducted using 10 m by 20 m drainage lysimeters instrumented for direct measurement of runoff, soil water storage, lateral drainage, and percolation flux for a full-depth cover profile. The DOE used the lysimeter monitoring data to develop insights into cover performance.

Because some processes that may influence the performance of engineered covers are difficult to address with short-term field tests or existing numerical models, the DOE also used natural analogs to help identify and evaluate likely changes in environmental processes that may influence the performance of engineered covers (Waugh, 2006). The DOE used the natural analog information to (1) engineer cover systems that mimic favorable natural systems, (2) bound possible future conditions for input to predictive models and field tests, and (3) provide insights about the possible evolution of engineered covers as a basis for monitoring leading indicators of change. The DOE considered a variety of natural analogs, including analogs of future climate states at the Monticello, Utah site that were developed using paleoclimate data. A preliminary analysis of paleoclimate data for Monticello yielded average annual temperature and precipitation ranges of 2 to 10 degrees Celsius and 80 to 60 centimeters, respectively, corresponding to late glacial and Mid-Holocene periods. Additional natural analogs were developed for future soil development and ecological change.

2.2.5 Documentation

Communication, transparency, and clarity are essential to engage the public in licensing decisions. As discussed in Section 2.2.1.4, transparency and traceability should be applied to technical analyses (e.g., documentation, calculations, data). The primary standard for documentation is to provide sufficient detail such that the analyses can be understood without recourse to the originator of the documentation. Various stakeholders will have different backgrounds and education levels. The safety case should be documented using plain language that will be understood by a broad audience while preserving the pertinent technical details. The requirement to develop a safety case should help ensure that a plain-language description of the basis for the safety decision is available to the public.

Technical documentation should clearly state assumptions, limitations, and conditions. Data should be identified, and the source of the data should be provided. Use of data tracking

numbers is a good practice that can address some QA issues associated with data and provide transparency and traceability. Source references (i.e., those from which the data or information originated) should be provided. Regulatory requirements should be presented.

In general, documentation supporting licensing decisions should be publicly available. This includes the licensee's information and the regulator's review of the licensee's information. It is especially important that the disposed radiological inventory and the characteristics of that inventory are publicly available if unforeseen problems occur well into the future. References should be provided. If restricted by copyright or other limitation, a source where the reference can be obtained should be provided. The NRC staff understands that the use of complex models has the potential to be a challenge to public involvement; however, the licensee should attempt to alleviate this burden. Regulatory decisions should be as transparent as possible with a clear description of the process and basis for the decisions.

2.3 Assessment Context

2.3.1 Context of the Performance Assessment

The first step in the performance assessment process is the assessment context. To develop the context of a performance assessment, a licensee should answer the following questions:

- What is being assessed?
- Why is it being assessed?
- What is the scope of the assessment?

Components of an assessment context may include the assessment purpose, regulatory framework, assessment end points, assessment philosophy, waste characteristics, disposal system characteristics, and assessment timeframes. For LLW, the regulatory framework is found in 10 CFR Part 61, including the performance objectives in Subpart C. However, the purpose of conducting a performance assessment may vary (e.g., safety decision, resource optimization). The licensee should consider the audience for the performance assessment results. Different strategies for the performance assessment may be used (e.g., conservative versus realistic, simple versus complex, deterministic versus probabilistic). A well-defined assessment context can be used to determine data, computational, and model abstraction needs. For example, if the assessment context calls for a simple modeling approach, a relatively simple mathematical model abstracted from the conceptual model may be sufficient. In addition, the assessment context may provide a comparison point to other performance assessments. For example, if another organization has modeled a similar site, a comparison, and any possible discrepancies, of the model outputs will be easier to understand. For a description of an assessment context as it pertains to biosphere models for geologic disposal, the BIOMASS program, as documented by the International Atomic Energy Agency (IAEA, 2003), provides information and guidance.

2.3.2 Approach to Different Timeframes

This section describes the information that a licensee should provide, and a reviewer should evaluate with respect to the timeframe for the inadvertent intruder assessment, performance assessment, and site stability assessment for the compliance period and for the analyses for the performance period. For operational safety assessments, the timeframe will be the period of operations of the facility.

When completing technical analyses, the licensee should select the period of time over which the potential future behavior of the disposal system will be evaluated against the performance objectives; this period of time is termed the "analysis timeframe." Licensees conduct performance assessments to understand how a waste disposal system may perform with respect to limiting releases to offsite members of the public. Stakeholders use performance assessments to understand the potential impacts of uncertainties. Numerous sources of uncertainty are associated with projecting the future radiological risks from waste disposal for thousands of years, including, but not limited to, natural, engineering, and societal sources. Section 2.2.3 discusses the types of uncertainties that licensees commonly consider explicitly in technical analyses.

One of the outputs of technical analyses for LLW disposal is the projected dose to a member of the public from radioactivity released to the environment. The projected doses are compared to regulatory requirements and the regulator determines whether there is an acceptable degree of confidence that the dose is likely to be lower than the regulatory limit. In the NRC's terminology, that degree of confidence is described as "reasonable assurance." The results of a compliance analysis should not be interpreted as unequivocal proof of the expected behavior of a waste disposal facility, because of the uncertainties associated with the projected radiological risk over long time periods. Over extended periods of time, uncertainties associated with the performance of natural and engineered systems may increase, and uncertainties associated with human behavior definitely increase. In some cases, unmanageable uncertainty may be a suitable reason not to dispose of waste if the consequences are unacceptable. The technical analyses supply information to inform decision-makers and the public.

Analyses timeframes are specified in 10 CFR Part 61. For disposal of waste that does not contain significant quantities of long-lived radionuclides, the compliance period is 1,000 years and performance period analyses are not required. If the waste does contain significant quantities of long-lived radionuclides, then the compliance period is 10,000 years and performance period analyses are required. When implemented properly, this approach should accomplish the main goal of licensees communicating the near- and long-term risks in an appropriate uncertainty context to regulators and other stakeholders. The analyses timeframes are defined in 10 CFR Part 61 as follows:

- *Compliance period* — Compliance period means the time from the completion of site closure to 1,000 years after site closure for disposal sites that do not contain significant quantities of long-lived radionuclides. For sites that contain significant quantities of long-lived radionuclides, the compliance period ends 10,000 years after closure of the disposal site.

- *Performance period* — Performance period is the timeframe established for considering waste and disposal site characteristics to evaluate the performance of the disposal site after the compliance period.

The principle of transgenerational equity is applied to the disposal of radioactive waste. The principle means that protection is afforded to future generations based on what is deemed acceptable for the present generation. This protection of future generations can be achieved in different ways. Some programs use long-term technical analyses whereas others do not. Correspondingly, the establishment of timeframes for the technical analyses has taken different

approaches. These approaches are briefly summarized to provide context for the NRC requirements.

Figure 2-3 shows the concentration of select long-lived alpha-emitting radionuclides that have been disposed in operating facilities. The radionuclides included are americium (Am)-241, plutonium (Pu)-238, Pu-239, and Pu-240. It is understood that the figure does not include the complete list of that class of radionuclides; however, it can be difficult to identify the inventory disposed in the diverse set of facilities examined. The total curies are divided by the total volume of waste to provide an average concentration. The concentrations are then compared to the NRC's Class A waste limits to provide a fraction of the Class A limits for an apples-to-apples comparison. Figure 2-4 is a similar figure showing the concentration of long-lived mobile radionuclides disposed in operating facilities. The radionuclides included are carbon (C)-14, iodine (I)-129, and technetium (Tc)-99. These facilities are mostly near-surface, but some geologic facilities have been included because some programs dispose of all waste in deep geologic disposal because of uncertainties associated with near-surface disposal. These programs believe that the uncertainties of near-surface disposal are significant and can be better managed through geologic disposal.

Many programs use a waste classification system consistent with that outlined by the IAEA (IAEA, 2009b). That system develops classes of waste based on both concentrations and half-lives. By comparison, the NRC waste classification system allows mixtures of both long- and short-lived radionuclides within each waste class. A common dividing line between wastes that are acceptable for near-surface disposal and those that are considered intermediate-level waste (ILW) or long-lived LLW (LLW-LL) and are destined for some form of deeper disposal is 0.37 kBq/g (10 nCi/g) for long-lived radionuclides. It is important to note that the 0.37 kBq/g (10 nCi/g) concentration is not limited to long-lived alpha-emitting radionuclides. Many international programs specify a limit for long-lived radionuclides averaged over the waste in the disposal facility of 0.37 kBq/g (10 nCi/g) (the y-axis value of 1 in Figure 2-3). Waste with concentrations greater than this is ILW or LLW-LL. There are few operating ILW facilities in the world. In the United States, the DOE has disposed of TRU that is similar to ILW internationally. Some programs mitigate the impacts of the uncertainties of near-surface disposal by allowing only waste that will decay to levels such that the site can be released for unrestricted release at the end of a control period. The most common control period is 300 years. For these programs, long-term analyses of near-surface disposal facilities are not necessary.

The shaded area in Figure 2-3 approximates the NRC's requirements for analysis timeframes overlain against practice, to date, for technical analyses. It is an approximation because the transition from 1,000 years to 10,000 years is based on an evaluation of what is a significant quantity of long-lived radionuclides (see Appendix H). In addition, the upper bound on what may be acceptable at a disposal site will be determined by the site-specific technical analyses.

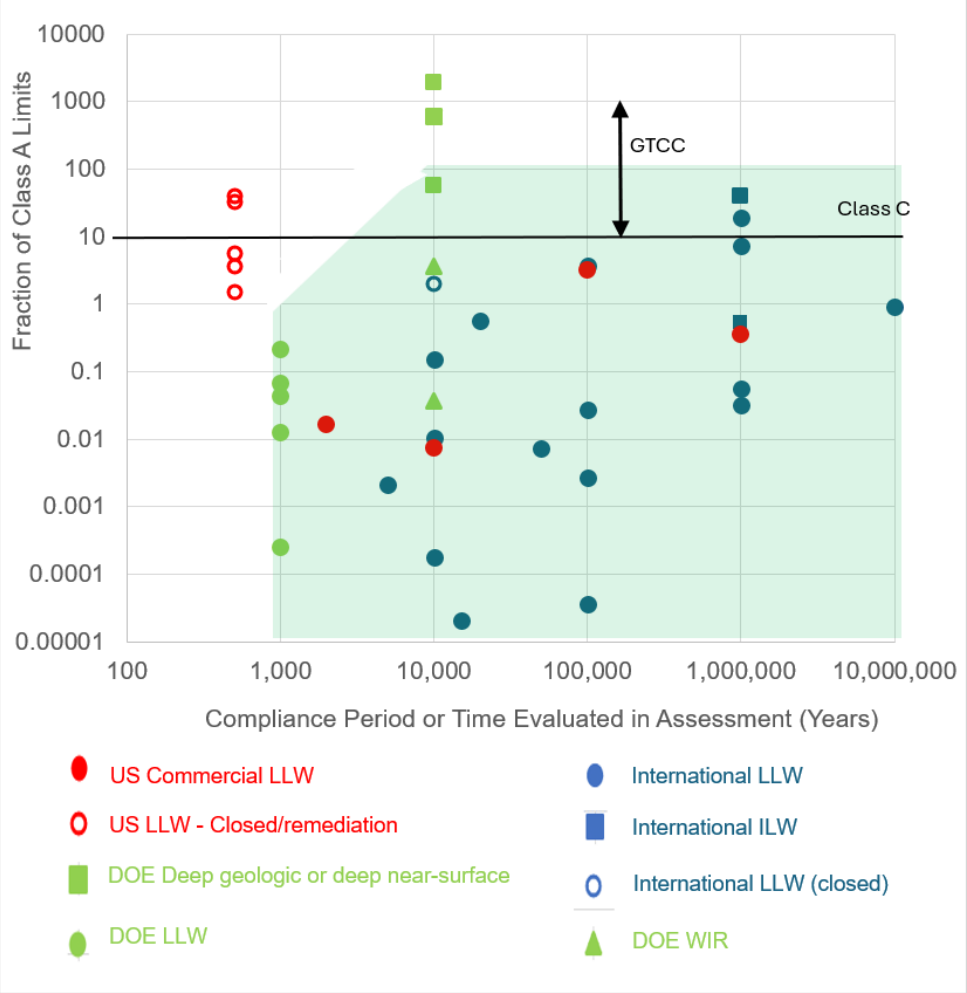


Figure 2-3 Average Concentrations of Long-lived Alpha-emitting Radionuclides in Disposal Facilities

A few important observations follow:

1) The NRC's current requirements for analyses timeframes are consistent with the analyses performed for the vast majority of operating facilities.

2) The analyses timeframes evaluated were not a barrier to licensing of these facilities.

3) The early U.S. facilities that were licensed before the issuance of 10 CFR Part 61 were mostly design based and had limited technical analyses. Most of these facilities are closed and are undergoing remediation because of performance issues.

4) At higher concentrations, the disposal method switches from shallow near-surface to deep near-surface (about 100 m or greater) or geologic disposal.

5) Across U.S. and international programs, longer analyses, different disposal methods, and higher waste conditioning are used for higher concentrations of long-lived radionuclides.

The data plotted for U.S. facilities are compliance periods. The data plotted for international facilities are mostly compliance periods, but the regulatory requirements associated with analyses were not clear for some programs. Some programs implement a peak dose approach, and therefore, the analyses are completed for sufficient time to identify the peak. That is why the x-axis is labeled “compliance period” or “time evaluated” in the assessment. In the United States, the DOE uses a 1,000-year compliance period, but they combine it with an evaluation period to perform sensitivity analyses. The DOE is self-regulating and operates as the licensee and regulator for its disposal facilities; therefore, it can implement changes to disposal actions based on the results of the analyses performed for the evaluation period. Some DOE analyses extend to 100,000 years.

Because all currently (as of 2026) the operating commercial LLW facilities in the United States are in Agreement States, licensees must meet the requirements established in those States. Based on the compatibility category of the compliance period definition in NRC regulations, an Agreement State may be more restrictive than the NRC. The NRC requirement for the compliance period is 1,000 years if a disposal site does not contain significant quantities of long-lived radionuclides, otherwise the compliance period is 10,000 years. Performance period analyses are required if the longer compliance period is used. Licensees should conduct a performance assessment and inadvertent intruder assessment for the compliance period. The NRC requirement is flexible such that different compliance periods can be used for different facilities and wastes. Some LLW may pose long-term risks while other types of LLW may not. Appendix H provides guidance on how to determine if a site will contain significant quantities of long-lived radionuclides. The guidance provides options for use of generic values (concentrations), as well as calculation of site-specific values based on the most significant parameters and pathways.

As shown in Figure 2-4, there is no prescribed upper limit on the concentration of long-lived mobile radionuclides that may be disposed (e.g., for GTCC waste). The dominant risk from these radionuclides is primarily from releases to an offsite member of the public rather than disturbance by an inadvertent intruder. Because the characteristics of the release pathways can differ substantially from site to site, there was not an effective way to prescribe generic concentration limits for offsite release pathways when 10 CFR Part 61 was developed in the 1980’s. The concentration of long-lived mobile radionuclides that have been disposed have been generally less than one-tenth of the Class A limits across US and international programs. A few facilities have waste with greater concentrations—those facilities are deep near-surface or geologic disposal facilities. To review a licensee’s assessment of what is a significant quantity, a regulator may consider this value as to when more scrutiny is needed of the licensee’s analysis and arguments.

Uncertainty is not an appropriate basis for truncating technical analyses without corresponding additional actions to ensure that the uncertainties are appropriately accounted for and managed. Radioactive waste disposal is an elective action and throughout most programs, different options are available or are being developed for different types of waste. For example, near-surface disposal of HLW has not been proposed by any program. Technical analyses are one of the important tools used to ensure that uncertainties are identified and managed so that public health and safety may be protected, but they are not the only tool. For example, a licensee proposes to dispose of significant quantities of DU in a near-surface disposal facility. If the licensee or regulator asserts that the long-term analyses are too uncertain to support regulatory decision making, then alternative restrictions may be implemented, such as license conditions. Disposal depth (10 to 30 m) below the natural land surface is very effective at

mitigating the future impacts from DU disposal at arid sites where the dominant releases are from radon emissions; therefore, an appropriate design-based license condition prescribing a certain depth may be used to mitigate uncertainties associated with the technical analyses.

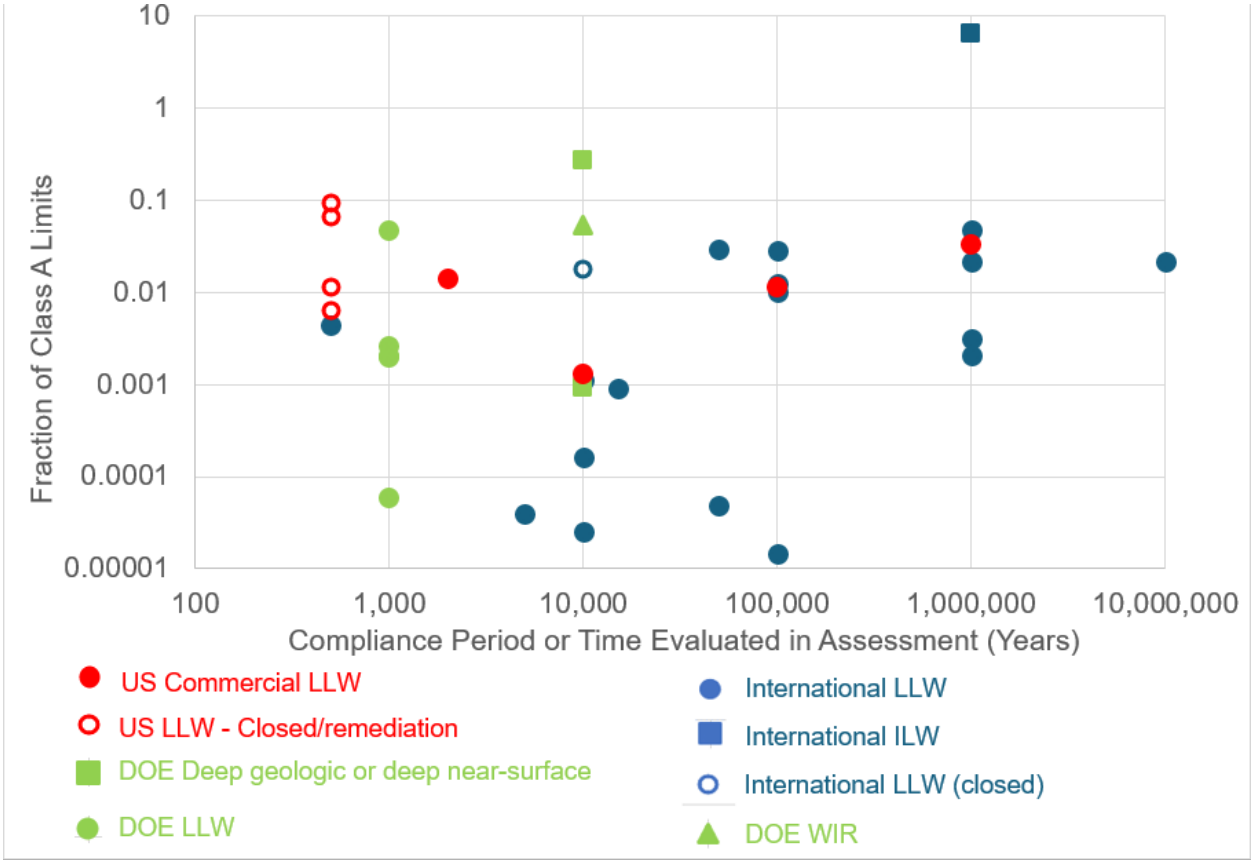


Figure 2-4 Average Concentrations of Long-lived Mobile Radionuclides in Operating Facilities

In 10 CFR 61.2, the NRC defines the compliance period of 1,000 years as the timeframe over which a licensee must demonstrate compliance with the quantitative limits found in 10 CFR 61.41(a) and 10 CFR 61.42(a) when waste does not contain significant quantities of long-lived radionuclides; otherwise, the timeframe is 10,000 years. Licensees should conduct a performance assessment and inadvertent intruder assessment for the compliance period. A quantitative assessment of disposal facility performance should be developed covering the compliance period. The requirement for site stability in 10 CFR 61.44 mentions “long-term” for the evaluation timeframe rather than the compliance period. The site stability assessment will generally encompass the compliance period, but the objectives of site stability change over the timeframes of the compliance period. For example, at site closure, site stability requirements are designed to ensure that a site will not experience significant erosion or differential settlement. Early disposal facilities (before 10 CFR Part 61) experienced significant stability issues that led to closure and remedial actions. By the time of completion of postclosure observation and maintenance, the disposal site should be stable. At the end of the active institutional control period, instability associated with previous operations should be resolved such that future instability would be associated only with long-term processes.

The 10 CFR 61.41(a) and 10 CFR 61.42(a) assessments must account for these long-term processes and their impacts. Therefore, it is redundant to specify a compliance period for 10 CFR 61.44. The waste characteristics (e.g., the amount of short- and long-lived wastes), complexity of the disposal facility design, and complexity of the disposal site and surrounding environment will influence the level of detail that should be provided to support the technical analyses.

The compliance period that should be used depends on the type of waste that will be disposed at the site. Waste that does not contain significant quantities of long-lived radionuclides should be analyzed for a 1,000-year compliance period. Waste that does contain significant quantities of long-lived radionuclides should be analyzed for a 10,000-year compliance period, as well as a longer-term performance period.

Appendix H provides guidance on how a licensee or regulator may determine if a waste disposal site contains significant quantities of long-lived radionuclides. Other methods may be suitable and will be evaluated on a case-by-case basis. In all cases, a licensee may elect to use the 10,000-year compliance period, regardless of the actual quantities and concentrations of long-lived radionuclides that are expected to be disposed.

The following is the recommended screening process to determine whether a site has significant quantities of long-lived radionuclides:

(1) generic concentration-based screening
(2) simplified dose-based screening
(3) site-specific dose analysis

Appendix H describes in detail the methods that may be used to determine whether significant quantities of long-lived radionuclides will be disposed of at specific disposal sites.

The initial step in determining whether a disposal site will contain significant quantities of long-lived radionuclides is consideration of the projected inventory. If a disposal site will not contain any long-lived radionuclides or will contain very minimal quantities of long-lived radionuclides, then further analysis is not necessary since it is clear that a 1,000-year compliance period for the technical analyses will be sufficient. Significance of long-lived radionuclides should not be determined by comparison with the overall radioactivity in the initially disposed inventory. Significance should be determined by the propensity of the radioactivity remaining after decay (at future release times) to potentially exceed the performance objectives.

Significance as used in the term “significant quantities” is defined with respect to the protection of public health and safety. Protection of public health and safety is achieved by demonstrating the performance objectives in 10 CFR 61.41 and 10 CFR 61.42 will be met. The performance objective for protection of the inadvertent intruder (10 CFR 61.42) is generally based on more constrained intruder scenarios and is less sensitive to hydrologic and environmental variability compared to protection of the general population from release of radioactivity (10 CFR 61.41). A concentration-based approach using tabular values similar to those in the 10 CFR 61.55 waste classification tables can provide for reliable determination of significant quantities with respect to 10 CFR 61.42. A tabular approach is more difficult to use for determining significant quantities with respect to 10 CFR 61.41. Site-specific design and hydrogeology and potential

disruptive processes and events can influence the pathways that are important, and the magnitude of risk associated with the different pathways.

During the original development of the 10 CFR 61.55 waste classification tables, long-lived alpha-emitting non-transuranic radionuclides were not included because it was expected that LLW would not contain those isotopes in sufficient quantities and concentrations that their disposal would have an impact on public health and safety (NRC, 1982b). However, there is no compelling reason for the long-lived non-transuranic radionuclides to be treated differently in the technical analyses if both transuranic and non-transuranic radionuclides are included in the wastes proposed for disposal. The radiological risk will be determined in part by the dose conversion factors of individual radionuclides and the concentration of those radionuclides. Dose conversion factors vary from radionuclide to radionuclide (EPA, 1988). The NRC staff decided to reduce this variability when deriving the 0.37 kBq/g (10 nCi/g) concentrations value for all transuranic radionuclides in Class A waste (NRC, 1982b). Because the dose conversion factors for non-transuranic radionuclides are generally comparable to those of transuranic radionuclides, the NRC staff believes it is appropriate to simplify the consideration of variability. The concentrations provided in Table H-1 in Appendix H should only be used to determine if a 10,000-year compliance period is necessary, not to determine compliance with performance objectives.

A licensee should estimate the disposal site-averaged concentrations of radionuclides to determine if a longer compliance period is necessary. Disposal site-averaged concentrations can include the volume of the waste, uncontaminated materials used to stabilize waste or reduce void space within waste packages, the volume of uncontaminated materials placed in the disposal cells, and the volume of engineered or natural materials used to construct the disposal cells. The material between disposal cells may also be included as long as the disposal cells are located proximally as part of a single disposal facility. The licensee should base the disposal site-averaged concentrations on the total amount of waste averaged over the total volume of all disposal cells. Though the disposal site definition includes a *buffer zone*, the buffer zone should not be included in the calculations. A reviewer should consider the variability in radionuclide concentrations over the disposal site.

Most disposal facilities are expected to dispose of waste containing a mixture of different radionuclides. To determine if a disposal facility will contain significant quantities of long-lived radionuclides, a sum of fractions (SOF) approach should be used. Licensees should estimate the disposal site-averaged concentrations of each radionuclide. They should divide the resultant values by the concentrations found in Table H-1 to estimate a fraction for each radionuclide, and then sum the fractions over all radionuclides. If the total is greater than 1.0, then the site will contain significant quantities of long-lived radionuclides and a 10,000-year compliance period should be used. Examples 2.1 and 2.2 provide SOF calculations.

2.3.2.1 Performance Period

The performance period is the time after the compliance period when longer-term doses could result from the disposal of significant quantities of long-lived radionuclides. The performance period begins at the end of the compliance period and extends as long as necessary to demonstrate that the metric of the performance period can be met. Licensees should perform a quantitative assessment though uncertainties may be large. A qualitative interpretation of the quantitative results by the licensee is appropriate in most cases. The objective of performance period analyses is for a licensee to demonstrate that releases from long-lived waste have been

reduced to the extent reasonably achievable and that the facility has been designed with consideration of the potential long-term radiological impacts, consistent with available data and current scientific understanding. The analyses should identify the features of the design and site characteristics that will reduce long-term impacts and describe the capabilities of these features. Analyses for long-lived waste should provide the range of peak annual doses that are projected to occur after 10,000 years following site closure, or other metrics such as concentrations of radioactivity in the environment and fluxes to the environment. Comparison of releases from the facility to natural fluxes and concentrations of radioactivity in the environment is appropriate. The long-term performance period analyses are designed to complement the technical analyses performed for the compliance demonstration.

As described in more detail in Chapter 6.0, different approaches are acceptable for providing the necessary information for performance period analyses. A licensee should provide sufficient information and analyses for the performance period to demonstrate that 10 CFR 61.41(b) and 10 CFR 61.42(b) performance objectives will be met. The licensee should use available data and current scientific understanding to assess the performance of the waste disposal facility, including uncertainties.

Example 2.1
A licensee wishes to dispose of waste at a disposal site that does not have a potable groundwater pathway or any credible mechanisms for release other than from disturbance by inadvertent intruders. The total volume of disposal cells for existing waste is 400,000 m^3. The inventory of waste located in the facility comprises 50,000 m^3 of C-14 containing waste at 0.2 curies per cubic meter (Ci/m^3), 200,000 m^3 of waste containing C-14 at 0.1 Ci/m^3 and I-129 at 0.002 Ci/m^3, and 50,000 m^3 of Tc-99 containing waste at 0.01 Ci/m^3. The uncontaminated fill and material used to construct the cells represents 100,000 m^3.

Conclusion: The licensee examines the characteristics of the hypothetical site presented in Appendix H and determines that its site has characteristics that are consistent or conservative with respect to the hypothetical site. The licensee uses the values in Table H-1 to calculate the volume-averaged SOF per the following equation. This equation is used to calculate the SOF for *n* waste streams containing *m* isotopes. *V* is the volume, *C* is the concentration on a volumetric basis, and *CA* is the value from Table H-1 for each radionuclide.

$$SOF = \frac{1}{V_T}\sum_{i=1}^{n}\left(V_i \sum_{j=1}^{m}\frac{C_{i,j}}{CA_{i,j}}\right)$$

$$SOF = \frac{1}{400{,}000\ m^3} * \left(50{,}000\ m^3\left(\frac{0.2}{0.48}\right) + 200{,}000\ m^3\left(\frac{0.1}{0.48} + \frac{0.002}{0.0036}\right) + 50{,}000\ m^3\left(\frac{0.01}{0.1}\right)\right) = 0.45$$

Because the SOF is less than 1, a 1,000-year compliance period can be used, and performance period analyses are not required.

Example 2.2

The licensee from Example 2.1 would like to dispose of a new waste stream but it is unsure if it would need to develop performance and inadvertent intruder assessments for a 10,000-year compliance period and performance period analyses.

The new waste stream is a particulate waste. The raw particulate waste contains average concentrations of 0.7 Ci/m^3 Tc-99, 1.2 Ci/m^3 C-14, and 0.09 Ci/m^3 of Pu-239. The total volume of waste is 100,000 m^3. To reduce the potential for dispersion, the particulate waste is solidified in grout before disposal. The ratio of grout to waste inside the disposal package is 3 to 1 (stabilizer ratio). Because the density of the grout and the waste stream are similar, the 3 to 1 ratio holds for both a mass and volume basis.

The drums of waste will be stacked within the disposal cells achieving a disposal cell packing efficiency of 67 percent (i.e., the ratio of cell volume occupied by waste packages to the total internal volume of the cell—*Cell Eff.*). In addition, the disposal cells are constructed of a variety of natural materials to provide structural stability, to reduce water inflow to the waste, and to provide chemical retention of the waste. The volume of material comprising the disposal cells is 30 precent of the total internal cell volume available for disposal (i.e., *Inert frac.*).

Conclusion: First the licensee calculates the cell volume required for the new waste, V_{Tn}:

$$V_{Tn} = Waste\ Volume * (1 + Stabilizer\ Ratio) * \frac{1}{Cell\ Eff.} * (1 + Inert\ frac.)$$

$$V_{Tn} = 100{,}000\ m^3 * (1 + 3) * \frac{1}{0.67} * (1 + 0.3) = 776{,}000\ m^3$$

Next, the licensee calculates the SOF for the new waste stream (SOF_n) containing m isotopes, where C_m, V, and CA_m are the concentration, volume, and long-lived waste concentration (for isotope m from Table H-1). Only one new waste stream is considered, so the equation from Example 2.1 is simplified:

$$SOF_n = \frac{V}{V_{Tn}} \sum_{i=1}^{m} \frac{C_i}{CA_i}$$

$$SOF_n = \frac{100{,}000}{776{,}000} * \left(\frac{0.7}{0.1} + \frac{1.2}{0.48} + \frac{0.09}{0.025}\right) = 1.69$$

Then, the licensee calculates the average SOF for both old and new waste where V_T is the combined volume of old and new waste (total disposal cell volumes) and SOF_o is the sum of fractions for the old waste:

$$SOF = \frac{1}{V_T} * (SOF_n * V_{Tn} + SOF_o * V_{TO})$$

$$SOF = \frac{1}{(776{,}000 + 400{,}000)} * (1.69 * 776{,}000 + 0.45 * 400{,}000) = 1.24$$

In this case, the SOF is greater than 1 on a disposal site-averaged basis when the new waste stream is combined with the existing waste stream. Therefore, a 10,000-year compliance period and performance period analyses are required.

The performance period analyses should cover the period of geologic stability at the disposal site, limited to a maximum of 1 million years or the peak dose[4] considering uncertainty. It would not be appropriate to constrain the analyses to the period of near-surface geomorphic and geologic stability, as one of the reasons for undertaking the performance period analyses is for a licensee to communicate to decision-makers the potential range of consequences from the disposal action. Near-surface geologic instability may result from a process such as fluvial erosion, which could have severe impacts at an unstable site. Near-surface geologic instability may indicate that the site is unsuitable for the disposal of significant quantities of long-lived radioactive waste. If the analysis for LLW disposal was limited to the period of near-surface geologic stability, the analysis could be truncated prematurely or not performed, and the long-term risks and uncertainties may not be communicated. Chapter 6.0 provides more detail on specific technical issues that may be relevant to technical analyses for the performance period.

2.3.2.2 Site Characteristics

Site characteristics are identified in 10 CFR 61.50 that either must be avoided or must be present at a proposed disposal site. Site characteristics play an important role in ensuring that the performance objectives are likely to be met. A licensee needs to consider past, present, and future site characteristics, considering the radiological characteristics of the waste. A licensee should evaluate site characteristics for at least a 500-year timeframe and understand that the time period evaluated in the assessments will be different for different types of waste. Licensees are allowed some flexibility to ensure that they can consider site characteristics in a risk-informed manner. The site suitability requirements are designed to help ensure that the long-term performance objectives of 10 CFR Part 61, Subpart C, are met. A cornerstone of radioactive waste disposal is the stability of the disposal system. Sites for which the site characteristics requirements cannot be satisfied are unlikely to meet this regulatory objective.

The hydrological site characteristics identified in 10 CFR 61.50(a)(2) are required[5] for the next 500 years. For example, disposal should not be permitted at a site where the facility is expected to be in the 100-year floodplain over the next 500 years. The regulation in 10 CFR 61.50(a)(5) specifies other characteristics of the site that should not be present at any timeframe because they significantly affect the ability of the disposal site to meet the performance objectives of 10 CFR Part 61, Subpart C (e.g., natural resources, tectonic processes). Appendix B presents currently understood hazard maps related to the features and phenomena of 10 CFR 61.50 criteria.

Whether a licensee's consideration of site characteristics needs to be extended to the end of the compliance period (i.e., 1,000 or 10,000 years) or into the performance period depends on the type of waste that will be disposed. For a disposal facility that accepts only short-lived waste and contains minimal quantities of long-lived waste, a licensee's consideration of the site characteristics over the next 500 years would ensure that a proper site has been selected and that it would be capable of being characterized, modeled, analyzed, and monitored. Most of the radioactivity decays in place in this site scenario. Licensees are not prohibited from considering site characteristics for more than 500 years, which may enhance the robustness of their

[4] As discussed in Chapter 6.0, the performance period analyses may include metrics other than dose (e.g., concentration, fluxes).

[5] In 10 CFR 61.50, the NRC lists the hydrological characteristics a site is required to have, as well as other hydrological characteristics a site must not have.

technical evaluation. However, a consideration of site characteristics for more than 500 years for the disposal of waste with limited long-lived radioactivity would be unnecessary. To determine whether the amount of waste to dispose is a minimal quantity, a licensee may simply calculate the product of the projected facility volume (or mass) and the concentrations provided in Table H-1, adjust the calculated values for the unique characteristics of the site and waste, then apply the SOF rule for mixtures of radionuclides described in 10 CFR 61.55(a)(7). When considering site characteristics in this context, a "minimum quantity" is defined as the product of volume and concentration of waste that corresponds to an SOF of 0.1.

The following are the timeframes that reviewers should use to evaluate the site characteristics requirements to determine whether the requirements of 10 CFR 61.50 are met:

(A) $C < 0.1 \Longrightarrow$ evaluate 500 years

(B) $0.1 < C < 1 \Longrightarrow$ evaluate 10,000 years

(C) $C > 1 \Longrightarrow$ evaluate performance period

where C is the actual or projected disposal site average waste concentration SOF for long-lived waste (see Example 2.3). The reason why the recommended value of C is set to 0.1 and not 1 to determine when 500 years is sufficient is because some conservatism is warranted when establishing the scope of the assessment. Errors of omission of significant characteristics are more serious than including characteristics that are later determined to have little influence on the results. The evaluation of a licensee's site characteristics information may be qualitative because modern technical analyses are site-specific and should include all important phenomena. If a licensee cannot demonstrate that a site characteristic is present (or absent) (i.e., that the site characteristics requirements can be met), then the licensee should provide a basis and seek an exemption to the requirement.

The calculation of disposal site average waste concentration includes the waste, disposal units, backfill materials, and materials between disposal units but does not include the buffer zone. It is intended to include the activity in the waste at the time of disposal but also account for peak concentrations resulting from decay/ingrowth for those short-lived isotopes that produce long-lived alpha-emitting progeny. Because Table H-1 provides values for commonly disposed long-lived alpha-emitting radionuclides, most isotopes of concern (including uranium isotopes) will be considered using this approach. However, in some circumstances, a disposal facility's inventory may contain other long-lived isotopes. If there are long-lived isotopes observed to be key contributors to projected risk that are not included in Table H-1 (e.g., chlorine (Cl)-36), a licensee should evaluate the site characteristics over a timeframe as long as necessary to support the relevant performance demonstration. In most cases, the licensee would have performed this iterative evaluation (from characterization to performance analyses and back to characterization) before submitting the analyses to the regulator.

Example 2.3
A licensee performs analyses of a potential disposal site for disposal of waste with characteristics described in Example 2.1. The licensee had calculated the SOF for the waste as 0.45 and determined that a 1,000-year compliance period was required. The licensee uses a period of 500 years to evaluate site characteristics but wonders if they need to evaluate a longer timeframe to establish site characteristics.

Conclusion: Because the SOF was above 0.1 but less than 1(see item B above), the licensee should consider a period of 10,000 years when developing the site characteristics. In general, the timeframe of consideration for developing the site characteristics should be comparable to or longer than the compliance period being evaluated. The NRC staff should ask for additional information to support the development of site characteristics for the disposal site.

2.4 System Description

The next step in the performance assessment process is for the licensee to describe the LLW disposal system and the natural environment of the site and surroundings. The objective of the system description review is to ensure that the information used to develop the performance assessment models and the information describing the overall disposal system have been adequately described. The description should be adequate to allow an independent reviewer to understand the LLW disposal system. The system description should describe, at a minimum, the following:

- site
- natural environment
- disposal facility
- major barriers to isolate and contain waste including the wasteform
- interaction of the site and disposal facility
- waste to be disposed of including its radiological, chemical, and physical characteristics
- potential disruptive processes
- characteristics of members of the public potentially affected by the facility

The specific technical information that must be provided is identified in 10 CFR 61.12. In the system description the licensee should provide estimates of the temporal changes to the aforementioned information, especially for disposal of long-lived waste. A practical metric to determine if the system description is adequate is if the system can be understood without seeking clarification from the document authors.

2.5 Scenario Development

The third step in the performance assessment process, site scenario development (also referred to as “scenario development”), is the development of the scope of the analysis that will be implemented in the conceptual and numerical models. Development of a model that represents current and future features, events, processes (FEPs), and their interactions, is a complex process. Formal approaches to scenario development are usually either bottom-up or top-down

(see Section 2.5.3 for more detail). The bottom-up approach involves the identification, categorization, and systematic screening of FEPs. The bottom-up approach is commonly used for complex sites. The top-down approach uses analyses such as a *safety assessment* and the identified safety functions to develop site scenarios (see Section 2.5.3.2.1). Both approaches may be iterative.

Typically, a process or event acts upon a feature, and as time progresses, processes and events (both can be referred to as *phenomena*) act to modify the system. As previously defined, FEP stands for feature, event, and process. A feature is an object, structure, or characteristic that has a potential to affect the performance of the disposal system. Examples include rocks within an erosion layer of an engineered cover or a drainage layer of an engineered cover. An event is a change, usually abrupt, that has the potential to affect the performance of the disposal system and that occurs during an interval that is short compared to the analysis timeframe. Examples of events that cause relatively rapid change are earthquakes, floods, storms, well drilling, and excavation. A process is a change, usually gradual, that has the potential to affect the performance of the disposal system and that occurs during all or a significant part of the analysis timeframe. Examples of processes that cause relatively gradual change are radionuclide transport, differential settlement, leaching, and erosion. In the bottom-up approach, a comprehensive set of FEPs should capture the features and phenomena that are potentially relevant to the near- and long-term performance of a disposal system. A comprehensive set of possible safety functions should also capture the components and their functions that are potentially relevant to the near- and long-term performance of a disposal system. Safety functions are used in the top-down approach to scenario development and are defined as a function through which a component of the disposal system contributes to safety and achieves its safety objective. Section 2.5.3 discusses these two main approaches to scenario development.

For the bottom-up approach, the FEP analysis developed by a licensee should produce an FEP list that is broad enough to result in a systematically categorized, but manageable, number of FEPs, yet specific enough to provide the complexity required for screening, modeling, or both. From this set of potentially relevant FEPs, a licensee can define a subset of FEPs that are used to identify a probable future evolution of the disposal site (i.e., a site scenario). Remaining FEPs not incorporated into the original or *central site scenario* (also referred to as the “central scenario”) may include disruptive events. A central scenario is the most likely evolutionary path a disposal site will take consistent with available data and current scientific understanding (i.e., the most probable site scenario) (Section 2.5.3.1.2 discusses probability in the context of scenario development). Relevant FEPs, especially plausible events, not incorporated into the central scenario form the basis for *alternative site scenarios*. An alternative site scenario is a plausible evolutionary path a disposal path can take. It is less likely to occur than the central scenario and may include disruptive events (e.g., earthquakes, volcanoes). Usually, the central scenario does not include disruptive events while alternative site scenarios for the same site could include disruptive events, depending on the results of the scenario development.

The licensee can develop a connected sequence of features and processes describing the behavior of the system of concern. This description of how the disposal system will function and how the site behaves within a given site scenario, is the conceptual model. A qualitative description of the conceptual model would include how the incorporated FEPs (e.g., precipitation, water flow, dissolution, degradation, erosion) and significant barriers interact with one another and how the site functions in a given site scenario. Plausible conceptual models of a system are estimates of how the system may function. The distinction between a site

scenario (in which an event, or lack of events, dictates how the site will evolve) and a conceptual model (which details how the system functions in a site scenario) can be somewhat blurred. It is important that the complete set of site scenarios developed by a licensee represents the full range of plausible future states of the disposal system, and that the conceptual model for the central scenario of the site, or a set of alternative conceptual models if there is significant model uncertainty, incorporates the retained FEPs.

Site scenarios are assembled and classified based on their likelihood of occurrence. As previously discussed, this document uses the term "central scenario" for the most probable site scenario, although "base case" and "nominal" scenarios are also frequently used to describe the expected future state of the system. Alternative site scenarios are typically considered less probable, but still plausible, while implausible "what if" site scenarios may be used to explore the robustness of the system. "Stylized scenarios," or select *receptor behaviors and activities*, are used to represent future human actions. Receptor activities and behaviors describe the end process by which people may become exposed to radiation. A discussion of the activities and behaviors of a residential farmer receptor, for example, is a description of the pathways leading to possible exposure and of the behavior and lifestyle of the hypothetical receptor. Section 2.5.4.2 discusses the relationship between site scenarios and receptor activities and behaviors.

Often, licensees will exclude certain phenomena from the central scenario in the performance assessment based on the scenario development process but will include the phenomena in "what if" site scenarios. "What if" site scenarios can include site scenarios that are extremely unlikely and may originate from specific concerns expressed by stakeholders. They may include site scenarios that would normally have been rejected in a scenario development process because of their very low probability of occurrence or low impact on the results. If stakeholders are interested, site scenarios (or implausible conceptual models) that are usually excluded by a licensee during the scenario development process could still be included in the performance assessment analyses. However, the interpretation of the results may be difficult since it is not clear which results are realistic and which ones are very unlikely. Therefore, it is generally recommended that licensees avoid "what if" type site scenarios and use the results of the scenario development process. If "what-if" site scenarios are used, a qualitative or quantitative likelihood of the site scenario should be developed to provide context for the results.

2.5.1 Scope of Analysis

A performance assessment does not need to incorporate all FEPs for a disposal site. The performance assessment should include those FEPs that can either individually or in combination impact the disposal facility's ability to meet the performance objectives. Because the significance of FEPs to performance may be difficult to determine a priori, FEP screening and performance assessment model development are usually iterative processes.

Some events and processes may be interrelated. For example, a large rainfall event may cause erosion, as well as damage to a protective layer that then allows more erosion in the future. A process may be driven by events of variable frequencies and duration. Strict classification of phenomena into events or processes is not as important as ensuring that licensees include the combinations of events and processes that may significantly impact the performance assessment.

Formal or informal approaches may be used to define and screen the FEPs relevant to a particular disposal facility. A licensee using internal (i.e., in-house) subject matter experts to define the scope of the assessment is an example of an informal approach and is appropriate for simple sites and short-lived waste inventories. A formal approach, as discussed in detail in the following sections, may use lists of FEPs developed by others (NEA, 2002) with independent technical review. A formal process is appropriate for complex sites and long-lived waste inventories. Example 2.4 provides some guidance on determining whether a site is simple or complex. The purpose of defining and screening FEPs is to ensure completeness of the evaluation. Whether a formal or informal approach is used to define and screen FEPs, the assessment should provide the following information:

- a clear description of the FEPs included in the assessment
- a description of the FEPs that have been excluded from the assessment and the bases for their exclusion
- a description of the process for determining the significance of FEPs and a consideration of combinations of phenomena for FEPs excluded on the basis of significance

A licensee may use an iterative process for FEP identification and selection within an informal or formal approach. As the performance assessment model is developed, new information needs may be identified that will allow a licensee to further refine the scope of the performance assessment. Licensees may exclude FEPs from the assessment based on regulations or due to lack of relevance (probability) or limited impact (consequence) accounting for the proposed inventory for the disposal site.

Lists of FEPs can be quite extensive (Section 2.5.3.1.1 discusses FEP lists; Appendix C contains a generic FEP list for LLW disposal). Some aggregation may be necessary to make the screening, assessment, and implementation process manageable. Lists that are too general will not be useful, as key FEPs may not be included at the implementation stage or may not be included with appropriate responses and functional behavior because of the coarseness of their definition. The licensee should consider whether inclusion of the FEP at a more refined level of detail would improve the assessment of system performance. For example, if a relatively large or broadly defined FEP with many components has a high likelihood of not being significant to performance, inclusion of FEPs at a more refined level of detail would not improve the assessment of system performance. On the other hand, if a more detailed FEP is being relied on for performance (e.g., a geomembrane within an engineered surface cover), inclusion of such an FEP would be important for developing a more accurate conceptual model of the disposal system. In either case, a licensee should demonstrate that the FEPs included in the analysis are sufficiently comprehensive. Because LLW disposal has been performed at facilities throughout the world, the licensing and operational experiences of other disposal facilities can provide a good starting point for the assessment of a new facility.

FEP screening and scenario development are increasingly used to build confidence that the scope of a performance assessment is appropriate. The more complete the licensee's analysis, the more likely they will be able to gain the confidence of reviewers and stakeholders. The developed site scenarios can be used to focus stakeholder attention on key technical issues. Site scenarios provide an important framework for communication among various stakeholders and an opportunity to discuss and reach a consensus on topics of importance. Licensees can discuss their use of site scenarios and provide an accessible means for public involvement.

Example 2.4: Is my site simple or complex?

Simple sites are generally characterized by few disruptive processes, limited fast transport pathways, relatively homogeneous geology, high stability, and stable climatic conditions. Complex sites have higher uncertainty, driven by more disruptive processes (individually and with cumulative effects); complex geology including fast transport pathways such as fractures; decreased stability; and more highly variable climatic conditions. When there are more processes that can lead to significant releases, there will likely be greater complexity in the performance assessment of the site.

The interpretation of site complexity will also be influenced by the type of waste disposed. If the volumes or concentrations of waste are limited, relatively simple conservative analyses may be appropriate even for a complex site. Disposal of significant quantities of long-lived waste decreases the confidence that stability can be ensured and broadens the range of climatic conditions that could be significant within the assessment period because of the consideration of longer timeframes. In addition, the longer timeframes mean that unlikely disruptive events will be more likely to occur within the period of assessment.

One of the purposes of FEP screening and scenario development for a radioactive waste disposal system is to use scientifically informed expert judgment to guide the development of descriptions of the disposal system and its future behavior. Scenario uncertainty can be assessed by describing and modeling alternative site scenarios of the system; however, it is not an attempt to predict the future. The aim of FEP screening and scenario development is to investigate the importance of scenario uncertainty and provide meaningful illustrations of future conditions to assist in the decision-making process (NEA, 2001a).

2.5.2 Role of Qualified Specialists

The technical basis for FEP screening and the approach to site scenario formation depend on the judgment of the individuals performing the study. The completeness of an FEP analysis can be enhanced by including a broad range of people and diverse sources of information. If appropriate expertise is not available internally, a licensee may need to seek input from external experts. Licensees should obtain information from different sources, a variety of methods, and from all relevant disciplines. Licensees should document decisions based on expert judgment. The analysts performing the FEP screening should have appropriate qualifications to perform the work.

2.5.3 Approaches to Scenario Development

Licensees can use several methods to identify FEPs, screen FEPs, and construct site scenarios. A licensee should ensure that the process is comprehensive, logical, traceable, and transparent. Different approaches to scenario development include bottom-up, top-down, and a mixture of the two.

In bottom-up site scenario formation, the screened FEPs are combined to form a limited number of site scenarios for consequence analysis. Sandia National Laboratories developed a structured approach to site scenario selection (NRC, 1993a). This approach was initially applied to disposal of high-level radioactive waste (HLW) but was later expanded to disposal sites for other radiological *source terms*. Licensees should develop a comprehensive list of FEPs as a starting point. Development of a comprehensive list typically involves the use of generic FEP lists and the identification of other site-specific FEPs. Section 2.5.3.1.1 provides further details. This is followed by a screening process to exclude certain FEPs from further consideration. The retained FEPs are combined into site scenarios for evaluation. FEP screening criteria may include prohibition by regulation, low probability, or limited consequence. Section 2.5.3.1.2 provides further details. The site scenario in which disruptive events do not occur is usually identified as the central site scenario and represents a continuation of the estimated present-day undisturbed conditions. However, a licensee should not interpret "undisturbed" as "static." In some cases, the central scenario could include disruptive processes if they are expected in the normal future evolution of the site. For example, rip-rap on an engineered surface cover may continually be degraded at a slow rate by disruptive processes caused by environmental exposure; however, the design of the disposal facility may have taken considered this, and it is part of the central scenario.

In the top-down approach (see Section 2.5.3.2), licensees develop site scenarios based on analyses of how the safety functions of the disposal system may be affected by possible events and processes. First, the licensee should identify the safety functions of the waste disposal system and then consider the combination of conditions that could affect one or more of the safety functions. A top-down scenario development approach was used in the assessment of HLW disposal, and it consisted of iterative steps as described in detail by the Nuclear Energy Agency (NEA,1992). While NEA (1992) may serve as an example, less effort is appropriate for LLW disposal.

Scenarios derived from a top-down approach typically bound uncertainties potentially affecting the safety functions (e.g., barrier performance). However, to ensure completeness of the processes and events used to establish site scenarios, a licensee may need to take advantage of systematic and comprehensive databases of FEPs. Appendix C provides generic FEP lists applicable for near-surface disposal of LLW. Information typically used in a bottom-up approach may be useful for a licensee to consider in a top-down approach.

Phenomena and features that could significantly influence the performance of the disposal system should be included in the scope of the assessment. A licensee should show that potentially significant transport pathways and plausible alternative site scenarios of the disposal system have been considered. Licensees should consider events that could occur repetitively during the assessment timeframe (e.g., droughts, floods, and earthquakes). They should consider the performance of the disposal system under both present and future conditions.

A central scenario is considered a starting point for each methodology. For FEPs relevant to, or present within, the land disposal facility, but not represented in the central scenario, the licensee should demonstrate that these FEPs are included when developing alternative site scenarios. Alternative site scenarios are generally less likely than the central scenario. Alternative site scenarios may be used to assess different types of disturbances to the normal evolution of the disposal system or to represent different amounts of degradation of the safety functions that were included in the central scenario.

A licensee should explain and justify which site scenarios are regarded as representing the expected evolution of the system (i.e., the central scenario), and which site scenarios address FEPs having a less likely, yet plausible, probability of occurrence (i.e., alternative site scenarios). The range of future environmental conditions at the site and the range of potentially exposed groups should be identified. Licensees and regulators usually assume that humans will be present and that they will make use of local resources. It is appropriate for licensees to assume that humans in the future will have habits similar to those of present humans, except where this is clearly inconsistent with the assumed variations in climatic conditions at a site. Section 2.5.4.2 provides additional guidance to licensees on the development of receptor behaviors and activities.

If disruptive events are represented in a probabilistic performance assessment model, a licensee should conservatively assign the probability of an event occurring if the frequency of the event is not known. Formal expert elicitation may be necessary to define event frequencies. Technical bases should be provided for the probability of disruptive events occurring. Alternatively, the licensee may represent a range of alternative site scenarios to capture the potential effects of disruptive events, without weighting the potential disruptive events with probabilities. Section 2.2.4.1 provides additional information on expert elicitation.

2.5.3.1 Bottom-Up Approach

2.5.3.1.1 Identification and Categorization of FEPs

Many FEP lists, catalogs, and databases have been developed in different countries and encompass a range of radioactive waste types, disposal system designs, and disposal environments. The size of these FEP lists varies, as do the content and level of detail of entries. The NEA (2002) has compiled internationally published FEP lists, catalogs, and databases. Licensees should perform a literature search when developing their site scenario analyses because FEP lists continually change as scientific information is developed. Licensees may want to consider the generic FEP list for LLW disposal that the NRC staff developed in Appendix C to this document.

2.5.3.1.1.1 Identification and Categorization of Present FEPs

To identify FEPs important to LLW disposal, licensees should review existing FEP lists. Typically, FEPs are first identified and then they are categorized. The first step in the identification process consists of reviewing the information provided by the assessment context, as it relates to the performance assessment of the site. For example, legislation or regulatory guidance may specify FEPs that licensees should consider (e.g., future human activity, the biosphere system, or future climate states). Licensees could use a previously compiled generic FEP list as a starting point to save time and effort. They will need access to information and documents pertaining to the characterization and description of the site being considered.

The identification process may be iterative. Initially, the licensee may identify the FEPs assuming the site has static conditions. Often a central scenario can be constructed from present FEPs if the proposed disposal site is in an area of geological and geomorphological stability. Next, dynamic conditions such as natural degradation processes are assumed to affect the natural and engineered barriers.

The following steps describe this approach:

(1) Use a generic list to identify features and processes that are currently present at the site. For this step, features or processes likely to be there in the future or that existed only in the past should be temporarily set aside until the next phase, as discussed in Section 2.5.3.1.1.2.

(2) Eliminate features and processes in Step 1 that have no potential to affect performance.

(3) Add features and processes that have occurred relatively recently (i.e., within 100 years) at the site, not events that have occurred in the long distant past (e.g., glaciers) or that may occur in the long-term future (these will be identified later as discussed in Section 2.5.3.1.1.2).

The identification step is followed by, and closely linked to, the categorization step, whereby FEPs with certain similar properties are grouped together. Categorization provides a framework for organizing FEPs for incorporation into site scenarios and the subsequent assessment. In addition, it provides a brief overview of relevant factors comprising the disposal system. Categorizing the FEPs of a land disposal facility can help a reviewer identify FEPs that may have been overlooked and is one of the objectives of placing FEPs in logical categories (NEA, 1992). It is usually better to initially categorize a range of broadly defined FEPs first to ensure the comprehensiveness of the FEP analysis (i.e., comprehensiveness of the FEPs considered and included in the assessment). Broadly defined FEPs encompass more detailed FEPs. For example, a broad FEP such as an engineered surface cover may include many individual FEPs such as a radon barrier, a drainage layer, a geomembrane, etc. The broad FEPs should then be replaced with the detailed FEPs comprising the broadly defined FEP unless there is little uncertainty that the broad FEP and its detailed components do not influence performance significantly (e.g., a vertical wall of a disposal cell has multiple components but there is little uncertainty that the vertical wall has no significance to performance).

Examples of categorization approaches include the following:

- grouping FEPs related to natural, human, or waste into separate categories
- placing FEPs into categories according to the time scale during which different events and processes occur
- organizing FEPs related to near-field, far-field, or biosphere
- grouping FEPs according to combinations of the magnitudes of probability and consequence
- classifying FEPs according to scientific discipline

2.5.3.1.1.2 Identification and Categorization of Future FEPs

In the previous section, the NRC staff suggests that a licensee could develop the initial FEP list by assuming the proposed disposal site is not intrinsically dynamic. However, the staff recognizes that most proposed disposal areas are dynamic. For some sites, a few hundred years may be sufficient to result in significant changes to the disposal site, which would make use of a present-day FEP list incomplete. Changes can occur over different temporal and spatial scales.

The NRC staff suggests that licensees continue the identification process after the completion of the steps listed in the previous section. Features and processes should be identified that may occur during the analysis timeframe. Processes considered should include degradation, deterioration, and alteration processes that are expected to occur during the period of interest. These processes may include chemical and physical degradation (e.g., fracturing or cracking) of a cementitious wasteform, corrosion of rebar, chemical and physical deterioration of a high-density polyethylene component of an engineered surface barrier, alteration of the soils on top of an engineered surface barrier, or changes in the groundwater bearing media (e.g., dissolution of limestone causing decreasing travel times). The analyst or *qualified specialist* should be aware that this step is for the purpose of identifying potential FEPs; screening of FEPs occurs in the subsequent step.

A licensee can identify FEPs using qualitative methods, such as literature searches by qualified specialists. Examination of past local natural history can identify processes that may have been relevant at a site in the past and could be an indicator of potential future behavior. Licensees can use the history of the disposal site as a basis to identify an FEP for further consideration. Some of the FEPs from the present will persist in the future; however, some processes may stop (e.g., groundwater recharge due to newly formed permafrost), and other features may appear (e.g., a forest may develop on land that was previously scrubland). For example, a present-day desert environment may have had higher average annual precipitation rates in the past with a higher density of rivers and lakes and different distributions of fauna and flora (e.g., Lake Manly covering present-day Death Valley). Another example could include a site adjacent to a river with numerous dams. Future removal of those dams may change the natural characteristics of the site (e.g., water table levels). If the past environment is considered plausible in the future, the FEPs associated with this past environment should be included.

More low-frequency events usually are included in the FEPs list for a long-term performance period than may be included in the FEP list for a 10,000-year compliance period (e.g., an event with a 10^{-6} per year frequency would be unlikely in a 10,000-year period but likely in a 1 million-year performance period). In addition, if the same set of FEPs is appropriate for both analyses (performance and compliance periods), they may need to be represented differently in each analysis due to the longer timeframe and potentially altered conditions. Section 6.2 provides additional guidance on FEPs for performance period analyses.

2.5.3.1.2 Screening of FEPs

A licensee should screen identified FEPs to determine the subset of FEPs important to disposal system performance and document the basis for excluding or including an FEP for further consideration. A licensee should use clear and justifiable criteria. Screening approaches can vary from using some type of “importance” ranking scale (e.g., 0 through 10) or simply an “include” or “exclude” system. Exhaustive analysis of importance is not necessary for FEP screening for LLW disposal. Simple calculations or bounding estimates can assist with the selection. Licensees should document the assumptions, data, or empirical information used to make these determinations. They should perform the screening process on a site-specific basis and evaluate FEPs or FEP categories one at a time against a screening criterion. During the screening process, interactions between FEPs should be considered. If there are uncertainties as to whether a FEP can be screened, then it should be retained. The FEP can be reevaluated at a later stage in the screening and evaluation process.

The following subsections discuss various types of criteria for screening of FEPs. The regulatory criteria are considered first since they take precedence over other screening considerations.

2.5.3.1.2.1 Regulatory

FEPs can be eliminated from further consideration based on inconsistency with applicable regulations. The NRC approach to analyzing timeframes is based on a compliance period of 1,000 years or 10,000 years following closure of the disposal site, and a longer-term performance period during which a licensee must demonstrate that effort has been made to reduce releases to the extent reasonably achievable. The performance assessment should reflect changes in FEPs of the natural environment such as climatology, geology, and geomorphology. For FEP analyses demonstrating a long-term geomorphic and geologically stable site, the scope of the FEPs considered may not need to be expanded.

The regulation in 10 CFR 61.50 is important as it affects the FEP screening process by providing the disposal site suitability requirements for the land disposal of LLW. The process for determining whether some of these criteria will be met is complementary to the FEP process. The criteria from 10 CFR 61.50 associated with the process of FEP analysis are 10 CFR 61.50(a)(2)(i-iv), 10 CFR 61.50(a)(3), and 10 CFR 61.50(a)(5)(i-iv). Appendix B includes hazard maps related to the features and phenomena of these criteria. The hazard maps provide a coarse estimate of impacted areas. Licensees that meet the criteria in 10 CFR 61.1(b) need not comply with 10 CFR 61.50(a) and (b), but must instead comply with 10 CFR 61.50(c), which preserve the original 10 CFR 61.50 regulations.

Site suitability requirements are treated differently in 10 CFR Part 61 depending on the type of characteristic and timeframe. Historically, hydrological FEPs were the key drivers of poor performance of early (before 10 CFR Part 61) LLW disposal facilities. Therefore, hydrological characteristics of the site are treated differently from other site characteristics. Regardless of the type of waste disposed, the hydrological site characteristics are either required to be present for disposal for a 500-year timeframe (e.g., the site must be generally well-drained) or should not be present for a 500-year timeframe because they will adversely affect the performance of the disposal site (e.g., waste may not be disposed of in the zone of water table fluctuation).

For the ensuing period beginning 500 years after closure of the disposal site, the licensee may, in accordance with 10 CFR 61.50(a)(3), demonstrate that the hydrological site characteristics will not exist or that 10 CFR 61.41 and 10 CFR 61.42 performance objectives can be met irrespective of the condition. Disposal systems with water-related challenges in the present day and foreseeable future are generally not amenable to stability and defensible modeling and assessment.

The licensee should review FEPs to identify those associated with natural resources (i.e., for natural material currently considered to be a resource and with a distribution that is currently known). If review of the FEPs determines that natural resources of the type described in 10 CFR 61.50(a)(5)(i) exist near a proposed disposal site and are likely to be exploited resulting in failure to meet the performance objectives of 10 CFR Part 61, Subpart C, then the site is not acceptable for LLW disposal.

> **10 CFR 61.50(a)(5)(i)**: *The land disposal facility must not be located in, or in the immediate vicinity of, an area having known natural resources which, if exploited, would result in the failure of the disposal site to meet the performance objectives of subpart C of this part.*

The NRC staff recommends that a licensee survey and evaluate an area for potentially exploitable natural resources within a radius of 5 kilometers from the future boundary of a land disposal facility.

> **10 CFR 61.50(a)(2)(i)**: [For at least 500 years following closure of the land disposal facility, the disposal site cannot have any of the following characteristics:] *Poor drainage, flooding or frequent ponding, located in a 100-year flood plain, as defined in Executive Order 11988, "Floodplain Management," or located in a wetland, as defined in Executive Order 11990, "Protection of Wetlands."*

A licensee must demonstrate that the disposal site is not located in a projected 100-year floodplain or a permanent or periodic wetland for a period lasting 500 years after closure. Section 2.4.1, Appendix A, in NUREG-1200 provides additional guidance. There are two main geomorphic zones to consider: (i) coastal areas, and (ii) flood plains and wetlands.

Coastal high-hazard areas are described as having a high hazard status due to various factors including proximity to the coastline and relative elevation of the site. Both the present and future should be considered for this criterion. Two aspects are important for the present and the future: vulnerability to erosion and vulnerability to flooding.

Possible FEPs that could cause detrimental erosion rates include increased rainfall or an increased gradient (and therefore, an increase in the erosional power of any runoff) due to changes in future climate states. As for the increased precipitation, the magnitude of the probable maximum precipitation (PMP)/probable maximum flood (PMF) is not likely to change significantly in a wetter climate, based on the conservatism associated with the estimation of the PMP and the proper computation of the PMF.

A drop in sea level could change the erosion rates of certain coastal areas because of an increase in the length of slopes and topographic gradients. Various organizations have documented sea level variations and there is a general consensus that sea levels have dropped to 110 to 120 m (340 to 400 ft) below current sea level during the glacial periods of the current Quaternary ice age (NOAA, 2008; Gornitz, 2007; IPCC, 2001). If a proposed disposal facility is associated with a coastal area, the licensee should perform an erosion analysis assuming a sea level drop unless the FEP analysis determines that such a phenomenon is implausible. The drop magnitude may be determined by a licensee with an adequate technical basis for the magnitude of the hypothetical sea level drop. Alternately, the consequences of sea level rise need to be analyzed if the proximity to the coastline and relative elevation of the site allow the proposed disposal facility's performance to be significantly affected.

FEPs that could cause detrimental flooding and may exist in future climates include increased high tides due to higher sea levels and potentially higher storm surges. Partial deglaciation of the Greenland ice sheet and the West Antarctic ice sheet cannot be excluded within a timeframe of 1 million years and would contribute a 4 to 6 m (13 to 20 ft) or more sea level rise during the performance period. For performance period analyses on the order of 1 million

years, high-tide levels near a proposed disposal site could be estimated assuming a sea level increase of 5 m (16 ft). A licensee may provide a technical basis for the variation in sea level used in the analysis for performance period timeframes longer than 10,000 years but shorter than 1 million years.

Even on a qualitative level, the probability of future tsunami hazards for a particular location would be too difficult to estimate over the performance period. The level of effort required to evaluate future tsunami hazards is likely to be excessive, and technical analyses are not warranted considering the short duration that a potential disposal site would be flooded by a tsunami and that engineered barrier structures would be underground.

Flooding may impact the performance of a disposal system. Important aspects include (1) the location of the current 100-year flood plains and wetlands in relation to the potential disposal site, and (2) how the current boundaries of the flood plains could change over the compliance or performance period.

Flooding can directly influence the performance of the disposal system or influence another process, such as erosion, that can impact the performance of the disposal system. Higher topographic areas generally have fewer floodplains and wetlands and the effects of erosion usually become more problematic. On the other hand, depositional areas and relatively flat areas would have the potential of becoming flood plains and wetlands.

A licensee can use knowledge of the past surface hydrology in the potential disposal area to strengthen the qualitative FEP assessment. For example, the surface hydrology of southern Louisiana is quite diverse. The Mississippi River and other rivers have changed river channel beds numerous times and 100-year floodplains would change in concert with river course evolution. Disposal sites for significant quantities of long-lived waste should not be in the vicinity of these channels.

Features and processes identified in studies of the natural history of the area often provide an indication of the processes that will likely be present in the future. If, for example, an area has been frequently flooded during and immediately after past glacial periods to create intermittent glacial lakes (e.g., Lake Missoula, Lake Lewis, and Lake Bonneville), the probability is high that similar processes and events will occur during the performance period.

Geomorphologic evidence may provide useful data on large paleofloods. The evidence can include slack-water deposits, scour lines, high-water marks, and undisturbed areas (Stedinger and Cohen, 1986). The advantage of paleoflood data is that it can improve a probabilistic flood hazard assessment and extend short or limited flood records to include the last 1,000 to 10,000 years (Klinger and England, 2013). Paleoflood data can improve the estimates of the magnitude and frequency of large floods. Licensees should develop an assessment of the potential for future flooding and for local floodplain and wetland formation in the future.

In addition, paleoflood data may provide evidence of large, dynamic floods, similar to the “outburst” floods that occur when the water flow of dammed glacial lakes is suddenly unobstructed. Massive floods could destroy a disposal site located close to the surface and therefore the licensee would be unable to demonstrate that the site stability performance objective (i.e., 10 CFR 61.44) will be met. Potential sites located in areas affected by previous glacial processes or subjected to previous massive flooding would require additional analysis and careful evaluation. Although the site and immediate area would not be habitable for

humans and a potential receptor, the contaminated material may remain in a relatively concentrated form while being deposited downstream near more habitable locations where receptors could exist. Potential disposal sites located in areas subject to previous massive flooding may require deeper disposal and additional man-made barriers (e.g., engineered surface covers) to mitigate the destructive force of large floods.

> **10 CFR 61.50(a)(2)(iii)**: [For at least 500 years following closure of the land disposal facility, the disposal site cannot have any of the following characteristics:] *A water table too close to the land surface. The unsaturated zone must be thick enough so that waste is not disposed within a saturated zone nor a zone of water table fluctuation, perennial or otherwise.*

A potential site located in an area with data from wells and data on groundwater fluctuations should be sufficient for a licensee to perform FEP screening for the depth to water table. An area without sufficient data on the local water table and associated groundwater fluctuations may need additional site characterization. Features of interest for a licensee are current water table elevation and soil types, including sediment layers, with the potential for creating a perched water zone. A licensee should identify the hydrogeologic units of the layers below the disposal site. For example, a caliche layer or dense clay layer may cause temporary perched zones that may exist only on a seasonal basis. It should be sufficient for a licensee to identify such features in combination with meteorological data, topography of the disposal system, and processes of the local hydrogeology in the analysis of current water table behavior.

The requirement at 10 CFR 61.50(a)(2)(iii) is closely related to the requirement at 10 CFR 61.50(a)(2)(i) (i.e., poor drainage and flooding). Future water table elevation is partially a function of the current landscape and potential changes to the landscape in the future. As with the requirement in 10 CFR 61.50(a)(2)(i), topography, hydrogeology, erosion rate, and precipitation rates are determining factors for water table elevation. With updated studies on long-term trends, natural history, and PMP/PMF, it should be possible for a licensee to develop an assessment to bound potential water table changes. Water tables in topographic highs or steep areas may have a greater degree of seasonal variability but generally would be expected to have a smaller potential to significantly change during the long term than would a relatively flat, depositional area that is susceptible to drought and flooding. Depending on the hydrogeology of the region, a low-lying or depositional area today can experience a rising water table or even become a flood plain in a future wetter climate.

Tools and codes are available to licensees to support analyses and assessments of groundwater discharge that can be used to assist in the evaluation of long-term water table fluctuations. Knowledge of the paleoclimatology, paleopedology, and past hydrology of the potential disposal area can support the assessment of this requirement. For example, arid regions with past wetter conditions may have had higher water tables. However, evidence from the past has revealed that features such as wetlands and lakes had previously existed in currently arid areas. For example, Death Valley is generally dry and waterless; however, this valley had been flooded in the past (Lake Manly), and the moisture was persistent enough to support a native population. Death Valley is currently dry because of the extremely low rate of precipitation at and near the area. However, a geomorphologist, or a qualified specialist, would recognize that its topography is ideal for lake formation if more water were to become available (as evidence from its geological past has shown). Other areas may receive a high rate of rainfall, but the topography and geology will not allow for lake formation in the future.

In general, the climatic history during the Quaternary period has been of a cyclic nature consisting of glacial and interglacial stages. Features and other evidence of the past natural history can assist in reconstructing the full cyclic climatic history including precipitation rate and temperature ranges. Potential disposal sites further to the south may never have experienced glacial coverage in recent geologic history. Knowledge and evaluation of the full cyclic extremes and updated studies of topographic trends (e.g., surface uplift, tectonic subsidence, erosion vulnerability, etc.) can help a licensee in developing an assessment that could potentially bound long-term water table fluctuations, even if these fluctuations may have been extreme. Potential disposal sites further to the north may have been previously covered by glaciers that cause disruptive surface geologic processes so that the requirement in 10 CFR 61.50(a)(5)(iii) becomes the primary focus.

> **10 CFR 61.50(a)(2)(iv)**: [For at least 500 years following closure of the land disposal facility, the disposal site cannot have any of the following characteristics:] G*roundwater from below discharging to the land surface within the disposal site at any time.*

The criterion in 10 CFR 61.50(a)(2)(iv) is closely related to the criterion in 10 CFR 61.50(a)(2)(i) and 10 CFR 61.50(a)(2)(iii). Groundwater discharge areas are partially a function of the current landscape and changes to the landscape in the future. As with the requirement in 10 CFR 61.50(a)(2)(i), topography, hydrogeology, erosion rate, and precipitation rates are determining factors if water will discharge in a certain area. Any time a water table rises higher than the ground surface, groundwater will discharge to the surface. The assessment done for 10 CFR 61.50(a)(2)(iii) should assist in determining if the hydrogeologic units used for disposal are able to discharge groundwater to the surface within the disposal site under current or projected future conditions. A licensee should compare the approximate rise and fall of the water table assessed in 10 CFR 61.50(a)(2)(iii) to the changes in topography.

Licensees should consider different combinations of trends: net mass change (deposition/erosion) and net change on water table (rising/falling). For example, it is possible that areas with relatively stable long-term water table levels may also experience relatively high erosion rates; the net effect is a water table advancing to the ground surface. It is unlikely, however, that such sites would fulfill the requirement of 10 CFR 61.50(a)(5)(iii) which states that areas with significant surface geologic processes must be avoided. Site characterization should provide sufficient data on geology, hydrogeology, topography, paleoclimates, and features and processes to qualitatively assess the plausibility of groundwater discharge occurring in the future. Evidence supporting the existence of past groundwater discharge areas, such as calcite deposits or diatomite, increases the likelihood that the disposal area may have groundwater discharge in the future.

> **10 CFR 61.50(a)(5)(ii)**: *The land disposal facility must not be located in, or in the immediate vicinity of, an area where tectonic processes such as faulting, folding, seismic activity, or volcanism may occur with such frequency and extent to significantly affect the ability of the disposal site to meet the performance objectives of Subpart C of this part, or to preclude defensible modeling results due to large uncertainties.*

Tectonic processes, such as faulting, folding, seismic activity, and volcanism, might lead to short-duration disruptive events, but unlike the processes associated with other disposal site suitability requirements, these processes are usually linked with plate tectonics that proceed at

very slow rates. These are long-term processes and unlike the processes related to the previously discussed site suitability requirements, water is not directly involved. The assessment and evaluation carried out for the compliance or performance periods to approximate the frequency and extent of faulting, folding, seismic activity, and volcanism in a particular area, as discussed in NUREG-1200, should not lead to significantly different results when applied to the performance period. Tectonic processes are large-scale processes, and the frequency and extent of seismic and volcanic activity may be relevant over a time range of 100s to 1,000s of years. The licensee should assess 10 CFR 61.50(a)(5)(ii) based on the guidance in NUREG-1200 and NUREG-1199.

> **10 CFR 61.50(a)(5)(iii)**: *The land disposal facility must not be located in, or in the immediate vicinity of, an area where surface geologic processes such as mass wasting, erosion, slumping, landslides, or weathering occur with such frequency and extent to significantly affect the ability of the disposal site to meet the performance objectives of Subpart C of this part, or to preclude defensible modeling* due to large uncertainties.

Broad FEPs that could include more detailed FEPs and cause changes in the frequency and extent of the surface geologic processes are climate, topography, geology, soil type, flora, fauna, water chemistry, and water flow. The more detailed FEPs could include FEPs such as precipitation, temperature, lake effects, wave action, permafrost, soil transport, soil build-up, sandy soil, deep roots, burrowing animals, solvation, bioturbation, and fracture flow. Appendix C contains generic FEP lists applicable to near-surface disposal of LLW. Short-term, large-scale topographic change normally does not occur unless associated with a disruptive event. The frequency and extent of water (i.e., precipitation rate) have the greatest potential for driving rates of mass wasting, erosion, slumping, landslides, or weathering. NUREG-1623, "Design of Erosion Protection for Long-Term Stabilization," issued September 2002 (NRC, 2002b), provides guidance on surface geologic processes. FEPs that could cause unsuitable rates of erosion include increased rainfall and an increased gradient (and therefore, an increase in the erosional power of any runoff). As for the increased precipitation, the magnitude of the PMP/PMF is not likely to change significantly in a wetter climate, based on the conservatism associated with the estimation of the PMP.

A licensee can estimate the rates of surface geologic processes and determine if the FEP should be included in the analyses. The field of geomorphology has evolved significantly over the last half century and many technologies are available today to facilitate studying surface geologic processes, including various computer programs such as hydrologic, erosion, and landscape evolution codes (e.g., CHILD, SIBERIA). Chapter 5.0 provides more detail on the technical assessment of erosion. Landscape evolution is especially important with respect to potential changes to nearby streams and river channels in unconsolidated material. Given sufficient precipitation, large rain events, and time, the watershed of an area can change considerably. Specific parts of a facility may end up close to stream channels and with an increased gradient. Drainage patterns may change if variations in climate circulation patterns are great enough. Erosion and deposition rates may vary spatially and temporally across the site. Long-term erosion processes often produce gullies, which do not uniformly erode over their entire length or width. Peak erosion depth may translate into a total breach of a portion of the disposal facility.

The massive ice covers of the glacial periods were a source of extensive, large-scale surface geologic processes. Appendix B, Figure B-9, shows the approximate area covered by glaciers

during the last three glacial periods of the current Quaternary ice age (i.e., the Wisconsin, Illinoian, and Pre- Illinoian glacial periods). Glaciers can cause very disruptive surface geologic processes, and potential sites located in areas affected by previous glacial processes could require additional analysis and evaluation.

Evidence of the natural history of the area, or from an appropriate natural analog, should support licensee decisions to exclude or retain the FEPs associated with the surface geologic processes. Pedogenic processes, biotic activities, and bioturbation are processes that may impede or accelerate surface geologic processes. Thick root systems from certain plants are known to greatly reduce the erosional force; however, it is difficult to rely on the continuous presence of a specific plant that might be needed for longer timespans. A number of phenomena could influence flora, such as drought, fire, disease, fungi, and insects.

2.5.3.1.2.2 Probability

Licensees use screening criteria, based on the probability of occurrence, consequences to the performance of the disposal system, or both, to eliminate FEPs that are implausible or that have relatively minor consequences. Three methods that a licensee may consider for probability-based screening include the following (EC, 2009a):

- Quantitative methods—FEPs are represented numerically, and event probability is an explicit part of the performance assessment calculation, such as those methods employed in the probabilistic models used for the Yucca Mountain and Waste Isolation Pilot Plant projects.[6] The probability classification in its entirety would include the following and could be used in conjunction with the qualitative approach:
  - Implausible: Very Unlikely ($<10^{-8}$/yr)
  - Plausible: Unlikely or Less Likely ($<10^{-5}$/yr and $>10^{-8}$/yr)
  - Plausible: Reasonably Foreseeable ($>10^{-5}$/yr or >1 chance in 100,000 per year or greater than a 1 in 10 chance of occurring over a 10,000-year period)
- Qualitative methods—The probability or likelihood of occurrence of FEPs is described qualitatively or semi-quantitatively, and probability values are not an explicit part of the numerical modeling. However, a qualitative description of probability could still be used (e.g., unlikely versus very unlikely). Qualified specialists would need to determine the probability, as well as the terminology to be used to describe that probability. For example, experts on paleofloods may determine that a site's physical environment and topography preclude major flooding. The low probability from this qualitative determination could then be labeled as either "very unlikely" or "implausible" or some other term depending on the terminology agreed on. The level of effort for this method is appropriate for LLW disposal sites and is recommended for the analysis of FEPs.

[6] As an example, a performance assessment requirement in 10 CFR Part 63 prohibits consideration of very unlikely FEPs (i.e., FEPs that are estimated to have less than 1 chance in 10,000 of occurring within 10,000 years). As a result, FEPs with probabilities lower than 10^{-8}/year (yr) should be screened out during the scenario development process.

- Non-consideration of probability—Few or no relevant data are available and there are large uncertainties associated with describing the site scenario. FEPs are included because there is insufficient information to exclude them.

One technique applicable to FEP screening based on probability is the frequentist technique (EC, 2009b), in which probabilities are based on observations of how often a *phenomenon* has occurred at the proposed site or at a natural analog to the site. Constraints on using this technique include limited data or non-representative data.

A main consideration in the assignment of probabilities to scenario-forming FEPs is uncertainty. Assignment of probabilities relies on the skills and experiences of the analysts and on the qualifications of the independent reviewers. Most probability estimates developed by licensees will include considerable judgment. Because the FEP screening process can result in FEPs not being further considered, licensees should make conservative decisions when screening based on probability. Probability screening will involve the use of existing data in areas like paleoclimatology, plate tectonics, hydrology, geology, and natural resources combined with expert judgment. It is important to document probability estimates.

When sufficient information is available, an analyst can have confidence in probability estimates and understand the uncertainty in the probability estimates. If the sampled population on which the probability is based is small or the quality of the data is poor, or if the estimates are based on assumptions, then the uncertainty associated with probability can be high. As previously discussed, long-term analyses or estimates may be more difficult to quantify because of a scarcity of reliable data. For example, the probability of occurrence for an earthquake above a certain magnitude over a 100-year period would be uncertain. However, the confidence in that probability estimate may be relatively high when compared to the uncertainty in the probability of tundra-like conditions occurring at a particular location many thousands of years in the future. When using probability to screen FEPs, the uncertainty in the probability should be considered.

2.5.3.1.2.3 Consequence

Many of the same characteristics and difficulties associated with quantifying probabilities also apply to attempts to quantify consequences or impacts on disposal site performance. A conservative semi-quantitative approach is generally better suited than a quantitative approach when using consequence as a screening criterion. For example, a bounding consequence calculation could be, under the appropriate conditions, a helpful calculation to indicate that the associated impact will be insignificant. Consequence screening may be difficult to complete with simple analysis if the problem is complex. Shelf- and cliff-type responses, as well as local minima and maxima, can confound interpretation of bounding consequence or other types of significance determinations. Example 2.5 describes a cliff-type response and the way it may affect FEP screening.

A performance assessment model will commonly have inter- and intra-dependent components or submodels. For this reason, the initial development process should err on the side of including FEPs of unclear significance. Once the model is developed and the connection and communication of submodels have been established, then the significance determination process can more reliably eliminate FEPs.

The inclusion or exclusion of an FEP in a performance assessment model depends on whether it has a measurable, observable, or significant effect on disposal system performance. Since

FEPs are not measurable or observable in the far future, expert judgment will constitute a key element of the screening process. Experts or specialists would need to determine the magnitude of the consequence, as well as the terminology to be used to qualitatively describe that impact (e.g., significant, major, substantial). For example, if it is plausible that the consequence of an FEP can be expected to change dose results that are relatively close to the performance objective, then the impact of the feature or phenomenon in question is significant. In addition, if the output results change by orders of magnitude depending on the absence or presence of a feature, or the occurrence or absence of a phenomenon, then the impact of the feature or phenomenon is significant. Previous performance assessments or additional modeling or sensitivity analyses may provide insights with respect to the significance of the consequences of FEPs. Since subsystem-level effects on system-level performance may be masked by certain designs or combinations of input parameter values, the quality of the FEP analysis relies on the qualifications and judgment of the analysts. In addition, the licensee using consequence-based FEP screening should consider the interrelationships of the FEPs with one another, and the effect that different combinations of FEPs may have on the consequence (e.g., if an engineered surface barrier cover is performing as planned, the performance of other engineered features may be masked for a specific period of time).

> **Example 2.5:** A good example of a cliff-type response is the transport of a short-lived, sorbing radionuclide. At high values of the distribution coefficient, the radionuclide decays in place. However, at low values of the distribution coefficient, the radionuclide may be transported to a potential receptor location. A measure of the central tendency of the distribution coefficient distribution may inappropriately show that the radionuclide poses little risk even though a small change to the sorption coefficient would allow the radionuclide to arrive at the receptor location during the compliance period and potentially cause a significant consequence.

When sufficient and reliable information is available, an analyst can have confidence in consequence estimates, and the uncertainty may be low. If there is limited information, the quality of the data is poor, or if the estimates are based on assumptions, then the uncertainty associated with the consequence may be high. As previously discussed, the interrelationships of the FEPs with one another and the effect that different combinations of FEPs may have on the consequence should be considered to the extent possible. For example, variable temperature and chemical composition of groundwater may affect the rate of radionuclide transport. *Distribution coefficient* (K_d) values for radionuclides may also change because of changing environments. However, these different FEPs (inherent groundwater variability versus temporal variability in the environment) may affect the K_d values for individual radionuclides in opposite ways. In other words, it may be challenging to determine the consequence of a single FEP when many FEPs have an uncertain influence on the results. High uncertainty in the consequence of an FEP should result in the FEP being included, unless even with the uncertainty, the FEP can be shown to be unlikely to be significant or the timing of occurrence will be delayed outside of the regulatory analysis timeframes.

2.5.3.1.2.4 Screening Techniques

Although there may be different ways to organize, evaluate, and present the FEPs, most methods have certain similarities and the lists of retained FEPs obtained by different methods should be similar. Techniques are not mutually exclusive, and licensees may use several tools

in combination. The following section outlines one screening technique, but there may be other valid techniques that a licensee may use.

Table 2-11 Example of Screening Criteria Based on Qualitative Probabilities, Consequences, and Uncertainty Associated with an FEP

Case	Probability Below a Qualitative Screening Limit (e.g., "very unlikely")	Uncertainty Associated with Probability	Consequence Below a Qualitative Screening Limit (e.g., "not significant")	Uncertainty Associated with Consequence		Screening Outcome
1	yes	low	yes	low		Out
2	yes	high	yes	low		Out
3	yes	low	yes	high		Out
4	yes	high	yes	high		In
5	yes	low	no	low		Out
6	yes	high	no	low		In
7	yes	low	no	high		Out
8	yes	high	no	high		In
9	no	low	yes	low		Out
10	no	high	yes	low		Out
11	no	low	yes	high		In
12	no	high	yes	high		In
13	no	low	no	low		In
14	no	high	no	low		In
15	no	low	no	high		In
16	no	high	no	high		In

FEP screening by licensees will involve the use of professional judgment. Screening criteria matrices categorize phenomena into various elements according to the magnitude of the probability and the consequence (Hommel, 2012; NEA, 1992). For example, Table 2-1 illustrates the approach of using a combination of probability and consequence of an FEP, in conjunction with the uncertainty associated with that FEP, to retain or screen out the FEP (IAEA, 2004). Screening criteria matrices require specialists and experts to select the FEPs they estimate to be important and include in the assessment. Screening criteria matrices are one method that can provide transparency. Table 2-1 presents an example matrix that licensees can use to perform FEP screening for a LLW disposal facility performance assessment. Factors used for initial screening purposes include (1) probability of the FEP; (2) consequence of the FEP, and (3) the uncertainties associated with the probability and consequence.

Depending on the probability, a phenomenon or feature can be placed either below or above a screening value as seen in column 2 of Table 2-1. FEPs can be screened in (included) or screened out (excluded) if a qualitative screening limit is selected and applied. For example, analysts may decide that FEPs with "very unlikely" probabilities should be screened out, while FEPs with an "unlikely" probability should be included in the scenario development. The third column divides the uncertainty associated with the probability into high and low uncertainty. Consequence is handled in a manner similar to that used for probability, which can be seen in columns 4 and 5 in Table 2-1. Analysts can assign a qualitative consequence screening limit in addition to an uncertainty estimate associated with the consequence. After being evaluated, an

FEP will have assigned designations for the probability, the consequence, and their uncertainties. Based on these designations, the qualified specialists will screen FEPs in or out, as shown in the far right-hand column.

Whether a FEP is screened in or out is apparent in most of the 16 cases in Table 2-1. For example, in case 1, an FEP is clearly screened out since both probability and consequence are below a screening limit and their associated uncertainties are small. Conversely, FEPs in cases 13-16 are retained for scenario development despite the various degrees of associated uncertainty since probability and consequence are both above their screening limits. Some of the cases are not as straightforward and are discussed in more detail below.

For cases 2 through 4, cases 2 and 3 have probability and consequence below predetermined, qualitative screening limits. For case 2, the uncertainty associated with consequence is low, and for case 3, the uncertainty associated with probability is low, giving the analyst sufficient confidence to exclude the FEP from further consideration. For case 4, there is less confidence since both probability and consequence have a high uncertainty, and although probability and consequence are below screening limits, the uncertainties associated with both probability and consequence would lead the analyst to keep the FEP for scenario development during the compliance period.

Cases 5 through 8 all have consequences above the screening limits, indicating potential inclusion of these FEPs for further consideration; however, the uncertainty associated with the probability is important to determining their final inclusion or retention. If there is low uncertainty associated with a probability lower than a screening limit, there is confidence that despite the higher impact or consequence, the probability of occurrence is sufficiently low that the FEP can be screened out (cases 5 and 7). However, if there is high uncertainty associated with the small probability, as in cases 6 and 8, the FEP should be screened into the analysis.

Cases 9 through 12 are the converse of cases 5 through 8; all cases have probabilities above the screening limits indicating potential inclusion of these FEPs for further consideration. However, if there is low uncertainty associated with a consequence lower than a screening limit, there is confidence that despite the higher probability of the FEP occurring or being present, the impact of the FEP is sufficiently insignificant that the FEP would be screened out (cases 9 and 10). However, if there is high uncertainty associated with the small consequence, as in cases 11 and 12, the FEP should be screened into the analysis.

Increased uncertainties associated with the phenomena and features of a 10,000-year compliance period compared to a 1,000-year compliance period would be reflected in the potentially increasing number of FEPs and the uncertainties associated with probabilities and consequences of the FEPs. A licensee may extend 1,000-year compliance period calculations into the future without modification, provided that the calculations are complete with respect to including key FEPs relevant to the longer timeframe or excluding those FEPs relevant only to a 1,000-year compliance period. However, for time periods considerably longer than the compliance period, FEPs that have been excluded from further consideration in a 1,000-year compliance period may not be able to be screened out from further consideration for 10,000-year compliance period analyses. Potentially significant FEPs may need to be considered during scenario development if disruptive processes are expected to start occurring after 1,000 years or if the cumulative impact from repetitive events over longer timeframes is not included in the shorter timeframe and the repetition of those processes and events could lead to significant impacts.

2.5.3.2 Top-Down Approach

The concept of safety functions has been used with increased frequency as a scenario development methodology (EC, 2009b). An advantage of the top-down approach is that the licensee can focus on the capabilities of the significant barriers and consider behavior of individual features in the context of overall system performance relative to the decision to be made. Some analysts consider the top-down approach a simpler way to develop site scenarios since safety functions are more quickly identified than significant interrelationships between FEPs and since the probabilities of the resultant site scenarios may be easier to estimate. The discussion below briefly describes this approach. Licensees may refer to the sources given in the reference section (Chapter 11.0) for additional information. Both the top-down approach and bottom-up approach are often used in a complementary way. The success of either approach relies on the interrelationships between FEPs and influences of phenomena on features that reasonably approximate the real-world system.

2.5.3.2.1 Safety Assessment and Safety Functions

A safety assessment and its findings are essential components of the collection of arguments and evidence in support of disposal system safety. Other components supporting the safety of a radioactive waste disposal facility should include the following:

- a description of the waste and the rationale for the chosen waste management strategy
- descriptions of the disposal concept, the disposal facility, the disposal site and its safety functions
- description of the management system applying to the different phases of facility development
- other information that supports development, operation, and closure of the facility

Safety assessments are an analysis of the ability of the site and design to provide the safety functions and meet technical requirements. A safety function is defined as a function by which a component of the disposal system contributes to safety. Safety functions are the capabilities and components of the barriers found within a disposal system that are used to reduce the potential for the release of radioactive material and to ensure that any releases are within acceptable limits. For example, an engineered cover may be placed over a disposal site at closure to provide the safety functions of limiting water contact with the waste, limiting erosion, and providing shielding for direct radiation. The safety functions may differ as the time period changes. For example, the capabilities of a surface cover component may be relied on to achieve short-term safety objectives for an LLW disposal site, while the long-term capabilities of the wasteform itself may be relied on for the long-term safety functions. Analogous to the use of a fault tree analysis for nuclear reactor safety (NEA, 1992), knowing when a safety function is expected to be available and when it can be relied on will affect scenario development. A licensee should identify safety functions, determine phenomena that may influence the safety function, and estimate the persistence and effectiveness of the safety function.

The top-down approach to developing site scenarios has been used in assessments of HLW disposal system performance. Because of the relatively recent development and application of the approach for LLW disposal (EC, 2009b), fewer examples exist for LLW disposal, since

performance is focused more on the relatively active near-surface geomorphology than on the more passive deep geology. The top-down approach to developing scenarios was used in the development of the performance assessment for the Integrated Disposal Facility at the Hanford Site, Washington (Lee et al., 2018). The top-down approach is similar for both LLW disposal and HLW disposal, although less effort is expected to apply the approach to LLW since the amount of effort required (i.e., detail, comprehensiveness, completeness, and degree of iteration) is commensurate with the longevity, concentrations of radionuclides, and quantity of the waste.

2.5.3.2.2 Safety Functions and Scenario Development

A number of organizations that have developed performance assessments for waste disposal have developed site scenarios using a top-down approach to FEPs. Some national programs link FEP records with statements about safety functions (e.g., by specific tools such as FEP charts (SKB, 2006)). Uncertainties in the performance of systems may give rise to multiple site scenarios. A licensee can identify plausible, alternative site scenarios when safety functions are no longer expected to perform as intended. An aim of the scenario development process is to identify deviations from a central scenario, based on the failure of one or more safety functions or the extent or form of degradation of one or more safety functions. The safety functions are usually associated with the engineered barrier system and the barriers of the natural system. A licensee can develop alternative site scenarios by considering the timing of FEPs, their consequences in terms of safety function effectiveness, and the status of other safety functions. Generally, no safety function is assigned to the biosphere.

One proposed methodology for site scenario identification consists of six steps (EC, 2009c):

(1) Define a set of safety functions associated with the engineered and natural barriers for the considered disposal system.

(2) Develop a safety concept based on the functioning of the disposal system in the case of the central scenario. his is strongly directed by the question “when is a safety function expected to be available or when can it be relied on.”

(3) Build a structured set of safety statements. These statements are derived from the requirements for the disposal system, for the sub-systems and for individual components.

(4) Assess the uncertainty affecting the safety statements.

(5) Identify a list of possible alternative site scenarios by considering uncertainties and by incorporating these scenarios in the performance assessment to determine the effect on the safety functions.

(6) Derive a final set of alternative site scenarios. This is done by constructing functional diagrams illustrating the impact of the considered uncertainty in a safety statement on the functioning of the disposal system and by grouping, as far as possible, site scenarios with identical or strongly similar functional diagrams.

Structuring and identifying safety-relevant phenomena, information, and uncertainties are a prerequisite for scenario formulation using a top-down approach. The starting point for the identification of safety-relevant phenomena and uncertainties is the development of a detailed

description of the initial state of the system and its subsequent evolution. Several tools have been developed and applied, including FEP databases, interaction matrices, influence diagrams, assessment model flowcharts, phenomenological analysis of the disposal system, storyboards, timeline with subdivision of timeframes, and process description reports. Further information can be found in NEA (2012).

2.5.4 Constructing Site Scenarios

A licensee may evaluate multiple site scenarios to determine scenario uncertainty. Although a licensee can never eliminate the uncertainty altogether, the technical assessment is an attempt to constrain the uncertainty associated with future events and processes. The following section outlines possible methods a licensee may use, although other valid techniques exist. Appendix D provides additional information on techniques that may be useful.

The output of site scenario construction is a set of site scenarios encompassing most of the plausible future system states and their potential impact. Scenario development should not be done in isolation from the rest of the technical analysis process because it is influenced by and uses information from previous modeling and consequence calculations. The method a licensee uses for developing and selecting site scenarios should be logical, traceable, and transparent. A licensee should document and describe the method used to identify site scenarios and the technical bases for choosing which site scenarios are considered plausible. The licensee should justify that relevant processes and events have been identified and that future evolutions of the disposal system have been considered in the development of the site scenarios.

As introduced in Section 2.5, site scenarios are developed and labeled as either a central scenario or alternative site scenarios based on their likelihood of occurrence (probability) of the FEPs comprising the site scenarios. Common terms are defined below:

- The central scenario represents the evolution of the disposal system within the expected range of uncertainty and in the absence of unlikely disturbances. For some sites, this may be the only site scenario developed. The central scenario may also be referred to with different terminology such as main, nominal, normal evolution, reference, design, or base case scenario.

- Alternative site scenarios, or altered evolution scenarios, represent less likely, but still plausible, representations of disposal system evolution. Alternative site scenarios describe how disturbances affect the evolution of the system.

- “What if” site scenarios are generally considered implausible site scenarios or site scenarios for which the likelihood cannot be accurately estimated. They frequently are the result of requests by stakeholders or other interested parties and lie outside of the normal scenario development process due to their implausibility. They explore the robustness of the system, such as complete failure of a barrier, without identifying a particular degradation mechanism and can be viewed as a form of *sensitivity analysis*.

- Stylized scenarios are typically associated with future human actions (e.g., intrusion) for which few or no relevant data are available, and the probability of an event (e.g., accidents) is assigned a value (e.g., a value of 1).

A licensee should provide the terms used to describe the different types of site scenarios in an assessment and clearly explain their purpose. Once the site scenarios have been developed, a licensee should develop a conceptual model of the disposal system that can represent the associated release, transport, and exposure mechanisms (discussed in Section 2.6).

2.5.4.1 Central Scenarios and Alternative Site Scenarios

A licensee may use site scenarios to evaluate uncertainty related to plausible representations of the disposal site. The central scenario is the expected evolution of the system; the alternative site scenarios are less likely but cannot be eliminated by the licensee. The selected site scenarios should together provide an appropriately comprehensive technical description of the estimated performance of the disposal system.

Central scenarios are usually based on extrapolation of existing conditions into the future and incorporation of changes expected to occur in the future. The central scenario is the scenario best supported by available information and is usually a benchmark scenario against which the impact of alternative scenarios can be compared. The central scenario represents how the licensee expects the system to evolve assuming the proper functioning of the design with anticipated degradation. This scenario should capture the significant features and processes that exist at a site but generally does not consider major events that change the future evolution of the site and the performance of the disposal system, since most licensees will select a potential disposal site where such events are not expected. It is acceptable for a licensee to treat anticipated future evolutions of the disposal system in one numerical model by varying parameter ranges. However, because the disposal system may evolve differently under alternative site scenarios, it may be difficult to represent the different plausible FEPs all in one simulation model, so additional models may have to be constructed. The central scenario is used to determine if the performance objective for 10 CFR 61.41 has been met unless dose assessment results for a receptor in a less likely, but plausible, alternative site scenario produce higher dose rates.

Licensees should develop plausible, alternative site scenarios to investigate the impact of site scenarios that are not expected but cannot be excluded. A licensee is not required to evaluate implausible alternative site scenarios. However, it may be useful for a licensee to evaluate sequences of events and conditions independent of probabilities, to illustrate the significance of individual barriers and barrier functions. In other words, the robustness of the disposal system can be examined in terms of a sensitivity analysis. Alternative site scenarios may represent less likely, but still plausible, modes of disposal site evolution (e.g., processes that impede the effectiveness of a feature important to waste isolation), as well as extreme natural events (e.g., earthquakes, volcanic activity), that are still within the range of realistic possibilities within the timeframe evaluated. Generally, a limited number of external FEPs will be of concern.

Various graphical and tabular techniques have been used to assist in scenario development (NEA, 1992; IAEA, 2004; NRC, 1995c; SKB, 2008). The licensee may find these techniques useful:

- event trees, logic diagrams, and related approaches that analyze alternative combinations of events, resulting system states, or both

- fault or dependency diagrams that set out in a hierarchical fashion the conditions or processes leading to, or contributing to, an end point of interest

- influence diagrams that map the dependencies or interactions between various processes, often indicating the importance of the interaction

- interaction matrices that force a comprehensive questioning of the dependencies between selected key features or processes

- audit tables that force a consideration of the representation of each FEP within the available models and system representation, and evaluation of bias due to omission or simplified representation

- approaches that rely on specialists in their field and expert judgment

The techniques listed above are not mutually exclusive and several tools may be used in combination. For example, influence diagrams and interaction matrices may be useful to explore and illustrate the connection between scientific understanding and the numerical models, whereas event trees and logic diagrams provide a logical structure for selection or generation of calculation cases. Whatever techniques are used, the judgment of analysts is critical to ensure that the scientific understanding is appropriately incorporated in the models. A key value of the graphical and tabular techniques is that they aid communication within projects enabling experts to see the significance of their knowledge within the system context. The techniques can also provide logical structure for the comprehensive documentation of the relevant processes and their representation in models.

2.5.4.2 Receptor Activities and Behaviors

After radioactivity is released to the environment, a receptor may be exposed to contaminated water, soil, air, or other media. A receptor is a member of the public who may be exposed to radiation from the disposal facility. Receptors include members of the public who may be onsite after the institutional control period (e.g., an inadvertent intruder) as well as offsite members of the public. Section 3.4 provides guidance on receptors for the performance assessment. Section 4.2 provides guidance on receptors for the intruder assessment. Some of the approaches used in Section 4.2 can be applied in the performance assessment and used to demonstrate compliance with 10 CFR 61.41.

Disposal facilities may be developed in considerably different areas of the country. A key siting characteristic requirement in 10 CFR 61.50(a)(4) states that a disposal site must not be located in an area where the projected population growth and future developments in the vicinity of the land disposal facility are likely to affect the ability of the land disposal facility to meet the performance objectives or to preclude defensible modeling because of large uncertainties. Sites with lower population density and longer distances from facility locations to current residents are preferred. Though not guaranteed, locations with lower population density in the present and limited natural resources, especially water, are likely to be associated with lower relative population density in the future. Lower population density will be correlated with a lower probability that an inadvertent intruder unknowingly uses the land above the disposal units in the future. A longer distance of a receptor from the facility will result in higher dispersion and dilution of radioactivity released to the environment before exposure of the receptors. The guidance on receptor activities and behaviors is not to take credit for present day low population density. The siting characteristics help to ensure that the uncertainties associated with future human activities are accounted for in part by the siting of facilities in remote locations.

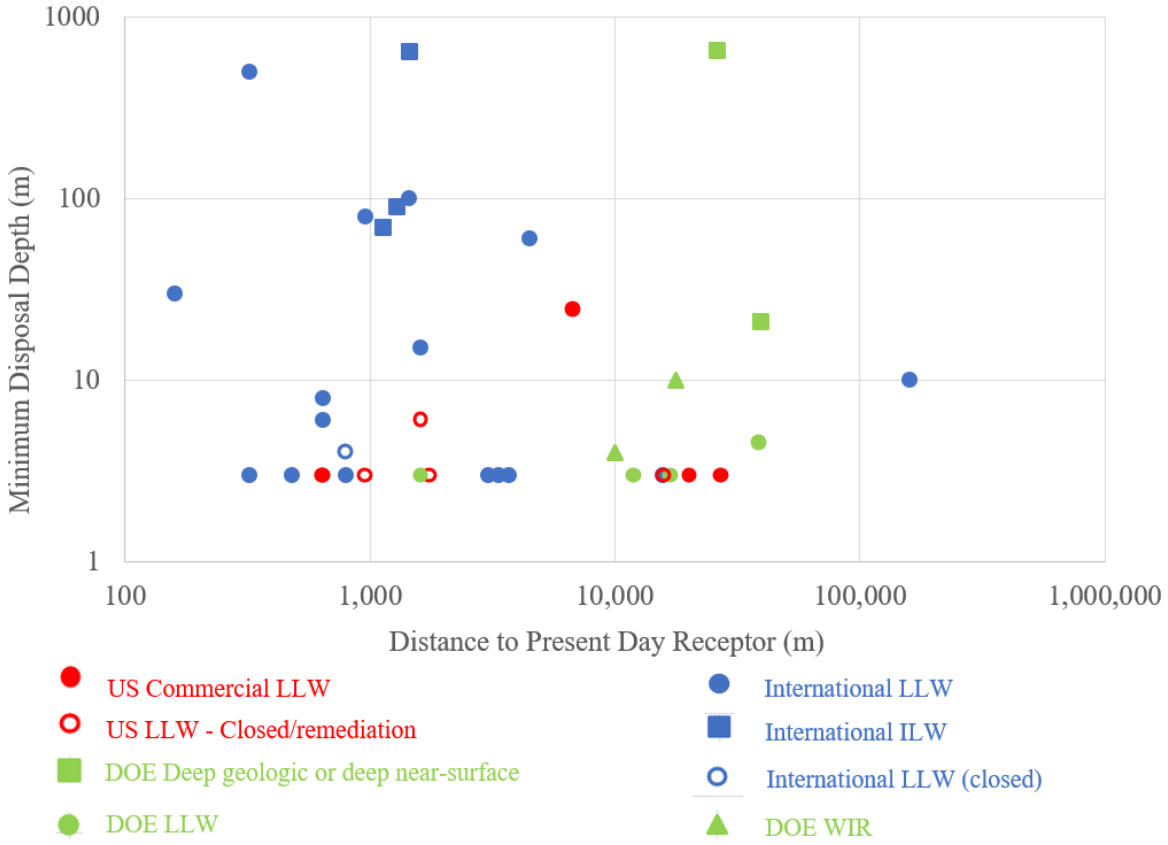


Figure 2-5 Minimum Disposal Depth and Current Residence Distance from Existing Low- and Intermediate-Level Waste Disposal Facilities

Figure 2-5 provides the minimum disposal depth and distance from the center of existing low- and intermediate-level waste disposal facilities to the present day nearest resident. These are facilities that have operated or are operating in the United States and throughout the world. Disposal depths range from a minimum of approximately 3 m (10 ft) for trench-type disposal facilities to more than 650 m (2,100 ft) for mined disposal facilities. Present-day nearest residences are at distances of approximately 0.3 kilometers (0.2 miles) to about 160 kilometers (100 miles). Currently operating facilities in the United States tend to be more remotely located than facilities in Europe. Isolation is a product of disposal depth and distance from the nearby population. Depending on the proximity of the current population to a disposal facility at the time of licensing, increasing the disposal depth can be used to provide more isolation.

For the performance assessment, it is typical to assume a member of the public is located 100 m (330 ft) from the disposal units after the institutional control period. On the other hand, for the intruder assessment it is assumed that a member of the public uses the land above the disposed waste after the institutional control period. As Figure 2-5 shows, this is a conservative assumption with respect to the present day but one that is used to account for uncertainties about future population growth. For disposal facilities located in very sparsely populated areas with conditions that are not very hospitable and no foreseeable mechanisms for the conditions to change, it may be appropriate to relax this assumption or evaluate the significance of the

assumption with sensitivity analyses. Disposal depth will impact the type of intruder scenario (also referred to as an "inadvertent intruder scenario") that may be appropriate to assess. For disposal facility designs in which the depth to the waste has a minimum value of 5 m (16 ft), excavation scenarios that directly disturb the waste are unlikely to very unlikely. In these cases, a drilling scenario is more appropriate to evaluate as the central intruder scenario. Drilling occurs over all depth ranges relevant to near-surface disposal. The occurrence of drilling is not influenced by disposal depth but rather by the availability of resources.

Licensees should estimate potential exposure to the average member of the *critical group* (see Section 3.4.1). The critical group is a group of individuals reasonably expected to receive the greatest exposure to releases over time. The average member of the critical group is that individual who is assumed to represent the most likely exposure situation, based on cautious but reasonable exposure assumptions and parameter values.

Stylized scenarios are commonly defined to represent the human component of the analyses. Limited scientific basis exists for predicting the nature or probability of future human actions over long timeframes. Regulators commonly advocate the use of stylized scenarios to avoid excessive speculation. A special category of FEPs is that related to future human activities that may disrupt the disposal system. Chapter 4.0 provides guidance on developing intruder scenarios for the inadvertent intruder assessment. The consideration of inadvertent intrusion is required by 10 CFR Part 61. To some extent, 10 CFR Part 61 constrains the types of receptor behaviors and activities that need to be considered.

Scenarios of receptor exposure, also known as "receptor activities and behaviors," should not be confused with site scenarios describing a future evolution of the disposal system. Receptor activities and behaviors are a subset of site scenarios defined for the performance assessment (e.g., central scenario, alternative site scenarios) and inadvertent intruder assessment (Figure 2-6). Receptor characteristics and behaviors may vary from site to site; however, certain pathways commonly contribute to exposure to or intake of radionuclides. For example, drinking water and agricultural food production (crops, livestock) commonly contribute to radionuclide intake by many types of receptors. External exposure to contaminated soils and inhalation of resuspended contaminants are also common *exposure pathways*. Recreational use of surface water (e.g., fishing and swimming, which may lead to exposure to contaminated sediments) may be an exposure pathway at some sites.

Licensees are required to consider uncertainty in the demographics and behaviors of potential human receptors in the future. Receptor activities and behaviors[7] are used to account for these uncertainties. Licensees should integrate these activities and behaviors with the site scenarios describing the future evolution of the disposal site. For example, the landscape and hydrological regime at and around a hypothetical disposal site may change in response to natural evolution of the climate, and with these changes, receptors and their habits may change. Depending on the number of plausible exposure pathways, multiple receptors may be associated with a central scenario or alternative site scenarios. In addition, the types of activities and behaviors of a receptor (e.g., farming, gardening, recreation) may differ between the central scenario and alternative site scenarios depending on the environments of those site scenarios.

[7] Receptor activities and behaviors are sometimes referred to as "receptor scenarios."

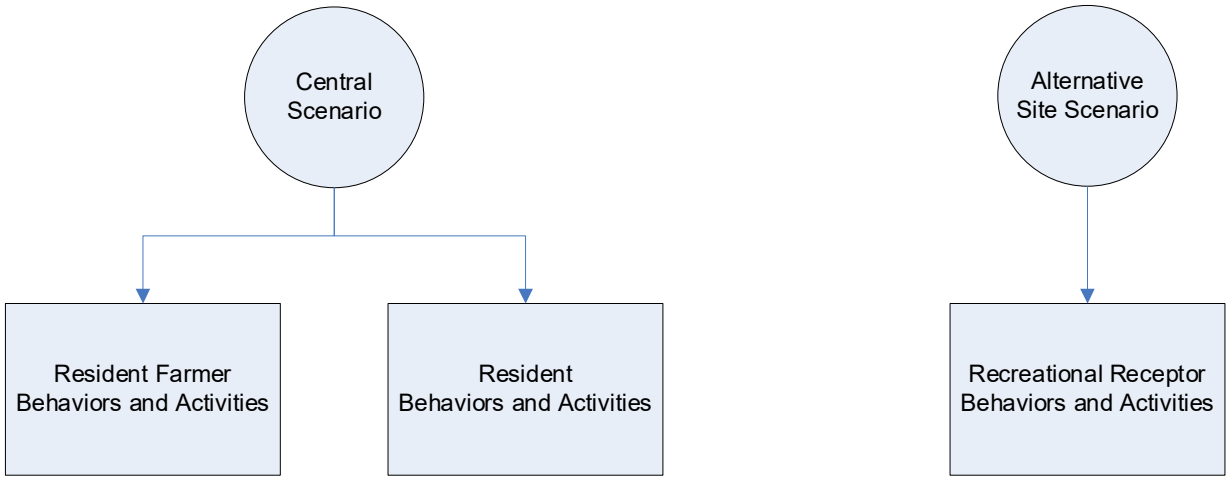


Figure 2-6 A Central Scenario and an Alternative Site Scenario with Different Plausible Receptor Behaviors and Activities

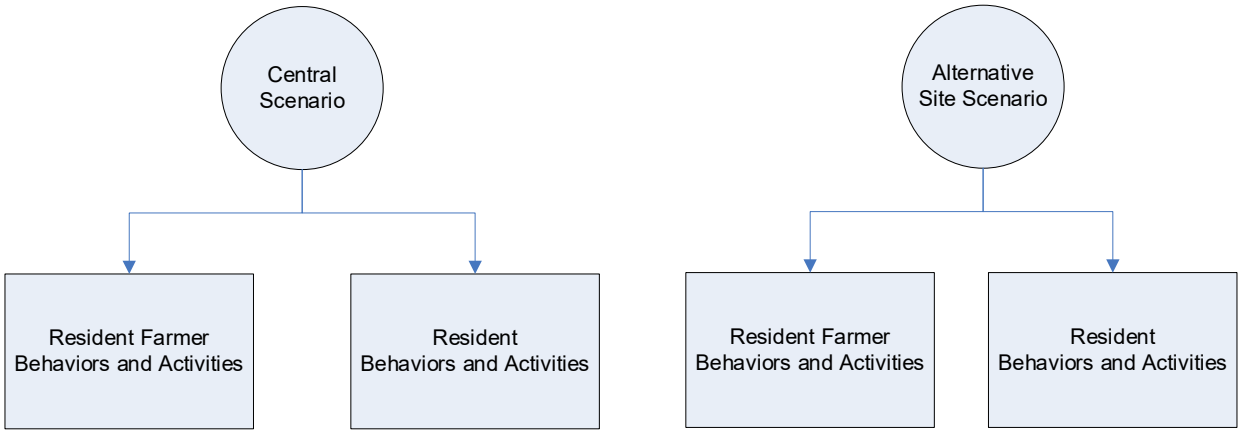


Figure 2-7 A Central Scenario and an Alternative Site Scenario with the Same Plausible Receptor Behaviors and Activities

For example, the future environment of the disposal site in the central scenario may allow a receptor to both live and farm there as shown in Figure 2-6 while the alternative site scenario with a disruptive event may preclude such activities and only allow recreational activities such as hiking and hunting to take place. For a different disposal site, future environments in both the central and alternative site scenarios may allow the same receptor behaviors and activities to occur as shown in Figure 2-7. Section 3.4.1 discusses this topic further.

Receptor activities and behaviors can have a large impact on the projected risks from the disposal facility. Licensees may use generic receptors as described in NUREG-0782, "Draft Environmental Impact Statement on 10 CFR Part 61 Licensing Requirements for Land Disposal of Radioactive Waste," issued September 1981 (NRC, 1981a), or they may develop site-specific receptors. Regardless of the specific method used to represent future receptors, the licensee should sufficiently justify its selection of receptors. Licensees may develop site-specific receptor

activities and behaviors by modifying the exposure pathways included for generic receptor activities and behaviors or consider waste characteristics, disposal practices, site characteristics, and, when appropriate, projected land use. As the assessment time increases from the present day, the relevance of current land use to projected future land use becomes more uncertain. For long-term assessments, dose calculations should use reasonably conservative receptor activities and behaviors, such as the generic receptors (i.e., resident-farmer or resident-gardener). The NRC staff continues to view the generic receptors as reasonably conservative to estimate potential radiological exposures to a member of the public while limiting excessive speculation about future human activities. Licensees should be cautious about adopting the generic receptors *and* exposure pathways by ensuring that the exposure pathways are appropriate for the technical analysis for their disposal site.

Licensees should provide a basis to justify the activities and behaviors of receptors. Use of conservative generic receptor activities and behaviors would require a minimal basis, whereas use of novel site-specific receptor activities and behaviors would require more information. There may be a need to thoroughly investigate and justify the appropriateness of the selected site-specific receptors, which may include evaluation of alternate receptors. If licensees evaluate a receptor based on site-specific conditions, they should provide transparent and traceable documentation of the justification for assumptions used in developing the receptor activities and behaviors (e.g., justify the inclusion (or exclusion) of a particular exposure pathway). If licensees were to provide a side-by-side comparison of the assumed characteristics of receptors (e.g., pathways, consumption rates, and exposure times) with those of generic “screening” receptors used in the dose analyses, the reviewer could determine the importance of the assumed receptor characteristics. This sort of comparison is strongly recommended. Generic receptor characteristics are found in a variety of documents (NRC, 1992; NRC, 1981a; NRC, 1982b). Estimates of future disposal site performance that are primarily based on the engineered design, waste characteristics, and site characteristics are likely to be less speculative than estimates that rely on assumptions about future human behavior.

When assessing the dose to the receptor beyond the site boundary, licensees should consider potential receptor locations. The site boundary begins at the end of the buffer zone, which should extend 100 m (330 ft) from the boundary of the disposal units. Generally, receptor locations at the site boundary would be expected to receive the greatest radiological exposures since these locations tend to minimize the opportunity for dilution that may occur at greater distances from the site. However, this may not always be the case, particularly for sites with preferential transport pathways (e.g., fracture zones) or physical constraints that might limit exposure of the receptor to the media (e.g., non-potable groundwater). Licensees should demonstrate that the selection of an offsite receptor location does not bias the outcome of the performance assessment such that radiological exposures are significantly underestimated. It may be conservative in most instances to assume the receptor is at the site boundary, but a licensee may want to provide a comparison of the results based on the present-day location of the nearest receptor to communicate the conservatism of the decision to stakeholders. Receptor locations may be different for transport in various environmental media; the highest impact from an air pathway may be at a different location than the highest impact from a water pathway. However, it is not expected that the peak exposures at different receptor locations would be additive. Rather, licensees need to demonstrate that the performance objectives will be met for the location in which the receptor would be expected to receive the largest annual dose from all significant exposure pathways (e.g., peak annual all-pathway dose). Determination of the receptor location should also consider the evolution of the site environment

during the compliance period. For instance, natural evolution of the climate over the compliance period may alter whether the groundwater is potable or of sufficient yield and thus, potentially, the location of the offsite receptor well.

A site-specific assessment would typically consider climatic and environmental conditions. As discussed in NUREG-1573, performance assessments should consider variability in natural conditions, processes, and events (NRC, 2000a). The selection of receptor activities and exposure pathways in the performance assessment and inadvertent intruder assessment should consider the variability in natural conditions. For a typical commercial LLW disposal facility, where the hazard from the inventory remaining at 500-1,000 years is expected to be low, and allowable limits on long-lived radionuclides can be set, licensees should avoid unnecessary speculation about major changes to future climate, such as glacier formation. This is because the human population would be dramatically affected by the natural process and the radiological impacts would likely be secondary, in part because the inventory of long-lived waste is limited. However, licensees should consider more gradual changes in performance assessment modeling for significant quantities of long-lived radionuclides, especially for performance period analyses (Chapter 6.0). For example, natural cycling of climates may induce variation in the nature, timing, and magnitude of meteorological processes and events. For long-lived waste streams, licensees should consider these gradual changes when evaluating impacts to members of the public. If a licensee uses current site conditions to eliminate what would otherwise be considered credible land use scenarios and the waste is long-lived, the performance assessment and intruder dose assessment should consider expected changes to climate and environmental conditions resulting from natural cycling of the climate. Changes to the climate may make the eliminated land use scenarios more or less likely to occur in the future.

A licensee's assumptions about land use should focus on current practice in the region of concern, which can be as large as an 80-kilometer (50-mile) radius. To narrow the focus of current land practices, the licensee can use information on how land use has been changing in the region and should give more weight to land use practices either close to the site or in similar physical settings. Licensees should also evaluate land uses that occur in locations outside the region of concern that share characteristics (temperature, precipitation, topography) expected for the region of concern over the duration evaluated. Consideration of environmental analog regions may help identify whether present-day land uses have been driven by past socio-economic development. Land uses primarily resulting from socio-economic development are generally more uncertain over longer time periods than land uses primarily resulting from physical conditions (e.g., climate).

2.6 Conceptual Model Development

A licensee should review the information provided for the assessment context, system description, and scenario development steps of the assessment approach and use it to develop a conceptual model, and if required, alternative conceptual models, of the site. A conceptual model is a well-defined, connected sequence of features and processes describing the behavior of the system within a site scenario, for example, the central scenario (see Figure 2-2). The conceptual model of the site should qualitatively describe how the FEPs and significant barriers interact with one another and how the site functions. The conceptual model of the future disposal site may also significantly affect the behavior and activity of a receptor. For example, the productivity and location of an aquifer can determine occupation and place of residency. Licensees should describe any simplifying assumptions, which may be necessary when a

licensee develops the site conceptual model. Regulators should review simplifying assumptions and determine whether the licensee has provided an adequate technical basis. Simplifying assumptions typically involve the geometry and dimensionality of the system, initial and boundary conditions, time dependence, and the nature of the relevant physical and chemical processes.

To identify areas that require more detailed consideration and reduce model uncertainty, an initial simple and conservative conceptual model could be developed based on limited data and design information. Further refinement of the conceptual model by a licensee should reflect an increased focus on significant radionuclides and processes. Licensees should identify simplifying assumptions. In the conceptual model, the simplifying assumptions should be internally consistent and should also be consistent with existing information. Licensees should justify simplifying assumptions based on the current understanding of the system (NCRP, 2005).

For the site scenarios that will be quantitatively assessed, the conceptual model must be amenable to mathematical representation. The conceptual model must have enough detail to allow mathematical models to be developed to describe the behavior of the system and its components. Conceptual models developed by the licensee provide the framework for the computational models. It is important that the conceptual model is transparent and supported with an adequate technical basis. More than one conceptual model may be consistent with available information or lack of information (i.e., uncertainty). If the set of alternative conceptual models does not represent the full range of possibilities, conceptual model uncertainty will be underestimated. Appendix D provides additional information on techniques that have been used to develop conceptual models.

NUREG/CR-6805, "A Comprehensive Strategy of Hydrogeologic Modeling and Uncertainty Analysis for Nuclear Facilities and Sites," issued July 2003 (NRC, 2003b), discusses conceptual model uncertainty and some of the most important activities associated with developing alternative conceptual models to reduce model uncertainty. These include the following:

- maximizing the number of experts involved in the generation of alternative conceptual models
- minimizing inconsistencies, anomalies, and ambiguities
- articulating uncertainties associated with each alternative conceptual model
- obtaining key data to support each alternative conceptual model
- considering alternative representations of space-time scales and features and processes

A variety of approaches have been used to facilitate the development of conceptual models in a traceable manner. Three examples taken from IAEA (2004) are given below.

The "safety assessment comparison approach" relies on the expert judgment and experience of the analyst carrying out the assessment. The first step is to identify the key release, transport, and exposure media by reviewing the relevant FEPs associated with each site scenario. The mechanisms by which the associated release, transport, and exposure may occur are considered for each site scenario.

Two strategies can be used:

- The deductive strategy reviews how releases might occur and considers the possible transport and exposure mechanisms and the associated impacts.

- The inductive strategy analyzes the impacts, considers the exposure and transport mechanisms that might have caused the impacts, and the associated release mechanisms.

The "interaction matrix approach" for developing a conceptual model allows the graphical representation of system interactions using formalized procedures but does rely on expert judgment and is data intensive. The method starts with a top-down approach—dividing the system into constituent parts. The resulting matrix and the FEP list contents can later be compared. Using the interaction matrix approach to facilitate conceptual model development has the advantage of disposal system components being included explicitly in the interaction matrix and analyzed in greater detail by creating one or more sub-matrices. The interaction matrix approach allows FEP interactions and pathways to be mapped, which is an important step in developing and defining a conceptual model and in the logical progression to a mathematical model. Moreover, the examination of how the system components relate to one another may help to identify new, previously unrecognized relevant characteristics of the system. When using the interaction matrix approach for developing a site scenario, the convention is to allocate off-diagonal elements in the direction of contaminant migration. In this way, contaminant migration pathways and the associated exposure pathways and exposure groups can be traced and translated into the conceptual model.

The "influence diagram approach" for developing a conceptual model allows the interaction between FEPs to be identified in a logical and systematic way. Advantages and disadvantages are similar to the interaction matrix approach, although the influence diagram generally contains more detail than the interaction matrix. FEPs are represented by boxes and interactions between FEPs are illustrated by arrows showing the influence direction. The number of arrows between two FEPs will be equal to the number of influences between them. Only direct influences should be represented in an influence diagram.

2.6.1 Alternative Conceptual Models

Licensees should adequately describe and document conceptual model uncertainties. The performance assessment documentation should include the assumptions, limitations, and uncertainties in the models. Multiple representations of the system may be consistent with the available data. An alternative conceptual model describes how the system might work that is different than the most plausible conceptual model, but yet consistent with available supporting information. For example, a site scenario may have two plausible conceptual models: a fracture flow conceptual model and an alternative matrix flow conceptual model; the model outputs from each may yield significantly different results. In general, licensees should select the conceptual models that best represent available data and associated uncertainty. However, when data are sparse, multiple conceptual models may represent the available data. In this case, licensees should select the plausible conceptual model that provides the most conservative result or collect additional data to reduce uncertainty and determine which alternative conceptual model provides the most realistic representation. All conceptual models do not need to be abstracted and evaluated (e.g., it is recommended that licensees avoid "what

if" type conceptual models and site scenarios), but all models that are reasonably consistent with available information should be considered. If less likely, yet plausible, alternative conceptual model describing the disposal site system within either a central scenario or an alternative site scenario produces higher dose rates than the original conceptual model, its numerical representation within the performance assessment should be considered in deciding whether the performance objective for 10 CFR 61.41 has been met. Reviewers should perform an independent evaluation of model uncertainty and consider whether more than one conceptual model should be evaluated, especially for complex sites.

2.7 Numerical Model Development and Assessment

The numerical model development and implementation process typically consists of representing the conceptual models and their associated processes in mathematical models and using or developing software to perform numerical simulations of the mathematical models.

This section provides guidance on the development of numerical models including (1) specific information on model abstraction of the conceptual model for it to be represented mathematically, (2) *model integration*, and (3) interpretation of model results.

2.7.1 Numerical Models

In general, licensees will rely on numerical or computational models[8] to estimate the future performance of a disposal site. Additionally, the analyses are likely to be iterative, and the detail and effort involved in a particular iteration may vary depending on the phase of facility development and the knowledge about the disposal site. For instance, during siting of the facility, simple models may initially be used to evaluate candidate sites, whereas more complex models may be needed during licensing of sites with unique characteristics such as complex site engineering or natural features. Although specific computer codes may be discussed or referenced in this guidance, the NRC staff does not endorse any specific code or modeling software package for analyzing the performance of an LLW disposal site.

A licensee may develop a new model or acquire an existing model. A previously developed numerical model may be used, or general-purpose simulation software (e.g., GoldSim) may be used to develop a model. This section discusses information relevant to the development of numerical models. Licensees should select a numerical modeling approach that can appropriately represent the site's conceptual model and implement the mathematical model. Regardless of whether a model is acquired or developed, the licensee is responsible for the QA status of the model. Licensees should ensure that the selected numerical model is developed with adequate QA/QC, as discussed in Section 2.2.1.

Licensees should select or develop a numerical model that can represent the conceptual model of the disposal site, including the site-specific FEPs. The numerical model(s) will express the conceptual model as one or more mathematical expressions (e.g., algebraic expressions, differential equations) with a set of boundary and initial conditions that are then solved. In many cases, more than one mathematical formulation could appropriately represent a conceptual model. Further, the expressions must be consistent with the scientific understanding of the

[8] For this document, the terms "computational model" and "numerical model" are used interchangeably. In addition, the terms "computer code," "program," and "software" are used interchangeably.

physical processes, information available to parameterize the equations, and the spatial or temporal scales that the expressions are intended to represent.

Three approaches used to solve equations are analytical, semi-analytical, and numerical. These methods are summarized here from IAEA (2004). Analytical methods provide exact solutions and can be computationally efficient; however, the methods are typically available only for simple equations involving homogeneous or uniform spatial domains (e.g., one-dimensional steady-state flow and transport with simple boundary conditions). Semi-analytical methods are more flexible than analytical methods for solving more complex problems (e.g., flow and transport involving multiple sources and sinks). Numerical methods often discretize the spatial and temporal domains into a finite number of compartments or increments and solve the equations by iteration, matrix methods, or some combination of the two. Numerical methods can include, but are not limited to, finite-difference, finite element, method of characteristics, random walk, and analytic elements. Numerical methods allow for consideration of spatial and temporal variability, complex geometry and boundary conditions, and dimensionality. However, numerical methods can be computationally intensive, require specialized education and training and can introduce numerical errors (numerical dispersion).

In the documentation of the numerical models, licensees should include the following:

- the mathematical formulation
- model assumptions
- sensitivity to ranges of input data
- consistency of the transport pathways in the numerical model with the transport pathways of the conceptual model
- accuracy of the software to reflect the model's mathematical formulation (discussed in Section 2.2.1.3)
- the correct representation of the process or system
- stability of the numerical methods (discussed in Section 2.2.1.3)

Licensees should provide comparisons between theory and experimental results, field observations, and other supporting information that may involve interpretation and extrapolation (see Section 2.2.4 for additional information on model support). In cases for which more than one mathematical formulation may be appropriate, licensees should document why a particular formulation is preferred over other approaches and, importantly, the limitations of the selected approach. Licensees may also wish to consult Appendix I to Safety Guide No. GS-G-3.4, "The Management System for the Disposal of Radioactive Waste," issued 2008 (IAEA, 2008), for information about controlling numerical models for waste disposal facilities.

Reviewers should evaluate the numerical models to ensure compatibility with the site conceptual model including the source-term, transport pathways, and receptor behaviors and activities. For example, numerical models designed for a particular onsite receptor behavior and activities may not be appropriate for assessment of offsite receptor behavior and activity. Reviewers should also ensure that the assumptions in the numerical modeling are consistent

with the site conceptual model and that deviations will not significantly affect the ability of the numerical model to represent the conceptual model. Reviewers should investigate the sensitivity of numerical model calculations to variations in input parameters to ensure that the model is numerically stable over the range of input parameters for expected site conditions. The range of input parameters may be different for alternative site scenarios compared to the central scenario.

Licensees may select modeling software, commercially available or otherwise, or develop software or models on a site-specific basis. The modeling may be completed with commercial (proprietary) software, modified existing software, or software specifically developed for the implementation of the chosen mathematical models representing the site-specific conceptual model. Modified existing software and software specifically developed for a site-specific application are referred to in this guidance as "user-developed" software. The use of commercial software has the advantage that the software has been verified and validated during development. Commercial software may have a history of application to a range of different analogous problems. In contrast, licensees need to develop, verify, and validate user-developed software. However, user-developed software does have the advantage of being tailored to the needs of the specific problem to be addressed. In all cases, it is necessary for a licensee to consider the process of software design based on a given mathematical specification (IAEA, 2004). Licensees should ensure that the software used for the numerical models conforms with the active standards of the IEEE standards for software development, QA, testing, verification, and validation (e.g., IEEE Standard 730-2002, "IEEE Standard for Software Quality Assurance Plans,"(IEEE, 2002)). Reviewers should be familiar with the NRC's Software QA/QC guidance (i.e., NUREG/BR-0167, NUREG-0865) to ensure that the licensee's QA/QC procedures for selection and development of numerical models are comparable. It is recommended that proprietary codes are peer reviewed by the appropriate technical community before use in a licensing decision; however, peer review is not required. The pertinent regulator may need to perform this function. Section 2.2.1.3 provides further guidance on software QA/QC.

2.7.2 Model Abstraction and Model Simplification

A licensee will need to abstract the conceptual models they develop to translate the concepts into mathematical terms. Model abstraction is the process of representing a conceptual model of a dynamic site in the physical world as a mathematical model governed by equations that are implemented in a numerical model. When abstracting a model, a licensee should determine the level of detail that should be preserved in the overall performance assessment model (or intruder assessment model). Figure 2-8 provides the types of abstraction that may occur in the model development process. The example provided is for development of a performance assessment model. Figure 2-9 provides a representation of the development of a hydrologic model (e.g., process model) for use in a performance assessment. Various abstractions may occur at many different steps in the model development process. Model abstraction builds on insights gained from FEP development and site scenario analysis. In an LLW performance assessment, several model abstractions typically support the licensee's overall assessment to demonstrate compliance with the performance objectives. These abstractions usually include models of projected climate and infiltration, degradation of barriers, source term release, transport through environmental media, and potential exposures to a receptor in the biosphere. Section 3.3 of NUREG-1573 (NRC, 2000a) and Chapter 3.0 of this guidance discuss these model abstractions.

Several factors can affect the complexity of the models that a licensee may use. Some level of simplification is generally required to translate the concepts of a conceptual model into mathematical terms. This simplification can take several forms, such as the following:

- the simplification of the geometry or structure—for example, representing a transport medium as homogeneous and isotropic
- the omission of processes and interactions—for example, neglecting kinetic terms in chemical reactions
- a reduction in spatial or temporal resolution

Model simplification is similar to model abstraction but more closely tied to modifying numerical models. It consists of simplifying a complex numerical model into a reduced numerical model that has fewer components and is quicker to execute while still maintaining the validity of the simulation results. For example, it may be appropriate for a licensee to reduce the detail of a submodel (e.g., for waste release) for incorporation in a performance assessment model. The licensee should demonstrate that the detail that has been reduced or eliminated by model simplification is not essential to the estimated performance. This can be difficult to do with complex models and when supporting information is sparse. In addition, the simplified model should be compared to the detailed model to show that the simplified representation is appropriate. This comparison should be performed for base case models, as well as for models used to represent alternative site scenarios. A simplified "model" can be something like a data value or lookup table. For example, use of distribution coefficients (K_d) for radionuclide transport is a simplification of more complex phenomena. The use of simplified models can introduce uncertainties and biases.

It is important that a licensee document both model abstractions and model simplifications. A licensee may need to perform abstraction of the real-world system into a mathematical model using an iterative process. It is useful for a licensee to factor in limitations of the computational platform during conceptual model development. It is not appropriate to limit the assessment to the capabilities of existing computational platforms. In some cases, a licensee may need to develop a new computational model to appropriately assess the problem. In other words, the problem dictates the solution rather than the solution constraining the problem.

A natural tendency of a model developer is to preserve all detail that may be developed through the iterative process of developing models supporting the performance assessment. However, this approach has disadvantages. The additional detail can make documentation, review, and understanding the modeling effort more difficult. In addition, many *sensitivity analysis* methods work better with a reduction in the number of inputs analyzed by the methods. Model simplification should achieve a balance by preserving essential detail and eliminating unimportant detail. Whereas a licensee may use documentation to preserve and discuss the modeling development process, the final model used for the licensing decisions should be as simple as possible.

2.7.3 Model Integration

The objective of the review of integration is to determine if the site and design description, data, parameter uncertainty, models, and model uncertainty have been appropriately integrated in the assessment. The reviewer should ensure that representations of the disposal system design

and natural system features have been adequately integrated into the technical analyses. Assumptions, data, and models should be consistent throughout these analyses. Inconsistencies should be explained in the documentation or be corrected. Boundary and initial conditions should be consistent between different submodels in the performance assessment. Information that is passed between submodels should be verified to be of the appropriate temporal and spatial scale. If different models are used, information passed between models should be assessed to determine if unit conversions were made. Graphical representations can be useful to communicate integration in the technical analyses. Influence diagrams can be useful to document concisely the connections among submodels in the performance assessment.

2.7.4 Analysis and Evaluation of Results

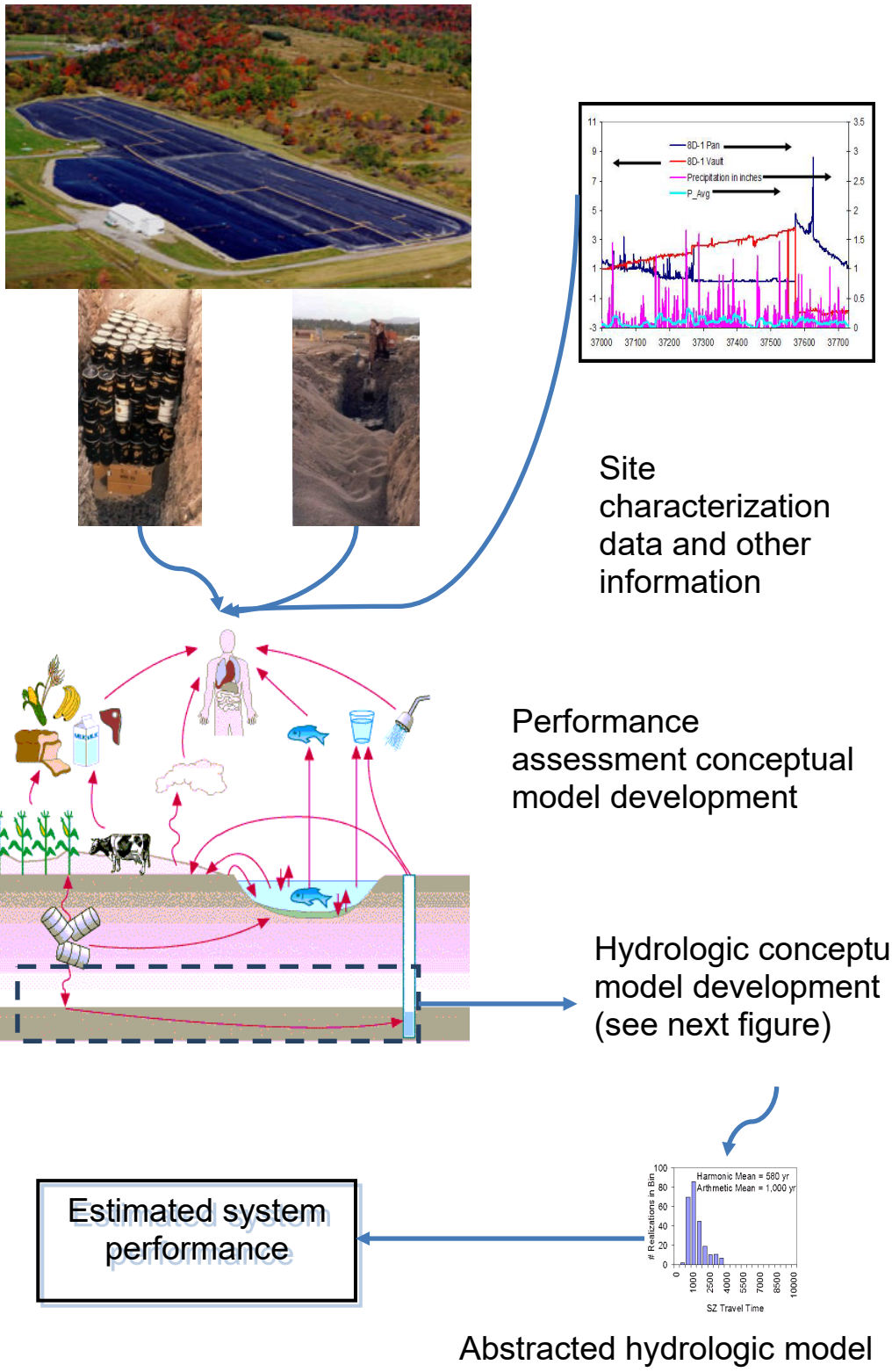
Probabilistic approaches to performance assessment are preferred because they readily permit the propagation and assessment of the impact of uncertainty on the model results. However, use of a deterministic model to demonstrate compliance with performance objectives may be acceptable. In general, if deterministic modeling is used, it should be reasonably conservative, such that a subject matter expert, with minimal interaction with those who performed the assessment, could conclude that the analysis is conservative. Because the models are used to make safety decisions and a central tendency, deterministic model cannot directly include the impact of uncertainty, a conservative representation is warranted. Independent reviewers should evaluate the modeling in sufficient detail to be confident that the analysis is conservative.

For probabilistic analysis, the appropriate metric to use for comparison with the annual dose limits is the peak of the mean result. The peak of the mean is calculated by estimating the mean dose result over all probabilistic realizations, then taking the peak value of the mean curve. In some cases, the mean curve may represent a high percentile of the projected output. Licensees may propose more conservative metrics; however, the peak of the mean is a protective metric for probabilistic analyses.

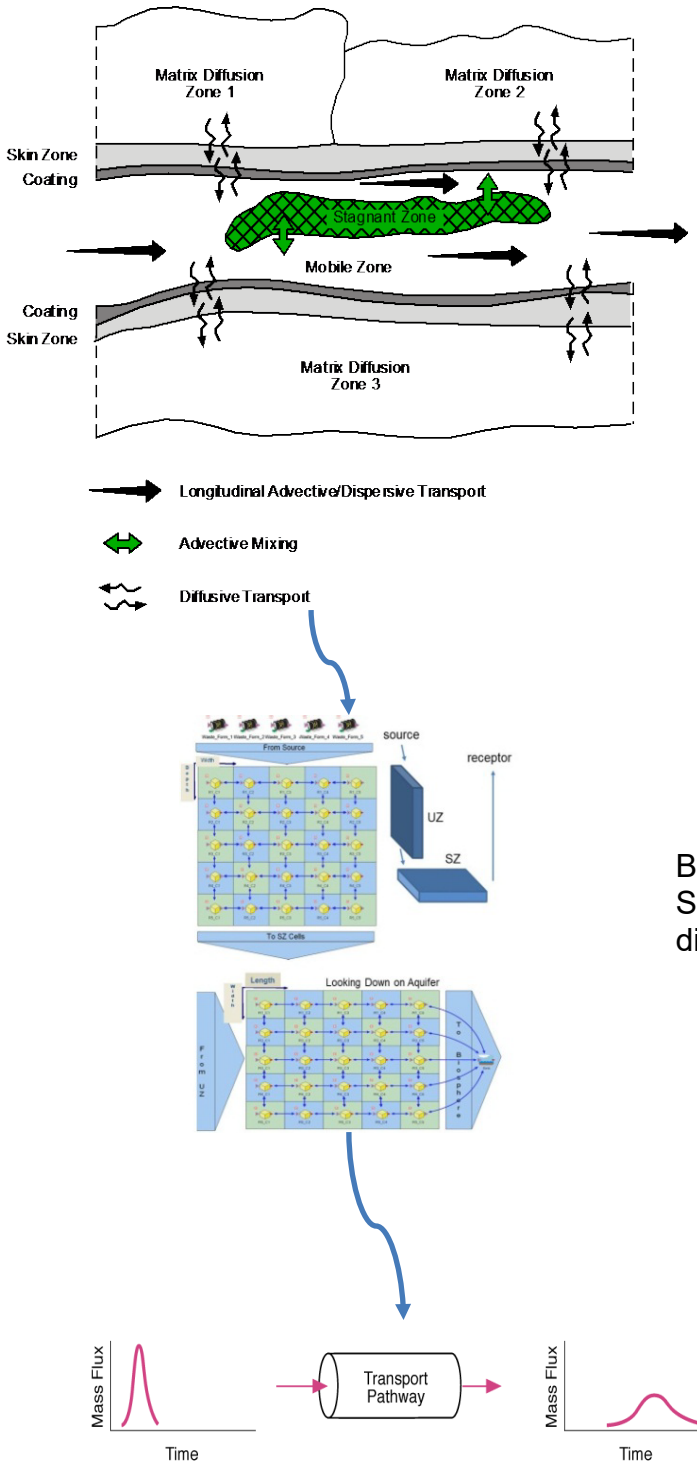
If a probabilistic approach is used for the performance assessment, the reviewer will need to determine whether there is significant “risk dilution” affecting the calculation results. Risk dilution results when overly broad parameter distributions are selected primarily for processes that affect the timing of impacts. Although large uncertainty ranges may seem to be “conservative,” overly broad uncertainty ranges can artificially depress the peak of the mean dose result. For example, selection of an overly broad range for the K_d for neptunium (Np)-237 in the saturated zone may result in the estimated time of arrival of the contaminant at a receptor location being artificially spread over the assessment timeframe, thereby “diluting” the risk at any one point in time. A comparison of the mean of the peaks and the peak of the mean can be useful to determine the significance of the timing of estimated impacts.

For modeled processes and input parameters that are highly uncertain and cannot be established as conservative, sensitivity analyses are necessary to establish the relative importance of these processes and parameters to the performance assessment dose calculations. NUREG-1757, Volume 2, Appendix I, Section 1.7, and NUREG-1573, Section 3.3.2, summarize different methods for sensitivity and uncertainty analyses. In addition, Mohanty et al. (2011) provides a historical summary of methods used or developed at the NRC. Sensitivity analyses may identify the need for additional site characterization to adequately support the technical analyses.

1



2 **Figure 2-8 Overview of Abstraction as Applied to a Performance Assessment Model**



Hydrologic conceptual model

Boundary conditions
Spatial and temporal discretization

Numerical model development

$$m'_{is} = -m_{is}\lambda_s + \sum_{p=1}^{NPs} m_p \lambda_p f_{ps} R_{sp} (A_s/A_p) + \sum_{c=1}^{NCi} f_{cs} + S_{is}$$

1 **Figure 2-9 Abstraction of a Hydrologic Model for a Performance Assessment Model**

Parameters to evaluate in a sensitivity analysis may be selected with a variety of different approaches, and the appropriateness of the approach depends on the problem. It is anticipated that sensitivity analysis evaluations and the performance assessment model development may be iterative. The initial approach evaluated may not be the final approach selected. The reviewer should keep in mind that the purpose of the sensitivity analysis is to evaluate uncertainty, and in more limited cases, variability in the assessment. One of the simplest methods for sensitivity analysis uses a top-down approach in which the risk reduction of each component (e.g., infiltration, unsaturated zone, wasteform, engineered barriers, saturated zone, biosphere characteristics) of the performance assessment model is identified by starting with a hazard and calculating how each component reduces the risk from the hazard. Subsequently, for the most important components, the licensee performs a quantitative or qualitative evaluation of the parameters to identify those that are most likely to influence the output from the component. Complications arise because an individual component's importance in the system can be relative to the performance of other components. For example, the hydraulic conductivity of a drainage layer may not appear to be risk significant as long as a cover limits water flow, but the drainage layer may be very important to performance if the cover is degraded (see Section 6.3.1). Developing an understanding of the importance of parameters and models in a performance assessment is a time-consuming process that is best accomplished by exploring a variety of approaches.

If the licensee has performed a deterministic performance assessment, then the reviewer should examine the sensitivity analyses provided. The reviewer should evaluate the licensee's basis for selecting the parameters and combinations of parameters used in the sensitivity analysis. The ranges in the parameters selected should be consistent with the variability and uncertainty in the parameters, and the selected ranges should give the reviewer confidence that the effects of the uncertainty on performance are bounded. The reviewer should examine the technical basis used to support the variability and uncertainty. Appropriate combinations of parameters should be used to capture the interdependence of key parameters and the consequences associated with changes in combinations of key parameters. The reviewer should consider combinations of parameter values that are likely to occur because of common causes. For example, an aggressive chemical environment could increase both the corrosion rate of waste containers and the rate of leaching of radionuclides from a wasteform. If a licensee were to analyze the increased corrosion rate independent from analysis of increased leaching rate of radionuclides from the wasteform, the risk significance of an aggressive chemical environment may not be properly identified.

Different approaches to performance assessment calculations (e.g., deterministic, probabilistic) have advantages and disadvantages with regard to uncertainty and sensitivity analysis. While deterministic analysis can be a suitable approach for performance assessment, it can also present a challenge for a dynamic system that responds nonlinearly to the independent variables. When numerous inputs (e.g., data or models) are uncertain, evaluating the impacts of the uncertainties on the decision can be difficult. Typical one-at-a-time (OAT) type of sensitivity analysis, where a single parameter is increased or decreased, will identify only local sensitivity within the parameter space such that it may not clearly identify the risk implications of the uncertainty in the parameter. When a licensee must address multiple uncertainties, assessment of the uncertainties with OAT sensitivity analyses should be avoided because it can lead to misleading results (see Section 6.3.1). In addition, uncertainties should not be relegated to assessment in OAT evaluations; uncertainties that are expected to apply to the system should be represented in the compliance case (base) results. A deterministic approach can be useful when the analysis can be demonstrated to be conservative. A probabilistic approach can

have distinct advantages when there are different uncertainties that may significantly influence the results of a performance assessment. For example, the uncertainty introduced by the changing effectiveness of a chemical barrier over time may be represented by selecting appropriate ranges for the radionuclide transport parameters for the materials of the barrier.

The type of analyses that may be suitable for a particular problem will be tied to the amount of model support available. A licensee should not perform complex assessments if model support is not available to provide confidence in the results of the assessments. Simple, well-supported assessments are preferred over complex, weakly-supported assessments, although the NRC staff is aware that simple, well-supported assessments may not be possible for complex sites with greater uncertainty. In that case, scenario, model, and parameter uncertainties should be adequately bound, and the results well documented in the assessment, using approaches discussed in Section 2.2.

2.7.4.1 Parametric Uncertainty and Sensitivity Analyses

This subsection briefly lists the various techniques for parametric uncertainty and sensitivity analyses. Information was compiled from a variety of sources including Till and Grogan (Eds.) (2008) and Mohanty et al. (2011). This section was written to provide summary information rather than specific direction to licensees and regulators.

The most typical representation of uncertainty is through model input parameters. Represented uncertainties are categorized as aleatory and epistemic. Aleatory uncertainty, also known as variability, refers to the stochastic properties associated with many parameters (e.g., average annual or seasonal stream flow in a stream is well known but difficult to predict for a single specific day in the future). Epistemic uncertainty is the uncertainty in the value of a parameter that is fixed but unknown (i.e., with additional research or field studies the uncertainty could be reduced). This uncertainty can be represented as a distribution of possible values. A parametric uncertainty analysis allows uncertainty to be represented and propagated throughout a model, uses various techniques to examine uncertainties associated with various parameters, and subsequently analyzes its effects on the output results. Natural variability and lack-of-knowledge uncertainty are more frequently being separated in the methods used to propagate the uncertainties through models. For example, Monte Carlo realizations of a model can be used to calculate the effects of both aleatory and epistemic uncertainty.

The Monte Carlo method involves choosing parameter values using a random, or stochastic, selection scheme. The analysis can be used to sample possible values of parameters from probability distributions. A modified sampling procedure, known as Latin Hypercube Sampling, may be more advantageous because it forces a more even coverage of the sampled space. Convergence of model results to a stable mean dose curve is important because it is the mean value of the dose that would be compared to the regulations. Depending on the models, a large number of realizations may be needed to reach a stable, converged result. Additional methods for propagating uncertainty through models besides the Monte Carlo method, include mathematical approximation techniques and analytical methods using mathematical statistics.

Additional components of a parametric uncertainty analysis involve descriptive statistics such as probability distributions that can be described in terms of their functional forms and parameters, their graphical forms, and various descriptive statistics. Probability density function is an example of a probability distribution that describes the likelihood of observing a particular value for a variable across a range and is given the functional representation $f(x)$. A cumulative

distribution function is another functional form of a probability distribution and is obtained by integrating the probability density function. Tolerance levels reflect the uncertainty in the estimates of the mean and variance that are obtained from sampling a distribution.

Because of the lack of data, it may be difficult for a licensee to construct probability distributions for some parameters. Depending on the available data and the assessment of the probable shape, distributions can take many forms such as, but not limited to, uniform, log-uniform, normal, exponential, lognormal, Poisson, triangular, and chi squared.

Correlation among input parameters to a model can have large impacts on an analysis. Correlation in the broad sense refers to any statistical dependence. In common usage it may refer to the statistical relationship between two parameters. Consideration of correlations between parameters can be very important in technical analysis supporting LLW disposal. Lack of correlation of parameters may result in non-physical results. For example, if the consumption rates of all food items are represented by probability distributions the total caloric intake may be non-physical in some realizations.

Statistical correlations should be appropriately considered and represented in the analysis. Parameters may be negatively correlated, not correlated, or positively correlated. In some cases, more complex statistical relationships may be present. Spurious correlations can sometimes occur. If correlation between parameters is not intuitive, consideration should be given to the potential for spurious correlation. Correlation is not causation and is not a substitute for a proper conceptual model. However, it is normal in the technical analysis process to correlate parameters that may in fact be driven by more complex phenomena that has not been explicitly represented in the analysis. For example, the corrosion rate of carbon steel could be correlated to the infiltration rate because a detailed model for carbon steel corrosion was not necessary in the performance assessment model.

Confidence intervals are interval estimates of population parameters calculated from observations, but where the interval computed from a particular sample does not necessarily include the true value of the parameter. A 95 percent confidence interval suggests that, if the same population is sampled on numerous occasions and interval estimates are made on each occasion, the resulting intervals would bound the population parameter in approximately 95 percent of the cases. Confidence intervals are a technique to understand the role of sampling error in the averages and percentages that are common in research. Three properties impact the width of a confidence interval: confidence level, variability, and sample size. A larger sample size normally will lead to a better estimate of the population parameter. In practice, confidence intervals are typically stated at the 95 percent confidence level; however, when presented graphically, confidence intervals can be shown at several confidence levels, for example 90, 95 and 99 percent.

Parameter sensitivity analysis includes methods used to assess the influence of changes to input parameters on model outputs. Sensitivity analysis provides insights that an analyst can use to refine the model or the model input parameters without devoting undue effort to unimportant areas of the assessment.

Parameter sensitivity analysis techniques used in the performance assessment analyses are categorized as statistical and non-statistical. Statistically based methods rely on some form of random sampling of input parameters to generate a set of responses from the performance assessment models. They use statistical hypothesis testing to analyze the relationships

between the input parameters and the model responses to make inferences about the most sensitive input parameters. Analyses can be performed on either the original input set or output value pairs which can be altered by transformations to enhance the sensitivity calculations. Transformations include normalization, rank transformation, logarithmic transformation, scaled-power transformation, standardization, and Gaussian transformation. Because risk is usually based on mean doses, regression results based on transformed variables should be used only on the independent variables. Some non-statistical sensitivity analysis methods are described later in this subsection.

One approach to performing parametric sensitivity analyses is local sensitivity analysis, which measures the impact of a parameter on model output in a portion of the output domain. Local sensitivity analysis is used to assess the effect of small perturbations in the value of a parameter around a particular value. Local sensitivity analysis is an OAT technique. OAT techniques analyze the effect of one parameter on a particular function by moving one input variable while keeping others at their baseline or nominal values. They explore changes in a parameter around a particular value, especially when there are many parameters. The amount of insight provided is limited since it does not account for the simultaneous variation of input variables. This means that the OAT approach cannot detect the presence of interactions between input variables. The OAT approach can determine only local sensitivity and not global sensitivity.

Another relatively straightforward technique is to generate scatter plots of the output variable against individual input variables after randomly sampling the input distributions. The advantage of this approach is that it can deal with a set of arbitrarily placed data points and gives a simple visual indication of sensitivity. Quantitative measures can also be drawn by measuring the correlations or by estimating variance-based measures by nonlinear regression.

Variance-based sensitivity analysis is a form of global sensitivity analysis (i.e., representing the impact of a parameter across its entire range of values). It is a class of probabilistic approaches that quantify the input and output uncertainties as probability distributions and break down the output variance into parts attributable to input variables and combinations of variables. The sensitivity of the output to an input variable is therefore measured by the amount of variance in the output caused by the input. For example, if a model has two inputs and one output, 60 percent of the output variance could be caused by the variance in the first input, 25 percent by the variance in the second, and 15 percent due to interactions between the two. These percentages are considered measures of sensitivity. Variance-based methods analyze the input space accounting for interactions and nonlinear responses but are meaningful only when the input factors are independent from one another.

Regression analysis is best used when the model response is linear, which can be confirmed if the coefficient of determination is large. The technique attempts to fit a linear regression to the model response and uses standardized regression coefficients as a measure of sensitivity. Regression methods are based on the correlation between the model results and the independent parameters under the assumption that they are related linearly. Stepwise multiple linear regression highlights those significant input parameters according to how much each input parameter reduces the residual sum of squares. The advantages of regression analysis are low costs and relative simplicity.

Nonparametric sensitivity analyses include the Kolmogorov-Smirnov (K-S) test, the Sign test, and the Wilcoxon Rank Sum test. Nonparametric tests are statistical but differ from regression tests because they do not require the assumption that the data have normal distributions.

The component sensitivity analysis method looks at the whole performing component, or nonperforming component, of a system at once. Component sensitivity analysis allows the exploration of the resiliency of components that would not be evident unless the other components have failed or degraded. Component failure or degradation is simulated by changing input parameters to degrade the performance severely either one component at a time or in combination with other components allowing interactions as well as redundancy in the system to become evident as selected components are forced to fail or degrade.

The strategic partitioning of assumption ranges and consequences method is based on concepts of PRA that aim to extract information on what sets of model parameter values taken together could produce substantially increased (or reduced) doses. The method also determines whether system features, working alone or in combination with other features, maintain their performance even under conditions that are highly unlikely.

Distributional sensitivity analysis can be used to determine the effect of estimation of parameter distributions on the performance results. In this technique, the input distributions can be changed either by shifting the mean of a distribution by a predetermined percentage or changing the shape of the distribution while keeping the minimum and maximum fixed. The distributional analyses can reveal choices of distribution functions that can affect the results significantly.

3.0 PERFORMANCE ASSESSMENT

A performance assessment is a type of risk analysis that addresses (1) what can happen, (2) how likely it is to happen, and (3) what are the resulting impacts (Eisenberg et al., 1999). These impacts can then be compared to the performance objective in 10 CFR 61.41 (radiological protection of the general public). Section 1.1.5.1 discusses the requirements for a performance assessment, set forth in 10 CFR 61.13(a).

This section describes acceptable approaches for conducting a performance assessment to demonstrate that the performance objectives specified at 10 CFR 61.41(a) and (b) would be met. This guidance supplements, rather than replaces, other NRC guidance on acceptable approaches for complying with the requirements specified in 10 CFR Part 61 (see Section 1.2.1). Specifically, this guidance supplements the approach discussed in NUREG-1573. The approaches described in NUREG-1854 (i.e., performance assessment for waste incidental to reprocessing waste determinations) for demonstrating compliance with 10 CFR Part 61 performance objectives may also be applicable to the development of a performance assessment for a land disposal facility for LLW. Table 10-5 provides a crosswalk of various technical topics to other relevant NRC guidance documents.

The following sections provide guidance related to the process for developing a site-specific performance assessment, including technical analyses information that a licensee should provide and that a reviewer should evaluate.

3.1 Performance Assessment Approach

The following five elements are essential to a performance assessment for an LLW disposal site:

(1) a description of the site and engineered system

(2) scenario development or an understanding of FEPs that affect, and will affect, the disposal system

(3) a conceptual model of the site including a description of processes controlling the movement of radionuclides from the LLW disposal units to the environment beyond the disposal units

(4) a computation of doses to members of the public

(5) an evaluation of uncertainties

The methods of performance assessment should be matched to the complexity of the problem. Deterministic, bounding analyses can be used for simple evaluations; however, probabilistic analyses may be more appropriate for evaluating the disposal of long-lived waste at LLW disposal sites, to account for uncertainties over long timeframes.

Many FEPs can influence the ability of a waste disposal facility to limit releases of radioactivity to the environment. While considering the uncertainties, a licensee should complete a

performance assessment that identifies the FEPs that might affect the disposal system, examines the effects of these FEPs on the performance of the disposal system, and estimates the annual dose to any member of the public caused by relevant FEPs. Section 2.5 provides guidance on the FEPs identification and screening process, including the development of the central scenario and plausible alternative site scenarios.

Disposal system behavior is characterized by the disposal facility design, the characteristics of the waste, the geologic and environmental characteristics of the disposal site, and processes and events that influence the features. The performance assessment identifies the specific characteristics of the disposal site (e.g., hydrology, meteorology, geochemistry, biology, geomorphology); degradation, deterioration, or alteration processes of the engineered barriers (including the wasteform and container); and interactions between the site characteristics and engineered barriers that might affect the performance of the disposal facility. The performance assessment examines the effects of these processes and interactions on the ability of the disposal facility to limit waste releases to the environment that could cause an annual dose to a member of the public.

The performance assessment may be conducted iteratively and is meant to be a tool for both the licensee and the regulator to use in assessing whether the disposal facility meets the 10 CFR 61.41 performance objective. During the design and licensing of a disposal site, assumptions may be made, based on expected waste volumes and compositions, about the possible final inventory of a site or a specific disposal unit within a site. As operations occur, these assumptions should be updated periodically with actual waste volumes and revised information on future waste to be received (see Chapter 9.0). The results of the performance assessment can then be used to evaluate whether reasonable assurance remains that the disposal unit or site will continue to meet the performance objectives. If the performance assessment shows that meeting the performance objective is uncertain or unlikely, then the licensee should consider (1) collecting additional data and revising the modeling, (2) modifying the facility, or (3) reducing future waste volumes or specific radionuclide amounts or concentrations (i.e., through setting “allowable” limits; see Chapter 8.0). The decisions on what actions to take should involve the site operator, the appropriate regulator(s), and other stakeholders.

3.1.1 Example of a Performance Assessment Approach

An example of an acceptable approach for conducting a performance assessment and demonstrating that the requirements in 10 CFR 61.41 would be met is outlined below. The approach is divided into 12 steps, as shown on Figure 3-1:

- Step 1: Conduct initial evaluation of information needed to describe the LLW disposal system environment — Formulate the context of the assessment. Describe the disposal system and the environment of the site.

- Step 2: Describe plausible evolutions (site scenarios) of the disposal site — Identify and consider credible factors or processes that could contribute to affecting a radionuclide release, including changes to the disposal site over time from natural processes and events. Construct a reasonably foreseeable site scenario (i.e., the central scenario) to evaluate for compliance with the performance objective (in Step 5) and less likely, but plausible alternative site scenarios for evaluating defense-in-depth (in Step 7).

- Step 3: Describe the conceptual model and the associated parameter values or distributions — Develop a site-specific conceptual model based on important disposal site features and processes. If model uncertainty is significant, the performance assessment may need to include one or more alternative conceptual models.

- Step 4: Formulate mathematical model(s) and select code(s) — Formulate mathematical representations of the conceptual model(s), using appropriate documentation and QA/QC. Numerical model development incorporates the mathematical representations and produces computational model(s) with which to perform simulations.

- Step 5: Conduct modeling — Describe receptor behaviors and activities (e.g., farming, gardening, fishing) near the disposal site in the central scenario and, if need be, in an alternative site scenario. Perform pathway analysis and dose assessment (e.g., potential dose to members of the public). Sections 3.4 and 4.2 contain further discussion. Much of the approach to development of receptor scenarios used in Section 4.2 can be applied in the performance assessment.

- Step 6: Perform sensitivity and uncertainty analysis — Evaluate which model(s), assumptions, and combinations of parameters are most significant with respect to the resulting doses. Evaluate the site scenario, model, and parameter uncertainties.

- Step 7: Demonstrate defense-in-depth — Describe defense-in-depth protections which include multiple, independent, and where possible redundant layers of defense using the results of the performance assessment. Chapter 7.0 includes further discussion.

- Step 8: Evaluate disposal site adequacy — Compare the results to the performance objective in 10 CFR 61.41 (radiological protection of the general public). If the performance objective has not been met, proceed to Step 9.

- Step 9: Reevaluate data and assumptions — Determine what information and data are needed to reduce uncertainty and demonstrate regulations are met. The conceptual model may need to be revised.

- Step 10: Collect new information or change the design — Gather needed information such as site characterization data and modeling studies or change the facility design. Redo Steps 2 through 8.

- Step 11: Update assumptions — Recalculate site performance using updated data and assumptions. Steps 9 through 11 can be iterated as needed until it can be demonstrated that regulations will be met.

- Step 12: Final determination — Make a final determination that regulations are met. If the licensee cannot demonstrate that 10 CFR 61.41 would be met at the selected site, it may choose to reject the site as a potential disposal option or place limitations on the inventory that can be accepted for disposal.

NUREG-1573, Section 3.1, includes additional guidance on an example performance assessment approach. Chapter 2.0 of this document provides additional guidance on Steps 1 through 4 above.

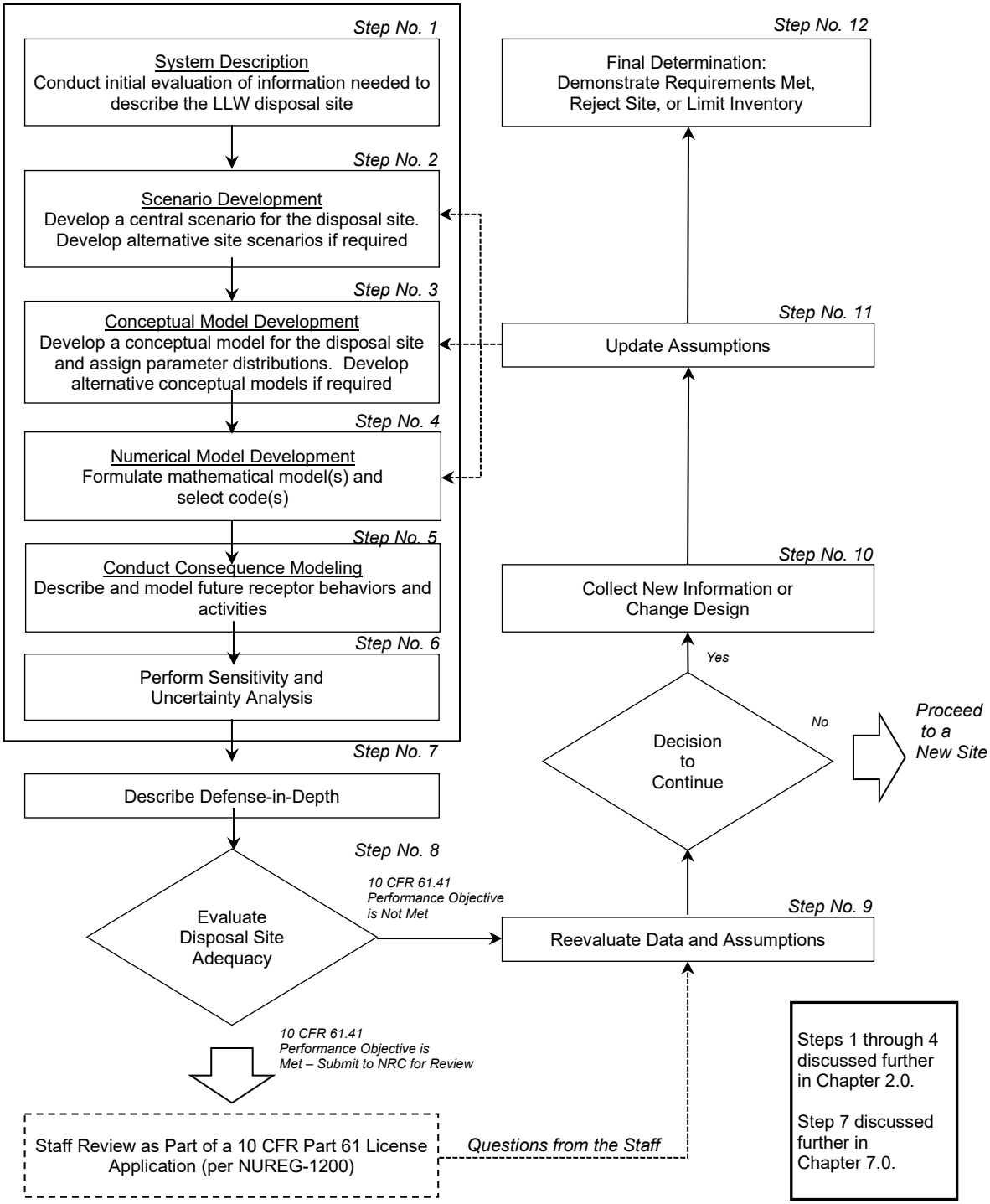


Figure 3-1 Example of a Performance Assessment Process

3.1.2 Role of the Performance Assessment

To obtain a license to receive, possess, and dispose of LLW, applicants should use the performance assessment to demonstrate, with reasonable assurance, that the 10 CFR 61.41 performance objective for protection of the general public will be met. The results of a performance assessment can be used to support the identification of where defense-in-depth protections enhance the resiliency of the facility in complying with the performance objective specified at 10 CFR 61.41. Applicants may use the results of the performance assessment to quantify barrier capabilities and their associated uncertainties to understand the contribution of the barriers to safety and defense-in-depth. Chapter 7.0 of this document provides additional guidance on using the results of the performance assessment to demonstrate that an LLW disposal facility includes defense-in-depth protections.

During construction, operation, and postclosure periods, the performance assessment can have an important role in demonstrating that the performance objectives continue to be met. The performance assessment is a projection into the future to provide the bases for the decision to proceed.

The other steps in the operation and closure process can be used to verify the basis for the initial decision. For example, 10 CFR 61.53, “Environmental monitoring,” requires a licensee to perform environmental monitoring during the construction, operation, and post-operational periods. Similarly, 10 CFR 61.28, “Contents of application for closure,” requires that a site closure application must include specific details of the disposal site closure plan included as part of the license application submitted under 10 CFR 61.12(g). The site closure application must also include the results of tests, experiments, or any other analyses relating to backfill of excavated areas, closure and sealing, waste migration and interaction with emplacement media, or any other tests, experiments, or analysis pertinent to the long-term containment of emplaced waste within the disposal site, including revised analyses for 10 CFR 61.13, using the details of the submitted site closure plan and waste inventory. Site closure may be authorized only if the final site closure plan provides reasonable assurance of the long-term safety of the facility.

The performance assessment can be used to address these requirements by updating the model developed for the initial license application with the new information from these monitoring programs. These new site data might validate or refute the key parameters or model assumptions used in the initial performance assessment. Section 9.3 provides information on mitigation, which might be necessary to continue meeting 10 CFR Part 61 performance objectives. Section 9.2 discusses the role of performance assessment in performance confirmation. Additional discussion of general technical elements as they relate to performance assessment appears in Chapter 2.0.

The following sections are a continuation of the discussion presented in Section 3.3 of NUREG-1573 but emphasize analyses for waste containing long-lived radionuclides. Licensees can use the guidance contained in this section to support demonstration that 10 CFR 61.41(a) performance objective is met and in some cases to support demonstration that 10 CFR 61.41(b) performance objective is met. Section 3.2 discusses radionuclide source term modeling and release of radionuclides from the disposal units. Section 3.3 discusses radionuclide transport through the environment outside the disposal units. Section 3.4 discusses modeling of the biosphere.

3.2 Source Term

The objective of source term modeling is to calculate radionuclide releases from the disposal units over time and space. Licensees can use the calculated release rates as input for transport models (see Section 3.3) that estimate offsite releases for the facility. The source term includes the inventory, physical and chemical characteristics, and other properties of the waste used to estimate release rates. The inventory of waste is the physical amount of material and quantity of radioactivity contained in the waste. Releases generally occur by advective or diffusive mechanisms, although direct release mechanisms may be possible (e.g., biointrusion, erosion). Release rates are a function of the conditions of the environment immediately surrounding the waste (i.e., the near-field environment). The near-field environment may have hydrological and chemical conditions that differ from the natural system in which the waste disposal facility is located. Additional releases to the aqueous phase and, for certain radionuclides (e.g., C-14, hydrogen (H)-3, radon (Rn)-222), to the gaseous phase can also occur.

Licensees may simulate many intermediate processes to estimate release rates. Release rates can be affected by the performance of engineered barriers (e.g., waste containers) and the wasteforms, the chemical properties of the disposal system, and the interaction of the disposal system with the natural environment. Licensees should identify these processes using the guidance in Section 2.5 for FEP analysis. Some disposal facilities will require detailed consideration of these processes and conditions, whereas simplified analyses may be justified for other sites and disposal facilities. Licensees should carefully consider the source term models for those disposal sites where (1) significant credit is taken for some aspect of the source-term modeling (e.g., low *solubility limits*), (2) attributes are relied upon for defense-in-depth protections, (3) a higher long-term hazard exists, or (4) there is limited model support. Facilities for which a simpler analysis may be acceptable include those where a licensee can show that a simple analysis is clearly conservative or where a licensee provides a simple model that is well-supported by multiple lines of evidence demonstrating that the model estimates are representative of or bounded by field data.

A reviewer may want to use the following factors when risk-informing the review effort: hazard, credit, and support. For example, a disposal site may have a hazard that is large and persistent, the licensee may have taken significant credit in its technical analyses for release rate modeling, and the licensee has limited information available to support its release rate modeling. In this case, the reviewer should apply additional resources to the review effort compared to a case in which a disposal site presents a small hazard, and the licensee develops well-supported technical analyses.

This section of the guidance focuses on the assumptions, data, and models (conceptual and mathematical) that a licensee may use to develop the source term abstraction for the performance assessment. Disposal sites and proposed disposal practices may vary significantly; therefore, source term abstractions are usually developed on a site-specific basis. A licensee should develop the source term abstraction to do the following:

- Include the effects of degradation processes on the performance of the wasteform and the engineered barriers.

- Consider the physiochemical processes associated with the partitioning of radionuclides between the waste and the physical phases in the disposal unit.

- Include temporal and spatial variation in the inventory, degradation processes, and physicochemical processes that are significant to performance.

- Adequately represent the features and processes of the source term significant to disposal system performance.

- Simulate the behavior of the source term to the extent that disposal system performance is adequately represented.

The source term model should be integrated with other models, such as climate, infiltration, and radionuclide transport, over both space and time. Guidance on modeling climate and infiltration appears in Section 3.3.3 of NUREG-1573 and Section 4.3.1 of NUREG-1854. Section 3.3.7 of NUREG-1573 and Section 4.3.5 of NUREG-1854 present guidance on radionuclide transport, biosphere characteristics, and dose modeling.

Sections 2.2 and 2.7 of this document describe general elements of technical analyses that are applicable to the review of a source term abstraction for performance assessment. Since this guidance supplements existing guidance, licensees and reviewers may also consult NUREG-1573, Section 3.3.5, for further guidance on source term modeling. Also, NUREG-1854, Section 4.3.3, may assist reviewers when considering acceptable source term modeling that may apply to a performance assessment for an LLW disposal facility.

The following sections provide specific guidance on source term modeling. Section 3.2.1 describes approaches for developing the inventory representation. Section 3.2.2 discusses guidance related to representing the chemical environment in the performance assessment. Sections 3.2.3 and 3.2.4 discuss guidance on representing the waste containers and wasteforms, respectively, and the effect of degradation processes on their longevity. Sections 3.2.5, 3.2.6, and 3.2.7 describe approaches for modeling aqueous, gaseous, and direct releases. Table 10-6 provides additional references relevant to wasteforms and waste release.

3.2.1 Inventory

The inventory representation provides the *radionuclide inventory* for which release rates are estimated with source term calculations. Sections 3.3.5.1, 3.3.5.2, and 3.3.5.7 of NUREG-1573 and Sections 3.1 and 4.3.3.1.1 of NUREG-1854, as applicable to LLW disposal facilities, provide additional guidance on recommended approaches for developing and reviewing waste inventory to support performance assessment modeling.

The radiological inventory to be disposed is a key input to the performance assessment. In addition, the physical and chemical characteristics of the waste can determine release rates. Physical characteristics can affect the stability of the waste and of the disposal facility (see Chapter 5.0). Chemical characteristics can affect the retention of radionuclides within the disposal facility. Chelating or complexing agents can significantly increase the mobility and effective solubility of the radionuclides in the disposal environment.

A licensee should develop an inventory description that provides the total activity (by radionuclide), concentrations, and physical and chemical forms of the waste to be disposed, including spatial configurations. The inventory description should include the radiological inventory at the time of disposal as well as the projection of inventory over the compliance and performance periods to account for decay and ingrowth.

Although Tables 1 and 2 of 10 CFR 61.55 list radionuclides for the purposes of waste classification, the inventory that a licensee evaluates in the performance assessment may be more extensive. The performance assessment inventory should not be limited to the radionuclides in Tables 1 and 2 unless justified. For example, Cl-36 and Np-237 are not listed in Tables 1 and 2 but are present in some LLW streams and are long-lived and mobile in the environment. Licensees should provide estimates of radionuclides that are not listed in these tables as well as a basis for the projected inventory. Licensees may use indirect methods of estimating the inventory of some radionuclides when direct measurement methods are not available, are not technically feasible, or are cost prohibitive. Indirect methods of estimating inventory have additional uncertainty that should be reflected in inventory estimates. When using indirect methods to estimate the inventory, the licensee should describe where the waste has been generated and what isotopes are used or created in the associated processes.

Different waste streams may have significantly different characteristics. However, it may be useful for a licensee to compare the projected inventory to the inventory of other LLW disposal facilities. The similarity or dissimilarity of the radiological inventories can help a reviewer to identify radionuclides that may dictate the need for a more detailed review of the inventory estimates. In some cases, specific radionuclides may be reported as less than a particular value because the radionuclide was below the lower detection limit of the characterization technique. It is acceptable to assign inventory based on the lower detection limit for initial screening. However, if the initial screening identifies that specific radionuclide inventory as significant, then additional characterization (either direct or indirect) may be necessary to determine the actual inventory with greater confidence. It is not acceptable to assume a value of zero for the inventory of radionuclides that are less than the lower detection limit unless there is adequate justification that the radionuclide is not present in the waste.

3.2.2 Chemical Environment

Estimation of the chemistry of the environment within the disposal units may be needed for source term modeling. The chemical environment may have impacts on, and be affected by, the lifetime of engineered barriers and waste containers, releases of radionuclides from the wasteforms, and mobility within the disposal units. The chemical environment in the disposal site is likely to be dynamic unless the site has been engineered to provide a more stable chemical environment. The chemistry of the disposal units would likely have an impact on solubility limits or retardation of radionuclides that are common parameters in modeling release rates from the source term and through the environment. Most chemical environments of near-surface disposal facilities are likely to be oxidizing. Geochemical considerations may include solid composition, pH, buffering capacity, reduction-oxidation potential, partitioning processes, and the presence of colloidal particles and ligands.

Licensees are encouraged to collect site-specific information to justify parameters that would be expected to significantly enhance or retard the release and migration of radionuclides in the performance assessment modeling or that are relied upon for defense-in-depth protections. Licensees may use sensitivity analyses to identify the importance of geochemical parameters, either to determine whether site-specific data should be obtained for significant parameters, or whether the range of values used is sufficient for parameters that do not have a significant effect. Section 2.2.3 presents guidance on the treatment of uncertainty. It may also be appropriate to use other means to justify parameters, such as values from available literature, relevant geochemical process modeling (e.g., MINTEQA2 (EPA, 1991); EQ3/6 (Daveler and

Wolery, 1992; Wolery, 1992a; Wolery, 1992b; Wolery and Daveler, 1992); PHREEQC (USGS, 1999); Geochemists Workbench® (Bethke and Yeakel, 2009a–d)),[1] or expert judgment and expert elicitation (see Section 2.2.4.1). Licensees should carefully select literature values for significant parameters and transparently describe the similarities and discrepancies between the site conditions they are proposed to represent and the conditions from which the literature values were derived. A reviewer should determine whether the literature values were developed under appropriately representative conditions. If the conditions under which the literature values were derived are not known, then the data should not be used.

The representation of the chemical environment should consider the uncertainty and variability in developing geochemical parameters. Spatial and temporal variability in the chemical environment of a site can occur. Spatial variability may arise from heterogeneities in wasteforms and other materials introduced to the disposal units (e.g., backfill materials, cementitious materials). Temporal variability may arise from the degradation of engineered barriers, waste containers, and wasteforms as well as the evolution of the disposal site environment to future conditions. Licensees should consider the evolution of the chemical environment resulting from degradation of the engineered barriers and natural processes such as climate change. A licensee should document its evaluation of spatial and temporal variability in the chemical environment. The documentation should provide the technical basis for the representation. For example, if the performance assessment modeling used a constant environment, the licensee should demonstrate why that environment was representative or conservative with respect to the projected future chemical environment.

Section 3.3.5.6 of NUREG-1573 presents guidance on considerations and issues associated with developing models for the chemical environment of the disposal units. Section 4.3.3.1.4 of NUREG-1854 provides guidance on reviewing a chemical environment representation. Additional considerations may also be found in Section 4.2.1.3.3 of NUREG-1804, Revision 2, "Yucca Mountain Review Plan," issued July 2003 (NRC, 2003c).

3.2.3 Waste Container

Waste container modeling describes the waste containers and estimates their longevity for inclusion in source term release modeling. In the past, licensees have generally not taken much credit for the performance of waste containers in their technical analyses; however, they are not prohibited from doing so. The waste container modeling, if used, should consider degradation processes (e.g., corrosion, mechanical degradation) for the various waste containers and should be consistent with the evolution of the chemical environment (see Section 3.2.2). The performance of the waste containers may affect the degradation of wasteforms and is important for estimating radionuclide release in the source term models. Section 3.3.5.4 of NUREG-1573 includes guidance on evaluating the performance of waste containers. Waste containers are a component of an engineered barrier system. Therefore, reviewers may want to consult applicable guidance on reviewing engineered barriers in Section 3.3.4 of NUREG-1573 and Section 4.3.2 of NUREG-1854. Sections 4.2.1.3.1 and 4.2.1.3.2 of NUREG-1804 present additional considerations for the performance of engineered barriers with respect to the waste containers.

[1] The NRC does not endorse or recommend specific commercial codes or models for geochemical modeling. Licensees are free to select and justify their code or model for their application. Regardless of the code selected, it is important that it meet all applicable QA requirements.

Metallic containers will undergo corrosion over time. Most metallic containers used in LLW disposal are carbon steel; however, other types of container materials may be used. If a licensee takes credit for waste containers in the performance assessment, it should provide an analysis to justify the performance of the containers. The analysis should look at the degradation processes (e.g., general corrosion, stress-corrosion cracking, localized corrosion, galvanic corrosion, radiolysis) and the disposal environment to estimate future performance. The analysis should consider the initial condition of the containers (e.g., corrosion before disposal). In humid disposal environments, failure of carbon steel containers due to general corrosion is expected to be relatively quick, such that the analyses commonly take little credit for performance (e.g., delay in releases). In arid environments, licensees may take more credit for container performance, if supported by the degradation analysis, which will examine the materials, environmental conditions, and degradation mechanisms. Licensees should support engineering analysis of container performance with measurements in relevant environmental conditions. In addition, licensees should provide real-world engineering analogs. The engineering analogs should be for similar exposure conditions and should not include maintenance since the disposal containers will not be maintained after disposal. Section 2.2.4 provides additional information with respect to developing model support, including the use of analogs.

Analysis of container failure should consider variability, as leakage from containers may occur as soon as the first perforation happens. The state of corrosion that defines “failure” will vary for different release mechanisms. A perforated waste container may still provide a significant barrier to mass transfer processes for some time. However, when the corroded area exceeds approximately 10 percent, the container can be considered to be failed from an advective release standpoint due to shedding of water from intact areas to failed areas.

Containers made of other (i.e., non-metallic) materials will experience degradation consistent with the material type. Reviewers should evaluate proposed container lifetimes for non-metallic materials on a case-by-case basis. In general, the licensee’s analysis should provide the following:

- an assessment of the relevant degradation mechanisms
- a description of the environmental conditions including variability and uncertainty
- analysis of the expected performance
- an assessment of potential unexpected performance and the likelihood of the unexpected performance
- data from laboratory, field, and/or analog observations that support the estimated performance

3.2.4 Wasteform and Waste Type

The wasteform modeling representation is used to describe wasteform performance, including variability. Wasteform performance is used in source term models. The modeling should consider degradation processes for the various wasteforms to estimate their performance consistent with the evolution of the chemical environment in the disposal units

(see Section 3.2.2). The performance of the wasteform is important for estimating radionuclide release in the source term models.

The branch technical position (BTP) "Technical Position on Waste Form," Revision 1, dated January 24, 1991 (NRC, 1991b), provides recommendations and guidance on acceptable methods to demonstrate that waste stability requirements have been met. Appendix A to that BTP provides methods to ensure the suitability of cement-stabilized wasteforms. Brookins (1984) describes some wasteforms and their properties. Every wasteform is different and can have different characteristics. However, the main elements in Appendix A to the BTP on wasteform test methods apply to both cement-based and other wasteforms.

These elements include the following:

- qualification testing
- quality of sample preparation
- consideration of variability
- waste characterization
- short- and long-term specimens including surveillance specimens

It is important for licensees and reviewers to note that, from a structural standpoint, waste stability does not necessarily translate into acceptable performance for waste release and leaching. Depending on the concentration of radionuclides in the waste, the fractional release rate of a radionuclide necessary to demonstrate that the performance objectives would be met may be lower than that necessary to demonstrate waste stability.

Radionuclide release from wasteforms can be complex, with numerous processes controlling the mobility and concentrations of radionuclides. Processes such as complexation reactions, acid-base reactions, oxidation-reduction reactions, dissolution and precipitation reactions, sorption and ion-exchange reactions, biodegradation of organic matter, and radioactive decay and ingrowth can impact waste release rates. A variety of test methods and modeling approaches have been developed to characterize the release of contaminants. The following guidance is developed from information in NUREG/CR-7025, "Radionuclide Release from Slag and Concrete Waste Materials, Part 1: Conceptual Models of Leaching from Complex Materials and Laboratory Test Methods," issued December 2010 (NRC, 2010a). This report identifies the following key lessons with respect to waste release:

- Test results can be misinterpreted if testing artifacts are not considered (e.g., interval for solution replacement, flow rates, failure to reach steady state, modeling of inappropriate processes).

- A single test method may not identify mechanisms controlling release.

- It is necessary to identify which processes control release under the conditions of interest (i.e., the test conditions should be representative of expected conditions, including variability).

- Laboratory test results need to be translated to long-term behavior (i.e., requires appropriate scaling, which may include multiple variables).

The objective of laboratory testing is for a licensee to identify the process (or processes) that will control the release of radionuclides over long periods of time, collect data to parameterize models that quantify release, and then integrate the waste release model into a performance assessment.

Test methods can be categorized as static, semi-static (where part or all the solution is exchanged during the test), or dynamic (where solution flows continuously). Different test methods can yield different results; licensees should carefully interpret the test results to avoid misinterpretation or mischaracterization of the release mechanism. For example, the leaching method in American National Standards Institute (ANSI)/American Nuclear Society (ANS)-16.1-2003, "Measurement of the Leachability of Solidified Low-Level Radioactive Wastes by a Short-Term Test Procedure," reapproved 2008 (ANS, 2008), is a solution-exchange test to characterize diffusion-controlled release. The test results are sensitive to the sampling intervals that are used to measure the extent of component releases. While the test is useful for screening, it is not recommended for modeling long-term waste degradation. Methods such as American Society for Testing and Materials (ASTM) C1308-08, "Standard Test Method for Accelerated Leach Test for Diffusive Releases from Solidified Waste and a Computer Program to Model Diffusive, Fractional Leaching from Cylindrical Waste Forms" (ASTM, 2009), which uses constant exchange intervals, may be more useful in determining whether waste release is diffusion controlled or affinity controlled. Column tests are useful in simulating the infiltration of water through the subsurface to waste and the resulting dissolution of contacted materials. The measured dissolution rate in column tests is a function of the flow rate of the solution and the reactive surface area that is contacted. Different column tests may need to be performed for different materials to account for the variable emplacement of those materials. A primary challenge of testing the release from wasteforms is the scaling of the laboratory results to the field, which may have much different flow rates and reactive surface areas than used in the laboratory. Additionally, the flow rates and reactive surface areas in the wasteform can vary over time. When some materials leach, the reactive surface area may increase or decrease from the leaching process. Release rates from some materials may be higher under conditions of wet-dry cycling as compared to non-dynamic conditions. The value of column tests (or tests such as field-scale lysimeters) is to confirm that the appropriate performance models are used to quantify the processes that control radionuclide release, especially when the tests have been calibrated to the system of interest.

Because of the complexity of waste release modeling, the NRC staff recommends that licensees complete a blind validation of the numerical model results to test the model's predictive capabilities. This is done by establishing a condition for which waste release measurements have not been completed and performing numerical simulations of the waste release that would be expected to occur under those conditions. This is followed by experimental or field measurements of the actual waste release that is compared to the numerical modeling. Licensees should establish criteria before the blind comparison to define acceptable agreement between the numerical modeling results and the experimental measurements.

Section 3.3.5.4 of NUREG-1573 includes additional information, such as guidance on characterizing wasteforms and types. Section 4.3.3.1.2 of NUREG-1854 presents guidance on wasteforms and their time-dependent degradation.

3.2.5 Aqueous Release

Models for aqueous release are used to estimate the rate of radionuclides released from water contacting the wasteforms. The four general types of aqueous radionuclide releases are (1) rinse release, (2) diffusional release, (3) dissolutional release, and (4) partitioning release. Rinse release refers to washing of radionuclides from the surface of a wasteform by infiltrating groundwater. Diffusional releases occur when diffusion limits radionuclide movement through a porous wasteform (e.g., a cement-stabilized wasteform). Diffusion can occur in media in the presence of a concentration gradient. Radionuclide releases resulting from corrosion of an activated metal or dissolution of glass wasteforms are examples of dissolutional release. Partitioning release results when radionuclide release is described by a characteristic partitioning parameter (e.g., sorption coefficient) that distributes the activity between phases in the system (e.g., between the solid wasteform and liquid water phases).

An aqueous release model is a function of the wasteforms, radionuclides, geochemical environment, and amount of water contacting the wasteforms. Licensees should integrate the aqueous release model with these related features. Methods to represent aqueous release have been incorporated into numerical models such as Disposal Unit Source Term — Multiple Species — Distributed Failure (DUSTMS-D) (Sullivan, 2006) and Breach, Leach, Transport, and Equilibrium Chemistry (BLT-EC) (NRC, 1995e). Though it is difficult to provide model support for the overall performance assessment model, licensees can support individual process models or abstractions (as discussed in Section 2.2.4). For example, licensees should provide support for estimated aqueous releases.

Licensees should consider comparing laboratory test results, such as leaching experiments, with numerical modeling results. If field data are available, they provide information that can be used for model support. For example, observed concentrations of radionuclides in leachate collection systems can be compared to modeled release rates. Section 4.3.3.2 of NUREG-1854 and Section 4.3.1.2.4 of NUREG-1804 provide additional guidance on potential considerations for development or review of aqueous release models.

3.2.6 Gaseous Release

Models for gaseous release estimate the rate of radionuclides released in the gas phase from the disposal units to the atmosphere above the disposal facility. A gaseous release model may also need to consider the generation of nonradioactive gases and their impact on the capabilities of the disposal facility to contain and isolate the radioactive waste. Examples of radionuclides that should be considered and evaluated for gaseous release include C-14, krypton (Kr)-85, Rn-222, H-3, and I-129. Gaseous releases are dependent on both generation of the gas and transport through fluids within the disposal units to the atmosphere. Discrete features such as fractures or a gap between materials can have a significant impact on gaseous release rates.

In Section 3.3.5.7.2 of NUREG-1573, the NRC staff recommends a screening approach to determine whether gaseous releases from the disposal facility might contribute significantly to dose. Licensees may apply this approach to determine whether more realistic release rates would be needed. Licensees may develop more realistic release rates using approaches described in NUREG-1573 or listed in the discussions below. Realistic release rate modeling should consider the specific processes affecting generation and transport of gaseous radionuclides or empirical models of gaseous release rates. In general, licensees are

encouraged to collect site-specific information to justify parameters that would be expected to significantly enhance or retard the release and migration of gaseous radionuclides should they significantly affect doses to a member of the public.

3.2.6.1 Gas Generation

Gases may be generated in an LLW disposal facility by or from the waste (e.g., Kr-85) or from various processes external or internal to the disposal site. Processes generating gases, many of which are discussed in Section 3.3.5.7.1 of NUREG-1573, may include (1) corrosion of metals in waste and its packaging, (2) microbial degradation of organic waste components, (3) radiolysis of waste, its packaging, or the surrounding backfill material, (4) volatilization of certain wastes (e.g., I-129), and (5) decay of radionuclides with primordial origin (e.g., uranium (U)-238 or thorium (Th)-232) to gaseous progeny (e.g., Rn-222). Rodwell et al. (2003) summarizes considerations for many of these processes as they relate to deep geological radioactive waste repositories. This information may be relevant for licensees to consider in a performance assessment for LLW disposal facilities. The evaluation of gas generation by a licensee should account for the nature of the waste, the radionuclides it contains, the waste packaging materials, and the chemical characteristics of the environment within the disposal units (e.g., saturation, groundwater composition, pH, Eh, and backfill materials) (NEA, 2001b).

3.2.6.2 Radon Emanation

Conservation of linear momentum in the alpha-decay process of high-specific-activity radionuclides can result in transport of radionuclides (i.e., alpha recoil). Typically, this process is not significant for the release and transport of radionuclides from an LLW disposal facility. However, alpha recoil can be significant for certain gaseous radionuclides (e.g., radon) associated with some long-lived waste streams. Alpha recoil can result in newly created radon atoms moving away from their original location. Some end up in the particles (media) where they were created, some end up in adjacent particles, and some end up in the pore space. The radon emanation coefficient is a dimensionless parameter that expresses the fraction of radon released to the pores (compared to the total volume). Guidance on the radon emanation coefficient is provided here due to the potential significance of this unique process and radon's presence in the decay chain for long-lived waste streams associated with uranium.

The radon emanation coefficient is strongly influenced by the liquid saturation of the medium (Nazaroff et al., 1988). As liquid saturations vary temporally and spatially, it is expected that radon emanation coefficients will also exhibit temporal and spatial variability. The values for radon emanation coefficients depend on the liquid saturation (or moisture content), the media, and the radon isotope.

Yu et al. (1993) compile radon emanation coefficients showing the large variability that can be observed; soil values ranged from 0.02 to 0.70. Nielsen and Rogers (1988) provide a probability plot of emanation coefficients for 56 soils. The values range from approximately 0.08 to 0.44 (with a mean of 0.22), consistent with the observed variability in the data of Yu et al (1993). Regulatory Guide 3.64, "Calculation of Radon Flux Attenuation by Earthen Uranium Mill Tailings Covers," issued June 1989 (NRC, 1989a), for calculation of radon fluxes from mill tailings covers recommends a default value of 0.35 for design, which would correspond to a 98th percentile value from the Nielson and Rogers (1988) data for soil. The challenge is that the data represent both intra- and inter-site variability. If site-specific measurements are not available, licensees should select conservative values from the compilations (as was done in

Regulatory Guide 3.64) because the inter-site variability should not be credited in a site-specific analysis. For long-term performance assessment calculations, it is important to recognize that the values selected for the analysis are representative of values anticipated for the evolution of site conditions. Average parameter values (spatial and temporal) can be used if it can be demonstrated that the variability is not important to estimating radon fluxes. However, because of the nonlinear processes governing radon release and transport, the variability in emanation coefficients and other parameters may significantly influence the long-term fluxes. For example, at a moderately humid site, contributions to the long-term average radon fluxes may be dominated by the periodic dry conditions.

3.2.6.3 Gas Transport

Once gases are released from the wasteforms and containers, they may migrate from the location where they are generated to the atmosphere above the disposal facility. If gaseous release is a significant pathway, licensees may need to perform gas transport modeling. Scanlon et al. (2000) summarize, in more detail, the mechanisms of soil gas transport in unsaturated systems, as are commonly encountered in the near surface, and mathematical models to simulate these mechanisms. Similar to aqueous migration in the environment, the transport of gas can also be described by advection and dispersion.

Advection of gaseous radionuclides from a disposal unit may result from pressure gradients (e.g., barometric pumping) leading to gas flows from areas of higher pressure to areas of lower pressure. Pressure gradients, particularly those driven by atmospheric conditions, can be variable, and releases may be dominated by episodic changes. Licensees should assess the impact of this variability, if significant, on releases of gaseous radionuclides. Advection is affected by the pressure gradient, gas characteristics (e.g., viscosity), and properties of the solid media (e.g., permeability) and generally dominates transport under a pressure gradient when the mean free path of gas molecules is much less than the pore radius and the particle radius of the solid media through which the gas is moving (Cunningham and Williams, 1980). Darcy's Law is typically used to mathematically model advective transport in gases. Depending on the dimensions of the pore space and pressure, a coupling of advective flow to diffusion occurs in gas transport as a result of the increasing importance of molecules-to-wall interactions (Bodvarsson et al., 2000).

Diffusive processes may include Fickian molecular diffusion as well as non-Fickian processes such as Knudsen and non-equimolar diffusion depending on the pressure gradients, dimensions of the pore space, and mean free paths of the molecular motion. Several mathematical models (e.g., Fick's Law, Stefan-Maxwell equations, Dusty Gas Model) exist for diffusive transport, depending on pressure gradient, permeability, and concentration conditions (Scanlon et al., 2000). Because of the challenges in mechanistically modeling diffusion through heterogeneous media, an empirical effective diffusivity of the gas in the geologic material through which the gas migrates is often used to account for the various characteristics. The effective diffusivity is sensitive to the availability of interconnected air space and thus to the total porosity, air-filled porosity, moisture content, tortuosity of pores and fractures, and soil structure. Various relationships have been hypothesized to estimate effective diffusion coefficients in soils (e.g., Buckingham, 1904; Penman, 1940; Millington, 1959; Millington and Quirk, 1960; Troeh et al., 1982; Nielson et al., 1984; Kristensen et al., 2010), but their predictive capabilities may be sensitive to actual site conditions (Moldrup et al., 2004; Allaire et al., 2008). Literature values for effective diffusion coefficients often employ varied definitions. For example, Culot et al. (1976) note that the literature contains various definitions of the effective diffusion coefficient for

gas diffusion of radon in concrete. Regulatory Guide 3.64 provides guidance on an acceptable method to assess long-term radon diffusion through earthen cover materials for uranium mill tailings sites and may also be applicable to LLW disposal. Licensees should demonstrate that parameters describing diffusion of gaseous radionuclides are adequately justified and account for actual site conditions including uncertainty and variability and the effect of the evolution of site conditions over time on the parameters.

Licensees that undertake more detailed modeling of gas phase transport from disposal units may need to consider the aforementioned processes in the models for gaseous release, depending on the site conditions. For instance, molecular diffusion may be the only mechanism to consider for equimolar gases under isobaric, isothermal conditions that diffuse in pores much larger than the mean free path of the gas molecules, whereas a Dusty Gas Model approach might be necessary for gas migration through low-permeability materials under a pressure gradient to account for coupled advective-diffusive processes. Important parameters of these models (e.g., permeability, saturation, pore space dimensions) should be well supported and, to the extent practicable, based on conditions that are representative of the site environment. In other words, site observations supporting parameter values should be made under expected conditions for the site and not biased toward conditions that are rare, extreme, or unlikely, particularly if the unlikely conditions lead to overly optimistic behavior of the disposal system. Uncertainty in the parameters may also be significant. For instance, permeability may span a broad range of values depending on the soil types (Nazaroff, 1992). Therefore, the analysis of uncertainty in significant parameters should consider the range of conditions expected for the disposal site. Partitioning of gaseous radionuclides between phases may also affect the transport of radionuclides that are generated in the gas phase. In addition, biological activity and isotopic dilution (i.e., dilution of a radioisotope with stable isotopes of the same element) may also have a significant impact on the transport of certain gaseous radionuclides (e.g., C-14) (Bracke and Müller, 2008).

Changes in cover properties that impact the hydraulic conductivity of cover materials may also affect the migration of Rn-222 through the cover. Licensees can design cover materials to inhibit radon migration, so that radon flux at the surface is reduced by radioactive decay during transport to the surface (NRC, 1984). However, preferential flow paths can develop in clay barriers in conventional covers (Albright et al., 2006a; Albright et al., 2006b). Preferential flow paths may lead to the advective transport of radon. Typically, radon migration is controlled by diffusion which is dependent on the inter-connected porosity and moisture content of the soil or clay radon barrier (NRC, 1989a). In addition to enhanced diffusive transport, advection may also contribute to migration of radon through these preferential pathways (NRC, 1984). NUREG/CR-3395, “Influence of Cover Defects on the Attenuation of Radon with Earthen Cover,” issued November 1983, (NRC, 1983a), states that models for radon migration in cracks are typically designed for simple geometries that do not account for all dynamic processes, resulting in a possible underestimation of radon flux. The recommendation from NUREG/CR-3395 is to use cover design and development methods to avoid the formation of defects (i.e., avoid cracks). Cover design considerations included selection of the cover soils (to avoid shrinkage cracks), vegetation (root formation), and burrowing animals.

3.2.7 Direct Release

Licensees may need to represent the direct release of radionuclides into the environment. Models for direct release estimate the rate of radionuclides released directly to the surface. Anthropogenic direct releases are considered in the inadvertent intruder assessment, which is

discussed in Chapter 4.0. Direct releases may be caused by processes such as plant uptake, bioturbation (e.g., burrowing animals), natural disruptive events (e.g., faulting), and geomorphological processes (e.g., erosion). For guidance on consideration of natural disruptive events and geomorphological processes, licensees and reviewers should consult Chapter 5.0 regarding the demonstration that the performance objective for site stability would be met. Direct release can impact the demonstration that 10 CFR 61.41 and 10 CFR 61.42 would be met. Licensees should integrate direct release calculations with other models as necessary, such as with climate and radionuclide transport. Additionally, direct release calculations should consider the long-term evolution of the site environment and its impact on the evolution of the site's geomorphology and ecology. For instance, future climate states may be wetter, leading to increased erosion rates or changes in vegetation or communities of burrowing animals. Licensees may use analog sites with climates similar to the expected future climate for the disposal facility to justify the progression of ecological communities at the disposal site.

3.2.8 Biota-Enhanced Release

"Biota enhanced release" is defined in this section as the direct and indirect processes that facilitate release of radioactivity from a disposal facility by biota. Damage of an engineered cover by burrowing animals is an example of an indirect effect of biota on performance of the LLW disposal facility. Biotic transport, as defined in Section 3.3.4, is the transport of radionuclides in the environment by biota after radioactivity is released.

Biota enhanced release is not usually a significant pathway in performance assessments. However, the importance of biotic release and subsequent human exposures depends on a variety of factors, including site location, design of the disposal facility, and the wasteforms. For example, plant uptake and the impact of burrowing animals are much more likely for shallow disposal than for deeper disposal. Therefore, a reviewer evaluating radionuclide release from a site should consider the impacts associated with biotic-enhanced release. Current LLW disposal facilities attempt to minimize the impacts of plants and animals by incorporating design features such as engineered cover systems and the use of concrete and steel in the construction of the vaults and waste containers. Waste conditioning may also significantly limit the availability of radioactivity for biota-enhanced release.

Section 5.9.1, "Background," of National Council on Radiation Protection and Measurements (NCRP) Report No. 152, "Performance Assessment of Near-Surface Facilities for Disposal of Low-Level Radioactive Waste," issued December 2005 (NCRP, 2005), lists numerous references of studies of biotic processes conducted at LLW disposal facilities. The long-term contributions of biotic enhanced release processes, including their impacts on engineered wasteforms and barriers, associated with recent LLW disposal facilities are uncertain. Although biotic-enhanced release has been observed for a wide variety of contaminant release problems, it is typically not a primary release mechanism.

A study conducted by McKenzie et al. (NRC, 1982d) identifies and discusses the mechanisms of biotic release and transport and human exposure associated with arid and moist sites. Based on this qualitative analysis, the authors of that study concluded that penetration of buried waste and enhanced release by burrowing animals and plant roots were potentially important mechanisms in the case of waste disposed in trenches. These studies demonstrate that biota have three general primary effects on an LLW disposal facility: biotic release, biotic transport (Section 3.3.4), and secondary transport. This section focuses on intrusion and indirect effects, which can enhance release through other mechanisms. Depending on the characteristics of the

disposal facility and its environmental settings, biota could influence the performance of a disposal system and lead to pathways for release of radioactivity and resultant human exposure.

Intrusion and active transport occur when biota penetrate the waste zone, causing the redistribution of waste material or contaminated soil and resulting in release from the disposal facility. By including land management within the scope of management activities, LLW disposal facilities can reduce the impact of biointrusion during the period of institutional control (e.g., limiting intrusive vegetation and minimizing the intrusion by animals). Licensees may want to consider observations of nearby biotic activity made during operations when revising the biotic intrusion assessment for the performance assessment analysis completed for site closure. A licensee should develop an assessment of projected biotic intrusion after the postclosure observation and maintenance period based on the estimated evolution of the disposal site. The projection should not include land management practices that will be used during the institutional control period to limit the effects of biotic intrusion on the site unless those practices are passively effective after the institutional control period. Many engineered cover designs employ biointrusion barriers. If the licensee provides a technical basis for the barrier's passive performance, it can take credit for biointrusion barriers in the performance assessment projections. Analog information may be useful in supporting predictions of the performance of biointrusion barriers.

A qualitative assessment of the biotic-enhanced release and its contributions to the dose may be sufficient depending on the site, its characteristics, and the characteristics of the waste disposed at the site. In some cases, biological processes can strongly influence the release of radioactivity from LLW disposal facilities because the near-surface environment is biologically active.

In addition to direct release (e.g., uptake of radionuclides by deeply rooted vegetation), biota can indirectly enhance release when plants and animals modify the buried waste or the design of the LLW disposal facility in such a way that there is an increased potential for radionuclide release and transport. The following are examples of these processes:

- Burrowing animals or root systems develop tunnels or conduits that can enhance the transport of groundwater or increase gaseous release (e.g., bioturbation of a clay layer).
- Plant roots have ligands that provide a source for radionuclides to bind, resulting in the formation of soluble radionuclide-organic complexes.
- Microbial degradation of the wasteforms leads to the enhanced generation of gases or creates waste materials that are more soluble and therefore more easily transported away from the site.
- Microbial-enhanced degradation of waste containers or concrete vaults in a disposal system occurs.
- A drainage layer is plugged in an engineered cover system.
- Pedogenic processes can modify the soil properties of an evapotranspiration cover system.

These examples are not meant to be an exhaustive list; other biological processes may affect the performance of the waste disposal facility. The licensee should provide a technical basis for its consideration of biological processes on the performance of the disposal facility, including the basis for excluding biological processes from the evaluation.

3.3 Radionuclide Transport

This section focuses on assumptions, data, and models (conceptual and mathematical) that a licensee may use to develop a radionuclide transport model for the performance assessment. A licensee may use radionuclide transport modeling to estimate the transport of radionuclides in environmental media (e.g., waste, soil, air) to receptor locations (i.e., human access locations) over time. Radionuclides released from LLW disposal facilities (see Section 3.2) can be transported through the environment by groundwater, surface water (including suspended sediments), air, and biota (e.g., rodents, insects). Radionuclide doses may also be linked directly to the media themselves, such as the consumption of contaminated well water, or indirectly between pathways, such as the transfer of radionuclides from the groundwater to the surface water and ultimately to the pathways comprising the food chain.

The licensee should identify and assess significant mechanisms of radionuclide transport. Section 2.5.3 contains guidance on identifying and screening FEPs, and Section 2.5.4 covers constructing site scenarios. For FEPs that are relevant, some features and processes of the central scenario or alternative site scenarios may require detailed representations in the performance assessment, whereas simplified representations may be justified for others. Radionuclide transport models may need greater review effort for those facilities for which licensees take credit for significant delay in radionuclide migration from the disposal facility to the receptor and those for which there is little model support. A simplified analysis may be acceptable if it is conservative or if the simple model is well supported by multiple lines of evidence, including field tests that demonstrate that the model and its parameters appropriately represent or bound site conditions. Likewise, a simplified analysis may be acceptable if the transport times are not significant in the context of the overall results of the performance assessment.

The complexity of most sites and proposed disposal practices usually results in radionuclide transport models being developed on a site-specific basis. The radionuclide transport modeling process should appropriately account for temporal and spatial variability in the environmental transport pathways resulting from natural heterogeneity in environmental media and the evolution of site conditions over the analysis timeframe. Variability in natural systems is often scale dependent. For instance, wind speeds may vary over a wide range of time scales (e.g., hourly, daily, annually). The variability considered should be consistent with the assessment context. In this case, the context is annual doses to an offsite receptor. It would be appropriate to consider how the scale-dependent variability may affect the outcome of the assessment through sensitivity analyses. If scale-dependent variability is found to be significant, a licensee may need to include short-term variability in the radionuclide transport modeling. For example, short-term variability may be considered when developing a distribution of wind speeds that could occur in any given year (in the above example), while longer term variability may be treated as variability in the mean annual wind speed. The approach a licensee selects to represent the uncertainty and variability resulting from natural heterogeneities should not bias the outcome of the assessment such that radiological doses to an offsite receptor would be significantly underestimated. Licensees should document analyses supporting their conclusion that the method selected to represent uncertainty and variability in radionuclide transport

abstractions does not result in biases that significantly underestimate radiological doses to an offsite receptor.

Licensees should ensure that the modeling of radionuclide transport is appropriately integrated with the source term releases (see Section 3.2). The radionuclide transport modeling should also be consistent with modeling of climate and infiltration, as well as the characteristics of the biosphere and the receptor (e.g., human access locations). Section 3.3.3 of NUREG-1573 and section 4.3.1 of NUREG-1854 contain guidance on modeling climate and infiltration. Guidance on biosphere characteristics and dose modeling appears in Section 3.4 of this document and in Section 3.3.7 of NUREG-1573 and Section 4.3.5 of NUREG-1854.

Section 2.2 of this document describes general technical considerations applicable to the development and review of a radionuclide transport model for an LLW disposal facility performance assessment. As mentioned previously, licensees and reviewers may also consult NUREG-1573, Section 3.3.6, for further guidance on transport modeling. Section 4.3.4 of NUREG-1854 may assist reviewers by providing technical considerations for acceptable radionuclide transport modeling. The following sections of this document provide specific guidance on aspects of radionuclide transport modeling in various environmental media.

3.3.1 Groundwater Transport

A licensee may perform groundwater transport modeling to assess the concentrations of radionuclides transported from LLW disposal facilities through the unsaturated and saturated zones. Groundwater transport is among the most likely processes for radionuclides to be mobilized from LLW disposal facilities. In addition to being a direct source of exposure to radionuclides, groundwater can transfer radionuclides to other exposure pathways that may ultimately result in doses to members of the public. These transfer processes include discharges to surface water through seeps and springs and vapor releases to the atmosphere.

A licensee may need to model the migration of radionuclides in groundwater and determine the contribution of the groundwater pathway to the total dose to the average member of the critical group. The complexity of hydrogeologic and geochemical conditions of the site may need to be reduced to form simplified models of groundwater flow and transport. A licensee should provide a technical basis to show that the reduction in complexity has not resulted in the loss of information necessary to demonstrate that the performance objectives of 10 CFR Part 61, Subpart C, will be met.

Development of a groundwater transport model should consider relevant FEPs associated with the site (Sections 2.5 and 2.6 present guidance on FEPs). Licensees should consider the characteristics of the wasteforms designed to limit releases of radionuclides, as well as the physical characteristics of the site (e.g., hydrogeology, faults, and groundwater flow boundaries). Physical and geochemical processes (e.g., advection, dispersion, diffusion, sorption, precipitation) are the primary phenomena included in groundwater transport models.

The transport of radionuclides in groundwater can be affected by many different physical phenomena associated with the hydrogeology of the system, as well as by the physical components (e.g., engineered barriers) associated with the facility. Licensees should consider information on the hydrogeologic characteristics of the site including the stratigraphy, the existence of preferential pathways (e.g., fractures), spatial variations in properties, and anthropogenic features that may affect groundwater flow. Processes such as diffusion and

mechanical dispersion should be considered. At some sites, other less likely phenomena may be significant, such as colloid-facilitated transport. In the case of the unsaturated zone, variations in properties and features such as thickness of unsaturated strata and hydraulic properties can impact water flow. The spatial variation of hydrologic properties of aquifers and aquitards, as well as geologic features (e.g., fractures) and engineered structures (e.g., slurry walls), can influence groundwater flow velocities, gradients, volumes, and estimates of recharge to and discharge from the aquifers associated with the saturated zone. In addition, licensees should consider differences associated with the unsaturated zone and saturated zone. Flow of contaminants in the unsaturated zone may be strongly influenced by discrete features, such as fractures or other heterogeneities. A licensee should provide its site-specific characterization data of the spatial variation in hydrologic properties and geologic features. If site-specific information is not available, the licensee should demonstrate how it incorporated and evaluated variability in properties and features in the radionuclide transport modeling.

A licensee should evaluate the chemical characteristics of the groundwater and host rocks including sediments. Chemical processes such as sorption, precipitation, ion exchange, and redox reactions can affect radionuclide transport in groundwater. NUREG-1573 and NUREG-1854 provide guidance for assessing groundwater transport. Additionally, the NCRP has published a summary of processes and parameters applicable to the development of groundwater transport models in the performance assessment for an LLW disposal facility (NCRP, 2005).

Table 3-1 summarizes site characteristics, physical processes, and geochemical processes that licensees should consider when developing groundwater transport models. Reviewers should use Table 3-1 to evaluate the scope of a licensee's radionuclide transport modeling.

Modeling groundwater transport can be challenging for a variety of reasons. The characteristics and features that introduce challenges include the following:

- complex interactions of the hydrologic, geologic, physical, and chemical processes associated with the system
- limited availability of site-specific data to describe the FEPs for a specific groundwater system
- natural heterogeneity of site characteristics associated with groundwater flow and transport and changes in characteristics over long times
- modeling artifacts introduced by numerical analysis techniques, such as numerical dispersion, which can bias results

A licensee should describe its radionuclide transport model to allow a reviewer to independently determine that it adequately addressed the challenges. Some disposal facilities may require careful consideration of these challenges in the groundwater transport model, whereas a simplified assessment of the significance may be justified for other sites and disposal practices. The groundwater transport models may need to consider these site-specific complexities for (1) facilities for which licensees take credit for significant delay in radionuclide migration by groundwater from the disposal facility to the receptor, (2) those that take significant credit for reductions in concentration during transport, (3) those relied upon for defense-in-depth, and (4) those for which there is limited model support.

Table 3-1 List of Parameters and Processes Associated with the Groundwater Transport Pathway

Hydrogeologic Characteristics
Stratigraphy Geologic structures (e.g., faults) Groundwater flow boundaries Zones of groundwater recharge and discharge Soil and rock characteristics (e.g., porosity, texture, mineralogy) Fractures and fast pathways (i.e., discrete features) Quantity and quality of groundwater
Physical Processes
Advection Dispersion Diffusion Deposition Radioactive decay and daughter ingrowth
Geochemical Processes or Characteristics
Composition of groundwater Geochemical environment (e.g., pH, Eh, organic content) Sorption/desorption Precipitation Complexation Resuspension Redox reactions Colloids
Design Features
Well depth Screen length Physical components (i.e., engineered barriers)

A simplified analysis may be acceptable if it is conservative or for which the simple model is supported by multiple lines of evidence, including field tests that demonstrate that the model and its parameters appropriately represent or bound site conditions. A simplified groundwater modeling approach may be sufficient to gain an adequate understanding of the FEPs associated with a specific system. More detailed groundwater flow and transport modeling, however, may be needed to identify important processes and assess the impacts of uncertainty and variability.

An important consideration in modeling groundwater flow and transport is the inherent spatial variability of structural, physical, and chemical properties of the environmental media and the uncertainty in characterizing the spatial domain. Heterogeneity in flow and transport processes can vary over many orders of magnitude from the molecule scale to the basin scale (Ledoux and deMarsily, 1997). Some properties exhibit scale-dependence (e.g., dispersivity) (Neuman, 1990). When warranted by the significance of the transport pathway, a licensee may need to include a representation of the variability in properties. However, it is challenging to

describe the heterogeneity due to the characterization that would be needed. Rubin (2003) summarizes some approaches (e.g., estimation, simulation) that licensees may consider for use in representing variability and assessing its impact on groundwater flow and transport.

Licensees should demonstrate that the method used to assess groundwater transport does not significantly bias the outcome. Emphasis should be placed on avoiding underestimation of projected radiological doses, though any form of significant bias should be avoided to the extent possible. A method to demonstrate this is to compare viable alternate conceptual models of the groundwater flow system that are consistent with the current understanding of the site.

For the performance assessment of an LLW disposal facility, the parameterization of processes should be made at the scale (spatial and temporal) over which groundwater transport is evaluated. However, measurements of flow and transport parameters on these large scales are often not available. Additionally, smaller scale measurements, such as from laboratory or field tests, can provide useful information to models, but the models can become intractable computationally. For instance, performance assessments often represent sorption of radionuclides on soils with a distribution coefficient (K_d). In a groundwater transport abstraction for a disposal facility, this coefficient may be representing the heterogeneities in the soil properties and groundwater chemistry over large spatial scales. In practice, distribution coefficients are often measured in the laboratory or field on a much smaller scale than what is modeled in the performance assessment and may not represent the actual heterogeneities in the groundwater pathway that the radionuclides encounter. Therefore, licensees should consider approaches to demonstrate the appropriateness of model parameters for the scale of concern for the performance assessment (e.g., upscaling, inverse modeling). The following two sections provide examples of specific issues for licensees and regulators to consider when implementing or evaluating the unsaturated and saturated zone models.

3.3.1.1 Modeling the Unsaturated Zone

This section highlights issues specific to the flow and transport of groundwater through the unsaturated zone. Section 4.3.4.1.3 of NUREG-1854 and NCRP Report No. 152 provide a general discussion of the issues that should be covered. Additional details regarding flow and transport in the unsaturated zone appear in other documents, including Bear (1979), Bear and Verruijt (1987), Domenico and Schwartz (1998), Evans et al. (2001), Faybishenko et al. (2000), Freeze and Cherry (1979), Hillel (1980), Looney and Falta (2000), NAS/NRC (2001), Nielsen et al. (1986), and Todd (1980). Documents such as NAS/NRC (1990) and NCRP (1984) discuss unsaturated flow and transport models. NRC-sponsored research related to the characterization and performance confirmation monitoring of the unsaturated zone is included in various references (Meyer et al., 1999; Rockhold, 1999; NRC, 1993b; NRC, 1999b; NRC, 1999c).

Modeling of flow and transport of radionuclides in the unsaturated zone provides a link between releases from an LLW disposal facility and the saturated zone. Inputs for a licensee to consider when developing unsaturated zone flow and transport models include infiltration and the release rate of radionuclides from the source; the resulting outputs, flow rate and radionuclide migration, are used as inputs when modeling saturated zone flow and transport.

Variations in soil characteristics and moisture content in the unsaturated zone, as well as the complex relationships between hydraulic head, moisture content, and hydraulic conductivity can make modeling these systems difficult. Site-specific conditions can further influence modeling

of the unsaturated zone. For example, at wetter sites the disposal facility generally lies within a few meters or tens of meters of an underlying aquifer. The flow and transport of groundwater and radionuclides over this short distance may not be significant when assessing the overall performance of the site. Modeling the unsaturated zone for these types of sites may require a simple approach, such as not taking any credit for delay or dilution associated with the transport between the LLW disposal facility and the aquifer. At arid sites, however, the aquifer may lay several tens or even hundreds of meters below the disposal facility footprint. The time required for radionuclides to reach the aquifer may allow for more physical and chemical interactions with the environment and significantly reduce the concentrations or the timing of when radionuclides reach the aquifer. As discussed in Section 2.2.4, model support is essential for providing a basis for a licensee's simulation results. Model support can be used to justify that the complexity (or lack thereof) in a licensee's model is appropriate. A licensee should demonstrate that (1) the complexity is necessary, (2) site-specific information is available to adequately characterize the complexity, and (3) model support is available to support the analysis results.

Arid sites may experience high evapotranspiration rates creating a situation in which there is no transport or very limited transport of radionuclides released from the disposal facility and the underlying aquifer. This is especially true for the disposal of inventories containing primarily short-lived radionuclides. Modeling this situation is typically more complex than a traditional flow and transport analysis because spatial and temporal variability can more significantly impact the results. Discrete features such as fractures or abandoned wells can limit the effectiveness of the unsaturated zone as a barrier. Completion of field tests and analysis of field observations are recommended to reduce the uncertainty associated with discrete features in the unsaturated zone. Tracer studies over a large area can be effective.

Many of the processes that govern transport of radionuclides in the unsaturated (vadose) zone (i.e., advection, dispersion, and sorption (retardation)) are essentially the same as those that govern transport in the saturated zone. However, since the pore spaces of the vadose zone are only partially filled, the effective hydraulic properties have a nonlinear dependence on soil moisture content. Hydrologic and moisture property relationships are commonly represented with moisture characteristics curves, which can be difficult to develop. Typically, there is considerable uncertainty in the moisture characteristic curves. In addition, the spatial variability of the moisture characteristic curves over the area of a disposal site may be unknown. A licensee should provide the technical basis for the moisture characteristic curves used or selected and demonstrate the sensitivity of the modeled results to the moisture characteristic curve(s).

Because of the nonlinear relationships, flow in the unsaturated zone can be strongly influenced by extreme conditions (e.g., a zone of discrete features such as fractures, wetter years, or seasonal variations). Therefore, it is necessary for a licensee to understand how hydraulic conductivity changes as a function of the amount of moisture and the pressure in the soil. For example, when the fluid pressure is less than the atmospheric pressure, the pressure head is negative allowing suction to occur (Freeze and Cherry, 1979). Additionally, the infiltration rate may vary with time as a result of the intermittent nature of precipitation. It may also vary with the depth of the unsaturated zone, resulting in lower water content closer to the surface and increasing with depth, although profiles of moisture content with depth are dependent on the properties of the materials present and the precipitation that enters the top of the column.

Given these complexities, a graded approach that starts with the simplest and most conservative representation is recommended for licensees. This approach starts with a

relatively simple representation of the system — assume homogeneous conditions and no time delay or dilution. Problems may arise with this simple approach for situations in which the unsaturated media are known to be highly structured or fractured, or if it is determined that there is significant time or distance to reach the saturated zone. For these cases, the modeling approach would need to be more complex. Some complexities may be beneficial to the estimated results whereas others are detrimental. In general, a licensee should only include beneficial complexity if that complexity can be supported. However, in an iterative modeling process, a licensee may need to add complexity to determine whether the complexity is significant. If so, the licensee may need to develop support for the representation or take a conservative approach.

3.3.1.2 Modeling the Saturated Zone

In a performance assessment for an LLW disposal facility, a licensee should consider the flow of water through underlying aquifers and the phenomena that influence transport of radionuclides in groundwater. A licensee should have an adequate understanding of the input flux of radionuclides from the unsaturated zone. For example, although an increase in flow velocity into the saturated zone would appear to yield higher concentrations of radionuclides being released to offsite locations or other exposure pathways due to decreased travel time, it also results in a greater volume of water available to dilute radionuclide concentrations entering the aquifer from the unsaturated zone. Thus, it is possible that the higher flow rate can result in a decrease in radionuclide concentrations. A licensee will need to have an understanding of the groundwater velocity field, radionuclide-specific release rates to the saturated zone, and the phenomena that influence the transport of radionuclides in groundwater (e.g., sorption, advection, diffusion, dispersion, and radioactive decay). Saturated-zone radionuclide concentrations are typically converted to two primary types of doses for receptor scenarios: the groundwater dose resulting from direct ingestion and the groundwater component of the dose resulting from ingestion of foodstuffs grown using contaminated water. Other pathways are possible and may be significant at specific sites, though they are less common (e.g., discharge of groundwater to a surface water body).

A velocity field provides a representation of the groundwater flow in the saturated zone. A velocity field, which is a function of location and time, is often derived using indirect, largely unmeasured, spatially variable, or unknown physical properties and boundary conditions that are interpreted subjectively. Licensees may develop the velocity field by using a site-specific model of the flow system that has been calibrated using site-specific data on hydraulic head. The hydraulic head can be obtained using monitoring wells and field experiments (e.g., slug tests). However, the derived velocity field is likely not going to be a unique representation, as multiple velocity fields will likely be consistent with the supporting information. Licensees should provide justification and support for their assumptions. It is often useful for a licensee to include the uncertainty associated with the calibration and inverse modeling to derive groundwater flow fields in the performance assessment groundwater flow model, thereby allowing the significance of the uncertainty to be directly assessed.

Licensees may describe groundwater flow in the saturated zone using a simplified flow equation, such as combining Darcy's Law with the principle of conservation of mass. This approach may be further simplified by assuming a steady-state flow rate and uniform recharge to the aquifer over the spatial domain being considered. However, uncertainties such as the presence of heterogeneities in the system, variable or unknown boundary conditions, and

scaling of data obtained from small-scale samples to field conditions need to be evaluated. A licensee must provide model support for the simplified flow model.

Various other phenomena may influence the development of models to represent the transport of radionuclides in groundwater. Radioactive decay is well understood; however, approximations may be required when implementing radioactive decay and ingrowth in groundwater transport models. For radionuclides with decay chains with multiple progeny, the behavior of the progeny in the environment may be substantially different than the behavior of the parent radionuclide (e.g., the K_d for Am-241 may be substantially different from the K_d for Np-237). In cases where a long-lived radionuclide has decay products of radiological significance, selecting the lowest plausible K_d value may not result in the highest estimated dose. A licensee should provide sufficient justification for the K_d values selected. Ideally, site-specific K_d values should be used. However, when site-specific values are not available, generic values may be assessed. Numerous publications have documented generic values; some even group distribution coefficients according to soil type (Sheppard and Thibault, 1990; Yu et al., 1993). Licensees should use these values with caution, as they often include variation between sites but do not include potential measurement errors (e.g., exceeding solubility limits in measurements of sorption parameters). In addition, the measurements in the compilation may have been taken using a variety of techniques.

A licensee may best assess the impact of uncertainties associated with radionuclide transport in groundwater by using multiple conceptual models. This enables a licensee to examine the effects of different credible assumptions and provides a better understanding of which processes are most sensitive and may need to be considered in greater detail.

3.3.2 Surface Water Transport

Surface water transport modeling is used to assess the radionuclide concentrations in surface water (e.g., rivers, lakes) at human access locations at or beyond the site boundary. By demonstrating that the requirements for site suitability in 10 CFR 61.50, “Disposal site suitability requirements for land disposal,” and site design in 10 CFR 61.51, “Disposal site design for land disposal,” are met, it is unlikely that surface water would directly intersect a waste disposal facility. Radionuclides will likely be transported from an LLW disposal facility by pathways other than a surface water pathway. However, radioactivity released to an aquifer may discharge to surface water bodies before contact with or use by the public. Mechanisms by which radionuclides may enter the surface water include groundwater discharge and deposition associated with atmospheric transport, and overland flow (e.g., associated with erosion).

Licensees should evaluate hydrological and chemical conditions at the site to assess surface water flow and transport if surface water transport is a viable exposure pathway for public receptors. Simplified analyses to assess surface water transport may be justified at most disposal facilities sited and constructed according to 10 CFR Part 61 requirements. However, some facilities, depending on site-specific conditions, may require detailed assessment of surface water transport. Licensees should demonstrate that the method used to assess surface water transport does not bias the outcome such that radiological doses to an offsite receptor are significantly underestimated. A licensee should consider contributions to surface waters from other environmental media (e.g., groundwater seepage, atmospheric deposition, overland runoff, and erosion).

Radionuclides that are released to surface waters may be transported by a variety of processes including water flow, sediment transport, and bioturbation (Onishi, 2008). Radionuclides entering surface water systems may remain in solution, be suspended in the water column attached to particulates, or settle to the bottom and become associated with the sediments. Advection and dispersion are typically dominant processes, especially for soluble radionuclides in flowing water. Radionuclides that readily partition to suspended particles or sediments may be affected by sediment transport processes (e.g., deposition, erosion, bioturbation). Other hydrologic processes, such as turbulence and thermal and density stratification, can also affect the distribution of radionuclides in surface waters. The chemistry of the surface water, rocks, and sediments may affect the speciation or partitioning of radionuclides (e.g., due to sorption, precipitation, ion exchange, volatilization). Licensees should consider these phenomena when developing surface water transport models.

Section 3.3.6.2 of NUREG-1573 provides guidance for the assessment of surface water transport. Section 4.3.4.1.2 of NUREG-1854 provides guidance for the review of surface water transport abstractions. Also, NCRP (1996a) and NCRP (1996b) describe screening models that may be appropriate for many sites to determine whether more detailed modeling may be required. Additionally, NCRP (2005) and Onishi (2008) discuss approaches for modeling the transport of radionuclides in a variety of surface water bodies that may be appropriate depending on site-specific characteristics.

3.3.3 Atmospheric Transport

Atmospheric transport models estimate concentrations of radionuclides released to the atmosphere at offsite receptor locations. Section 3.2.6 discusses guidance for assessing the gaseous release of radionuclides (e.g., C-14, I-129, Kr-85, Rn-222, H-3) from the disposal facility to the atmosphere. Licensees may also need to evaluate the transport of particulate releases caused by direct release (e.g., wind erosion), if significant (see Section 3.2.7). Once released, radionuclides can be transported in the atmosphere to locations downwind from the disposal facility where they could contribute to the dose to the average member of the critical group. To evaluate the impacts of gaseous radionuclides downwind from release points, licensees should consider the following FEPs (Crawford, et al., 2008):

- source characteristics (e.g., configuration of the release such as the release height from the surface, puff or continuous releases, gaseous or particulate)
- atmospheric transport processes (e.g., wind and turbulence)
- radionuclide removal mechanisms (e.g., rainfall, wet and dry deposition)
- general topography of the land near the disposal facility

Source characteristics listed above include information about the material released as well as the configuration of release. The characteristics of the released material include physical form (e.g., gaseous, particulate), chemical stability in the atmosphere, and radioactive decay. For instance, particles with diameters greater than 10 micrometers may experience significant gravitational settling and possibly limited atmospheric transport. The configuration of the release may also be an important consideration. Information on the configuration includes timing and spatial extent. For instance, radioactivity may be released continuously or nearly instantaneously, as well as from a single point or over an area. Atmospheric transport

processes listed above include a characterization of the physical processes affecting transport of radionuclides (e.g., advection, dispersion). These processes include movement by wind, represented by wind direction and speed, as well as mechanical and thermal turbulence, which can result in mixing caused by eddies. Removal mechanisms listed above include deposition, as gases and particles are deposited on surfaces and possibly removed from the atmosphere. Deposition may result from a variety of processes, both dry and wet, including impingement, electrostatic interactions, chemical reactions, and rainfall. Site topography as well as nearby engineered structures could affect atmospheric transport; licensees should exercise caution when using atmospheric transport data collected at a different location for a site-specific evaluation.

When release of gaseous radionuclides occurs at the receptor location, atmospheric transport phenomena may have minimal impact. However, the approach to estimating gaseous concentrations may have a significant impact on estimated doses. A common approach is to employ a box model. The box model estimates the average gaseous concentration in the box by estimating the flux rate from the land surface to the air and calculating the mixing associated with the annual average wind speed flowing over the surface into the box. The problem with this approach for calculating gaseous concentrations is that wind speed is highly variable on a much shorter timescale than the annual averaging period. The highest concentrations in the box are associated with the periods of calm winds and not the periods of highest winds. Integration of short-term wind speed data to estimate average concentrations should be used, or a conservative (e.g., low) value should be assigned for annual average wind speed.

An important consideration in atmospheric transport modeling is the spatial scale of the atmospheric processes. Atmospheric processes such as wind direction and speed may vary over a wide range on spatial scales from a kilometer or less to thousands of kilometers as well as over the course of hours to years. Therefore, the consideration of atmospheric transport processes and their parameterization is a function of the spatial and temporal transport scale of interest. Atmospheric transport from LLW disposal facilities to an offsite human receptor is typically on the order of less than a kilometer to tens of kilometers. Temporal scales of interest for LLW disposal range up to 10,000 years to assess the annual dose to an offsite receptor from releases. Licensees should consider processes consistent with the context of the transport scale, both spatially and temporally. Additionally, input data should be consistent with the scale of transport, both spatially and temporally. For instance, winds may vary in speed and direction with time and height. Therefore, meteorological data to support model parameterization should be consistent with and account for uncertainty and variability over the spatial and temporal scales of interest for the site. Licensees should demonstrate the representativeness of model parameters for the scales of concern for the performance assessment.

The assessment of the performance of some disposal facilities may require a licensee to evaluate the atmospheric transport of radionuclides associated with particulates. For instance, progeny of gaseous radon are charged and are attracted to atmospheric particles of opposite charge. Key factors for a licensee to consider when evaluating the transport of particulates include mass loading, resuspension rate, deposition rate, and wind speed. These factors are dependent on site-specific conditions such as soil type, wind distribution, and other meteorological conditions, as well as mechanical disturbances.

The assessment of the performance of some disposal facilities may require consideration of these processes in the atmospheric transport abstraction, whereas simplified analyses that eliminate some processes may be justified for other sites and disposal practices. The

atmospheric transport models may need to consider these site-specific complexities when (1) licensees take credit for significant delay in radionuclide migration by air transport from the disposal facility to the receptor, (2) licensees take credit for significant dilution, and (3) there is limited model support. A simplified analysis may be acceptable for facilities for which the model can be shown to be clearly conservative or for which the simple model is well supported by multiple lines of evidence, including field tests demonstrating that the model and its parameters appropriately represent or bound site conditions. A simplified atmospheric transport modeling approach may be sufficient to gain an adequate understanding of the FEPs associated with a specific system. More detailed atmospheric transport modeling, however, may be necessary to identify important processes and assess the impacts of uncertainty and variability needed to support the simplification of atmospheric transport. Licensees should demonstrate that the methods used to assess atmospheric transport do not bias the outcome such that radiological doses to an offsite receptor are significantly underestimated.

NUREG-1573 provides a screening approach to assess whether more detailed consideration of atmospheric transport modeling is required for gaseous radionuclides. As discussed in Section 3.3.6.3.2.1 of NUREG-1573, the approach uses the total gaseous radionuclide release over 1 year and conservative meteorological conditions for wind speed, atmospheric stability, and atmospheric diffusion. However, the licensee should justify the conservatism of the meteorological assumptions and parameters. NCRP Report No. 152 discusses possible approaches for estimating quantities of particulates suspended in the air. An NRC staff-recommended screening approach to evaluate the transport of particulate matter, which is suspended in the air and transported downwind, assumes that the radionuclide concentrations in the atmosphere are equal to the concentrations in the surface soil or other source from which they originated. Section 3.3.6.3.2.2 of NUREG-1573 provides guidance for sites where more detailed analyses may be required. In addition, Crawford et al. (2008) discuss models to assess atmospheric transport of radionuclides. Licensees should be aware that the output of the models is dependent on their assumptions and the data to support the parameters. Model selection and parameterization should be consistent with site conditions and supported by field observations, laboratory experiments, and other relevant information.

3.3.4 Biotic Transport

Biotic transport is defined as the transport of radionuclides from a disposal facility by biota (NRC, 1982d). Other sections of this document cover the indirect effects of biota on the disposal system, such as damage to an engineered cover by burrowing animals. This section does not cover plant uptake or radionuclide movement within the biosphere after radionuclides have been released through other mechanisms, such as groundwater release. Instead, this section focuses on offsite transport of radionuclides that have been released from the disposal facility by biota. One example of intrusion and active transport involves the Russian thistle (tumbleweed or *Salsola kali*). The long root system (on the order of meters) allows the plant to take up radionuclides from the soil (i.e., biota-enhanced release). In autumn, the mature plant dries and detaches from the surface; it is then blown by the wind until it encounters an obstruction (biotic transport). Thus, the plant can cause both vertical and horizontal movement of radioactive material from a disposal site.

A qualitative assessment of the biotic transport pathway may be sufficient depending on the site, its characteristics, and the characteristics of the disposed waste. Biotic transport may be the primary transport mechanism at sites without viable water pathways.

Secondary transport occurs after transfer of radionuclides to biotic sources (i.e., plants and animals) such as the movement of radionuclides within and among the plants and animals associated with the food chain. Examples of processes that transfer radionuclides to biotic sources include overland flow, atmospheric deposition, and use of contaminated groundwater for irrigation. Overland flow can result in the deposition of radionuclides into a surface water body. Once deposited, radionuclides can remain in the water where they may be taken up by animals drinking the water or by plants extracting water through their root systems; radionuclides settling to the bottom sediments of a surface water body may be extracted by plant root systems. Additionally, radionuclides released to the atmosphere may accumulate on plant surfaces or in the soil by wet or dry deposition and eventually become incorporated in plants and animals. The groundwater transport pathway is often the most common means for the transport of radionuclides from LLW disposal facilities to the surrounding biota since the waste is buried below ground. Secondary transport allows for additional displacement of radionuclides available to the biota. Food chain activities, such as the consumption by animals of fruits containing radionuclides, are an example of secondary transport. Upon entering the food chain, processes such as ingestion and excretion provide a means for the movement of radionuclides throughout the system. Secondary transport is evaluated using modeling of the biosphere.

3.4 Biosphere

For the purposes of a performance assessment, the biosphere is the physical environment of the central scenario or alternative site scenarios accessed by a receptor. The objective of biosphere modeling is to calculate estimates of radiological exposures to humans, in terms of the average member of the critical group, from radionuclide releases from the disposal site over time and space. Licensees can use the resultant exposures for comparison with the 10 CFR 61.41 performance objective. The biosphere includes the transfer of radionuclides through the human food chain and human *dosimetry*, including characteristics and lifestyles of the human receptors. There are two specific areas to consider in the assessment of doses to humans. First is the process of identifying and modeling the mechanism of radionuclide transfer through the biosphere to humans, often termed pathway analysis. Second, the dosimetry of the exposed individual must be modeled. This step is termed the dose assessment. Section 3.3.7 of NUREG-1573 discusses pathway analysis and dose assessment in detail and provides acceptable approaches for performing these analyses.

The pathways to include in a dose assessment will vary from site to site. A licensee can use a screening process to determine which pathways need to be included in the dose assessment. A licensee's documentation of the decisions made about inclusion or exclusion of the various pathways should be transparent and traceable. The assessment results do not need to include any pathways evaluated and found individually or in combination to contribute less than 10 percent of the total dose limit in 10 CFR 61.41. The sum of the doses from all the pathways that are excluded should not be more than 10 percent of the dose from included pathways. If the results of the analysis approach the regulatory limit, then the 10 percent that could otherwise be excluded must be included in the demonstration that the performance objectives will be met. If there are alternative receptors for a particular site, then the significance of the exposure pathways should be analyzed for each receptor. This approach is needed because pathways determined to be insignificant based on one site scenario may not be insignificant for other site scenarios.

3.4.1 Average Member of the Critical Group

The term “any member of the public” in 10 CFR Part 61 refers to the receptor for which the dose calculation is performed. The “average member of the critical group” is an acceptable construct to use as the receptor in performance assessments to demonstrate compliance with dose criteria. The critical group is the subset of the population that is most likely to be exposed to radiation. Figure 3-2 provides a conceptual representation (plan view) of the average member of the critical group concept for a hypothetical disposal site.

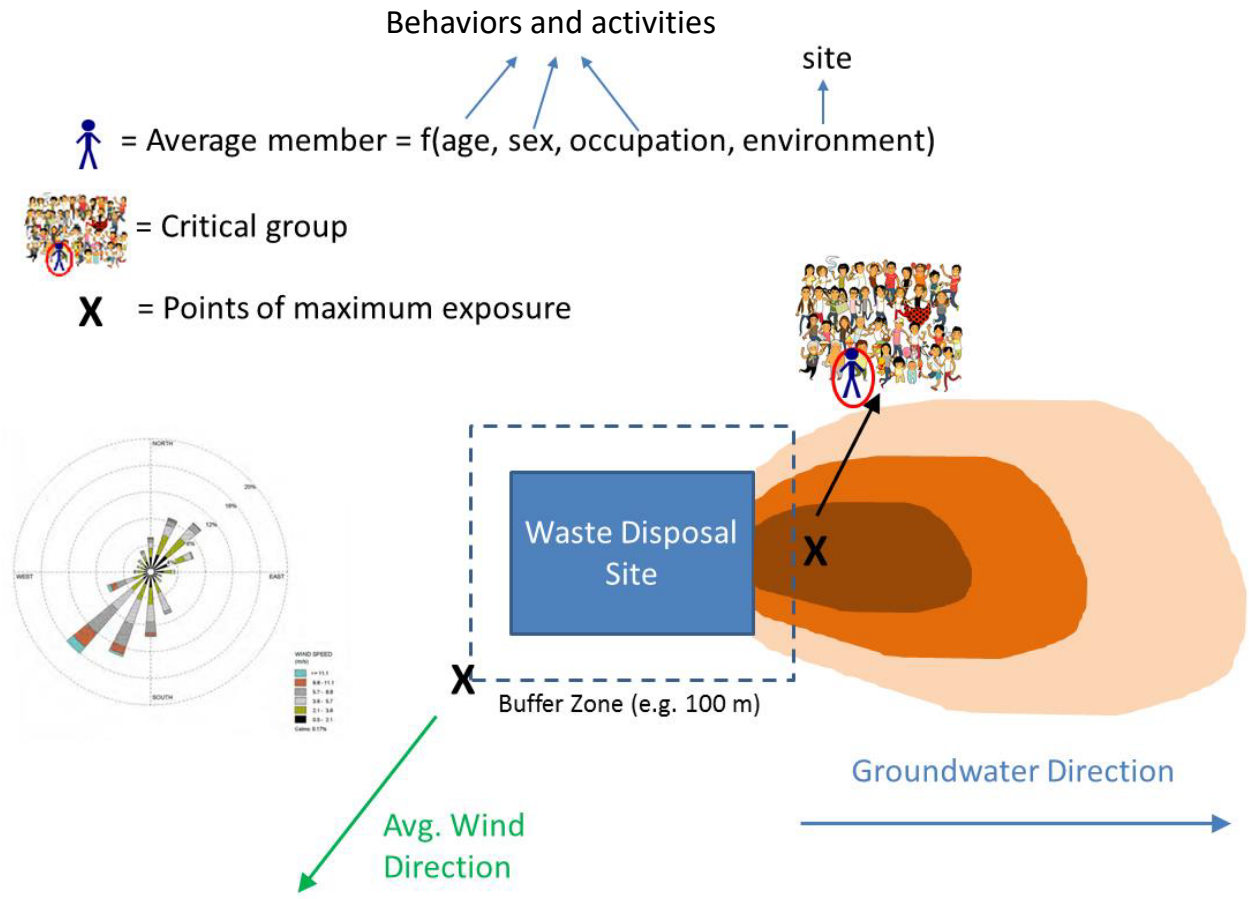


Figure 3-2 Receptor Concepts - Average Member of the Critical Group

The public is defined as all individuals who are expected to live in the vicinity of the LLW disposal site. Not all individuals will live in a location that could result in exposure to radiation, and, furthermore, the points of maximum exposure for individual pathways (e.g., groundwater, air) may not coincide. The critical group would be individuals living at the point(s) of highest concentrations outside the buffer zone surrounding the disposal units. The buffer zone is a portion of the disposal site that is controlled by the licensee and that lies under the disposal units and between the disposal units and the boundary of the site. Generally, the buffer zone is limited to 100 m (330 ft) from the edge of the disposal units. It is acceptable to take a conservative approach and sum the contributions from all pathways, regardless of differences in

location. The average member of the maximally exposed group is then used to compare to the radiological dose limits.

The term “any member of the public” is not referring to the hypothetical maximally exposed individual within the population subset. The variability in doses between the critical group and the members of public not exposed to radiation is, in general, much larger than the variability in lifetime doses within the critical group. Considering the unknown characteristics of the future exposure group, this simplification is reasonable and warranted. This approach ensures protection of the general population.

The purpose of the public dose limit is to limit the lifetime risk from radiation to a member of the general public. The conversion factor used to equate dose into risk is based on data from various populations exposed to very high doses of radiation, such as the atomic bomb survivors, and these populations contained individuals of all ages. Therefore, variation of the sensitivity to radiation with age and gender is built into the standards that are based on a lifetime exposure. A lifetime exposure includes all stages of life, from birth to old age. For ease of implementation, the radiation standards, which are developed to minimize the lifetime risk, limit the annual exposure that an individual may receive. The member of the public is not limited by regulation to be an adult, though in many cases, for practical application, it is an adult. The radiological dose is a product of the environmental concentrations, transfer pathways, uptake rates, exposure times, and dose conversion factors. These factors must be considered together when evaluating radiological doses. The behaviors and activities by a common receptor, such as the resident farmer, result in exposure times and uptake rates that are generally higher than for most other receptors. In this case, the adult resident farmer would be the average member of the critical group.

The pathways analysis and dose assessment describe the activities in which an average member of the critical group would be engaged. The input parameters that describe the transport of, and exposure to, radionuclides can generally be classified as behavioral, metabolic, or physical. Behavioral parameters collectively describe the behavior hypothesized for the average member of the critical group. The behavior is normally consistent with local practices (e.g., time spent gardening, vegetable consumption rates). Metabolic parameters also describe the exposed individual, but generally address involuntary physiological characteristics of the individual (e.g., breathing rates, factors converting intake of unit activity to dose by radionuclide). Physical parameters collectively describe the physical characteristics of the site (e.g., geological, hydrologic, geochemical, ecological, and meteorologic inputs). Section 4.3.5 of NUREG-1854 provides guidance on behavioral, metabolic, and physical input parameters used in the biosphere modeling.

Licensees may need to perform a pathways analysis and dose assessment more than once if the future evolution of the site is uncertain. Besides the central scenario, future site uncertainty may require the licensee to analyze pathways and assess dose in an alternative site scenario (see Figures 2-6 and Figure 2-7 in Section 2.5.4.2) to consider alternative changes to the physical environment over time and the impact that may have on receptor behaviors or physical parameters of the site. For instance, at a site that is currently inhospitable to gardening because of a harsh environment (e.g., in the central scenario), a licensee may need to consider an alternative site scenario where the evolution of the environment over time produces a site that is more hospitable and allows gardening. If future climate states are expected that may significantly change human behaviors, licensees may need to consider behaviors conducted in regions that are currently analogous to the expected climate. Because metabolic parameters

are involuntary and future evolution of humans is difficult to estimate, licensees should rely on current information and do not need to forecast changes to metabolic parameters over long timeframes.

3.4.2 Dosimetry

The dose methodology used to demonstrate compliance with the performance objectives of 10 CFR Part 61, Subpart C, should be consistent with the dose methodology specified in the standards for radiation protection set forth in 10 CFR Part 20. The dose methodology is how individual dose factors (e.g., external, ingestion, or inhalation) are calculated for each radionuclide. Licensees may use updated dose factors, which have been issued by consensus scientific organizations and incorporated by the U.S. Environmental Protection Agency (EPA) into Federal radiation guidance. Additionally, licensees may use the most current scientific models and methodologies (e.g., those accepted by the International Commission on Radiation Protection (ICRP)) appropriate for site-specific circumstances to calculate the dose.

In the performance assessment and inadvertent intruder assessment, licensees may use the dose conversion factors for inhalation and ingestion developed by the EPA as published in EPA-520/1-88-020, Federal Guidance Report (FGR) No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," issued September 1988 (EPA, 1988). Similarly, it is appropriate for a licensee to use the EPA's external dose factors as published in EPA-402-R-93-081, FGR No. 12, "External Exposure to Radionuclides in Air, Water, and Soil," issued September 1993 (EPA, 1993). These dose factors are selected to ensure consistency of the dosimetry models used in deriving these factors with 10 CFR Part 20.

Dose conversion factors are a function of the chemical form of each element, by either its gastrointestinal tract uptake fraction (known as the f_1 factor for ingestion dose factors), or its solubility class (solubility in lung fluid for inhalation dose factors). Licensees should provide justification in the performance assessment and inadvertent intruder assessment for the chemical forms assumed, particularly if a radionuclide has more than one value (e.g., strontium or uranium). Licensees should use reasonably conservative solubility classes for modeling radionuclides in a performance assessment, considering that the element may change in chemical form as it moves through the environment before reaching receptors. For acute inadvertent intruder scenarios, different sets of chemical forms may be expected as the chemical forms of radionuclides in the disposal unit environment may differ from those in the external natural environment.

An example of a conservative approach to selecting dose factors appears in NUREG/CR-5512, "Residual Radioactive Contamination from Decommissioning, Technical Basis for Translating Contamination Levels for Annual *Total Effective Dose Equivalent*," issued October 1992 (NRC, 1992). Appendix E, Section E.1, of Volume 1 of that NUREG describes the values recommended for use in the screening models developed for application to decommissioning. Specifically, Table E.6 gives inhalation (i.e., solubility) class and the f_1 factor for each radionuclide. Because the values are recommended for screening analyses, in most cases, the solubility class selection will maximize the potential inhalation dose. For plutonium, the solubility class represents the most common chemical form that will likely be encountered in environmental situations. For the other radionuclides, the solubility classes and gastrointestinal uptake fractions are defined for the combination resulting in the highest dose.

The licensee should use a consistent dose methodology for the performance objectives (i.e., for the performance assessment (10 CFR 61.41), inadvertent intruder assessment (10 CFR 61.42), and assessing doses during operations (10 CFR 61.43)). If licensees wish to use a methodology not consistent with the definitions in 10 CFR Part 20, they may request an exemption through 10 CFR 61.6, "Exemptions," from the definition of "weighting factor" as defined in 10 CFR 20.1003, "Definitions." For example, the licensee could ask to use (1) the latest dose conversion factors (e.g., ICRP Publication 72, "Age-Dependent Doses to the Members of the Public from Intake of Radionuclides, Part 5, Compilation of Ingestion and Inhalation Coefficients," issued September 1996 (ICRP, 1996)), or (2) EPA-402-R-99-001, FGR No. 13, "Cancer Risk Coefficients for Environmental Exposure to Radionuclides," issued September 1999 (EPA, 1999).[2] In SRM-SECY-01-0148, "Staff Requirements—SECY-01-0148—Processes for Revision of 10 CFR Part 20 Regarding Adoption of ICRP Recommendations on Occupational Dose Limits and Dosimetric Models and Parameters," dated April 12, 2002 (NRC, 2002a), the Commission directed the NRC staff to continue to consider and grant, as appropriate, licensee requests to use revised internal dosimetry models on a case-by-case basis.

Licensees must select organ dose weighting factors and corresponding dose factors (i.e., extrapolation of the dose to a specific organ to the effective dose to the whole body) that were developed using the same dose methodology. For example, if licensees use the organ dose weighting factors specified in 10 CFR 20.1003 that were developed from the ICRP Publication 26 dose methodology, they should use the corresponding dose factors that are tabulated in FGR Report No. 11 and ICRP Report 30, "Limits for Intakes of Radionuclides by Workers" (ICRP, 1979; ICRP, 1980; ICRP, 1982). If licensees use the organ dose weighting factors that were developed from the ICRP Publication 60 dose methodology, "1990 Recommendations of the International Commission on Radiological Protection," issued 1991 (ICRP, 1991), they should use the corresponding dose factors that are tabulated in FGR Report No. 13 and ICRP Publication 72. Licensees should not select organ dose weighting factors from 10 CFR 20.1003 and dose factors from ICRP Publication 72, because they were not calculated using consistent dose methodologies.

Licensees should provide an adequate technical basis for age-based considerations, *critical group* assumptions, and the chemical forms, consistent with the dose methodology system being used. Age-based considerations should evaluate the sensitivity to the total dose, rather than specific pathways.

If a licensee chooses to modify its existing performance assessment and inadvertent intruder assessment based on the availability of new dosimetry information, it will need to submit the updated analyses for review and approval, as with any other update (see Chapter 9.0). The licensee may need to request an exemption through 10 CFR 61.6 to use the latest dose conversion factors to perform dose analyses for individuals during operations to meet 10 CFR 61.43 performance objective.

[2] The regulations at 10 CFR 20.1003 define the weighting factors, based on ICRP Publication 26, "Recommendations of the ICRP," issued 1977 (ICRP, 1977), that apportion the risk of stochastic effects resulting from irradiation of an organ or tissue to the total risk of stochastic effects when the whole body is uniformly irradiated. Dose methodologies that differ from ICRP Publication 26 may deviate from those weighting factors; thus, licensees must request an exemption through 10 CFR 61.6 from the definition of weighting factor in 10 CFR 20.1003 to use the alternative dose methodology.

4.0 INADVERTENT INTRUDER ASSESSMENT

The regulations in 10 CFR 61.23(c) specify the standard that must be met to protect individuals who could occupy the disposal site after closure and unknowingly be exposed to radiation from the waste (i.e., inadvertent intruders as defined in 10 CFR 61.2). The requirement does not apply to protection of individuals who may recognize that a radiation hazard exists and knowingly or deliberately choose to ignore the hazard. The standard states that licensees should provide reasonable assurance that individual inadvertent intruders will be protected in accordance with the performance objective in 10 CFR 61.42. To demonstrate that the performance objective in 10 CFR 61.42 will be met, licensees must conduct an inadvertent intruder assessment. The requirements for the inadvertent intruder assessment are specified in 10 CFR 61.13(b).

Because the 10 CFR 61.42(a) requirements related to barriers to intrusion differ for near-surface and specialized land disposal facilities, the analyses of potential radiological consequences also differ. For a near-surface disposal facility, the inadvertent intruder analysis would generally involve contributions from multiple exposure pathways, including direct contact with the waste. For a specialized land disposal facility, the assessment would include exposure to any radiological releases from the disposal units that an intruder could be exposed to onsite without directly contacting the waste (e.g., consumption of contaminated groundwater pumped onsite). Section G.5 of Appendix G provides guidance on intrusion assessments for specialized land disposal facilities. The following sections provide guidance related to conducting site-specific inadvertent intrusion assessments for a near-surface disposal facility.

4.1 Inadvertent Intruder Behaviors and Activities[1]

In developing 10 CFR Part 61, the NRC staff recognized that the presence of institutional controls makes it unlikely, though possible, that an individual will occupy a site after closure as an inadvertent intruder and become a receptor for radioactivity released. The regulations in 10 CFR Part 61 state that licensees are to consider both (i) agricultural and residential activities and (ii) reasonably foreseeable activities that are consistent with activities and pursuits occurring at the time of development of the inadvertent intruder assessment. The regulations do not specify a particular set of inadvertent intruder behaviors and activities to be used in the assessment to demonstrate compliance with the performance objective for protection of inadvertent intruders. Licensees may use various methods, both generic and site-specific to demonstrate compliance with the criteria.

A licensee should address the following questions about the inadvertent intrusion scenario:

- What human activities are reasonably foreseeable in the vicinity of the land disposal facility?

[1] The behaviors and activities of inadvertent intruders are sometimes referred to as “intruder receptor scenarios.” To avoid confusion with the more global use of scenarios in technical assessment, this section uses the term "behavior and activity.”

- How might an intruder unknowingly come into direct contact with the disposed waste or be exposed to its radiation?
- How does the radioactive material move through the environment on the site?
- What are the inadvertent intruder's behaviors that will determine their exposure?

An intruder scenario is a type of stylized scenario composed of one or more exposure pathways and features, events, and processes associated with the behaviors and activities of humans who inadvertently occupy the disposal facility. Intruder scenarios are developed from reasonable sets of behaviors and activities related to the future use of the site and the associated receptor characteristics. Therefore, an intruder scenario describes potential future land uses, human behaviors and activities, and interaction with the engineering design and natural setting. In most situations, there are numerous possible activities through which inadvertent intruders may interact with or be exposed to radiation from the disposed waste. The criteria in 10 CFR Part 61 do not require an assessment of all possible intruder scenarios; their focus is on reasonably foreseeable activities on the part of the inadvertent intruder. Licensees do not need to assess inadvertent intruder scenarios involving behaviors and activities that are considered implausible because of persistent physical constraints at the site.

The inadvertent intruder is defined in 10 CFR 61.2 as "…a person who occupies the disposal site after closure and might engage in agricultural and residential activities and other reasonably foreseeable pursuits that could unknowingly expose the person to radiation emitted or released from the waste in the disposal units." The assessment of an intruder scenario should focus on those behaviors and activities from which the inadvertent intruder could reasonably be expected to receive the greatest exposure to radiation from the waste. Because of the uncertainties in predicting human behaviors and activities in the distant future, the NRC staff recommends that licensees consider a suite of generic intruder scenarios, described in Section 4.2.1, that represent reasonably foreseeable inadvertent intruder activities. To limit speculation about future human activities, licensees are also permitted to use a site-specific intruder scenario based on other reasonably foreseeable pursuits that are consistent with the activities and pursuits occurring in and around the site at the time of development of the inadvertent intruder assessment. Section 4.2.2 provides guidance on developing site-specific intruder scenarios. Sections 4.2.1 and 4.2.2 use several terms to describe inadvertent intruder scenarios, which are defined and compared in Table 4-1.

In their assessments, licensees may use either generic inadvertent intruder scenarios, similar to those described in NUREG-0782, or site-specific inadvertent intruder scenarios. A site-specific intruder scenario may be developed by modifying the exposure pathways associated with the behaviors and activities considered in the generic intruder scenarios, or it may be constructed based on waste characteristics, disposal practices, site characteristics, and, when appropriate, projected land use. The NRC staff views the generic inadvertent intruder scenarios as reasonably conservative for estimating potential radiological exposures to an inadvertent intruder while limiting excessive speculation about future human activities. These intruder scenarios include normal activities that humans typically engage in, in a variety of environments, and they contain a nearly comprehensive set of exposure pathways reflecting the generic nature of the original analysis for the development of 10 CFR Part 61. Licensees may use the generic inadvertent intruder scenarios described in Section 4.2.1 in an inadvertent intruder assessment to demonstrate compliance with 10 CFR 61.42.

Table 4-1 Definitions and Comparisons of Inadvertent Intruder Scenario Terms

Type of Scenario		Evaluation Purpose	Definition, Behaviors, and Activities
Plausible	Generic	All can be used to demonstrate compliance with the inadvertent intruder performance objective. Licensees should use the scenario from which the inadvertent intruder would reasonably be expected to receive the greatest exposure to radiation from the waste to demonstrate compliance.	The scenarios used to inform the waste classification criteria at 10 CFR 61.55 that include agriculture, dwelling construction, and drilling for water.
	Site-specific		A scenario developed using site information either from scratch or by modifying a generic scenario that is consistent with activities occurring in and around the disposal site at the time of development of the assessment.
	Reasonably foreseeable		Reasonably foreseeable scenarios are based on: (i) normal activities; and (ii) other pursuits that are consistent with activities occurring in and around the disposal site at the time of development of the assessment. Normal activities include agriculture, dwelling construction, resource exploration and drilling for water. The NRC staff believes that the generic intruder scenarios associated with normal activities continue to be plausible assuming the lapse of institutional controls and the loss or significant degradation of the capabilities of intruder barriers. The NRC staff also views the generic intruder scenarios as reasonably bounding over long timeframes, given the uncertainty in predicting human activities in the distant future. However, licensees can also rely on site-specific scenarios that are consistent with activities occurring in and around the site at the time of development of the assessment.
	Less likely, but plausible	Not analyzed for compliance but may be used to risk-inform the decision.	A scenario involving Intruder activities that are plausible, assuming the lapse of institutional controls, based on the capabilities of intruder barriers, site characteristics, and historical uses, but that are **not** reasonably foreseeable considering normal activities or other pursuits that are different from activities occurring in and around the site at the time of closure. These scenarios are usually site-specific.
Implausible		No analysis required.	A scenario involving intruder behaviors and activities that could not occur upon lapse of institutional controls, because of persistent physical limitations of the site.

However, in some cases, they may need to modify the generic intruder scenarios based on site-specific conditions (e.g., waste streams, facility designs, or environmental conditions) to demonstrate compliance.

Licensees should justify their scenario selections. Some licensees may need to provide only minimal site-specific data to support their assumptions. Other licensees may need to provide more thorough justifications, possibly including an evaluation of alternate intruder scenarios. If a licensee creates an intruder scenario based on site-specific conditions, it should provide transparent and traceable documentation of the justification for the assumptions used in developing the scenario (e.g., to justify the inclusion (or exclusion) of certain behaviors, activities, and exposure pathways).

4.2 Inadvertent Intruder Assessment Overview

The regulations at 10 CFR 61.13(b) require licensees to prepare an inadvertent intruder assessment to demonstrate that exposures to an inadvertent intruder will not exceed the objectives specified in 10 CFR 61.42. The primary objective of an inadvertent intruder assessment is to quantitatively analyze the potential radiological exposures to an individual who is assumed to occupy the site or to be using the site in some way after the loss of institutional controls. The assessment should consider reasonably foreseeable activities that the intruder might engage in onsite, through which they may unknowingly be exposed to radiation from the waste emplaced in or generated from the disposal facility. Reasonably foreseeable activities may include agriculture, dwelling construction, and drilling for water, as well as other pursuits that are consistent with the activities occurring in and around the site at the time of development of the inadvertent intruder assessment.

An ***inadvertent intruder assessment*** is an analysis involving site-specific, prospective modeling of the potential radiological consequences of reasonably foreseeable human activities that might unknowingly occur should an individual occupy a near-surface disposal LLW facility after the lapse of institutional controls. An inadvertent intruder assessment has two primary objectives:

- to determine whether reasonable assurance of compliance with the performance objective for protection of inadvertent intruders can be demonstrated
- to identify insights that support additional site-specific design and control measures to preclude intrusion or to limit radiological impacts to acceptable levels should an inadvertent intrusion occur

The process for conducting an inadvertent intruder assessment, shown in Figure 4-1, is similar to the process for conducting a performance assessment. It is designed to answer the following questions, often referred to as the risk triplet:

(1) What could occur?
(2) How likely is it to occur?
(3) What are the consequences?

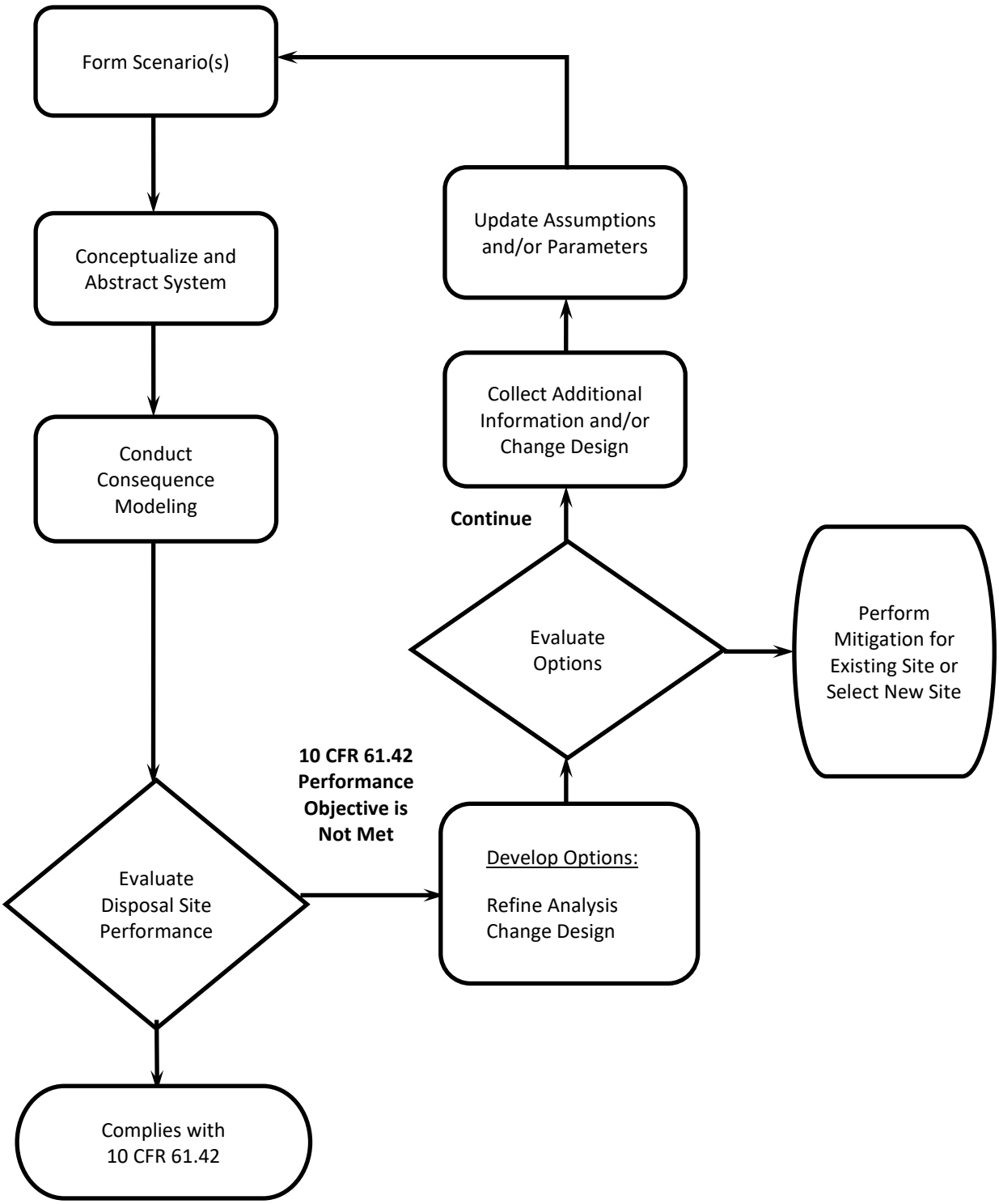


Figure 4-1 Example of an Inadvertent Intruder Assessment Process (10 CFR 61.42)

Inadvertent intrusion is considered unlikely; it is expected only if institutional controls are ineffective or societal memory is lost. However, as time passes after facility closure, these protections cannot be ensured and inadvertent intrusion may be possible. Thus, as stated in 10 CFR 61.59, "Institutional requirements," for technical analyses, institutional controls may not be relied upon for more than 100 years following transfer of control to the disposal site owner.

There is no scientific basis for quantitatively predicting the nature or probability of future disruptive human activity over long timeframes. This is in contrast to a natural process for which a scientific basis may be developed to support a probability of occurrence and relative uncertainty. Therefore, an inadvertent intruder assessment does not consider the probability of inadvertent intrusion occurring. Rather, consistent with recommendations from the NCRP (2005), to demonstrate compliance with 10 CFR 61.42, the assessment assumes that reasonably foreseeable receptor scenarios occur, in which the inadvertent intruder would be expected to receive the greatest exposure to radiation from the waste.

The approach to evaluating the first two questions of the risk triplet for an inadvertent intruder assessment is significantly different from the approach for a performance assessment. This is a key reason why 10 CFR Part 61 treats these two types of assessment separately. For the reasons mentioned in the preceding paragraph, inadvertent intruder assessments qualitatively consider the likelihood that human intruders will disturb the disposal site after a loss of institutional controls. In contrast, offsite exposures to the general population are driven by natural processes and events, such as those evaluated in a performance assessment, and could occur at any time after disposal. The performance assessment does not assume that institutional controls or societal memory would be lost (NCRP, 2005). Performance assessments can quantitatively evaluate the likelihood of natural events or processes and their effect on the performance of a disposal facility to limit radiological consequences to members of the public beyond the site boundary. The inadvertent intruder dose assessment is tied to specific future human behaviors and activities.

Inadvertent intrusion is unlikely, albeit possible, because of the presence of required institutional controls. A secondary objective of the inadvertent intruder assessment is to provide insights that support additional site-specific design and control measures to preclude inadvertent intrusion or to limit radiological impacts to acceptable levels should an inadvertent intrusion occur. The additional measures could include site-specific inventory limits or other mitigation techniques, such as additional intruder barriers or enhanced disposal practices. Chapter 8.0 provides additional guidance on defining inventory limits based on the results of the technical analyses, including the inadvertent intruder assessment.

As with the performance assessment, site scenario analysis and model abstraction are key attributes of an inadvertent intruder assessment. In a site scenario analysis, the analyst identifies, screens, and constructs site scenarios from relevant FEPs for the disposal facility. For an inadvertent intruder assessment, an analyst forming scenarios to evaluate should focus on identifying reasonably foreseeable activities that an inadvertent intruder might engage in on the site (i.e., receptor scenarios). In the near term, licensees can assess reasonably foreseeable human activities based on site-specific conditions. However, future human activities are uncertain at longer times. Therefore, 10 CFR Part 61 requires that licensees assess: (i) normal activities, such as agriculture or dwelling construction, which are typical of human pursuits in various times and locations and generally involve the pathways of most concern, and (ii) pursuits that are consistent with activities occurring in and around the site at

the time the inadvertent intruder assessment is developed. Consideration of both normal and reasonably foreseeable activities at the time the analysis is developed limits excessive speculation about human activities in the distant future, regardless of how features and processes relating to or affecting the facility design and site characteristics over the compliance period (e.g., the presence of intruder barriers and their degradation, future climate conditions) might affect intruder activities and potential pathways of radiological exposure. Section 2.5.3 discusses methods to identify and screen FEPs that are also appropriate for the inadvertent intrusion assessment.

Model abstraction is the process of incorporating the significant FEPs into conceptual and mathematical models that reasonably describe how the facility will limit radiological exposures to inadvertent intruders. Model abstraction for an inadvertent intruder assessment is similar to, although typically less complex than, model abstraction for a performance assessment. Section 2.7.2 discusses model abstraction techniques for technical analyses that are also appropriate for the inadvertent intrusion assessment.

Licensees should use the results of the inadvertent intrusion assessment to identify intruder barriers and understand the uncertainties in barrier capabilities. Reviewers should use insights drawn from their evaluations of the inadvertent intruder barriers and uncertainty analyses (see Sections 4.4 and Section 2.2.3, respectively) to focus their review of the inadvertent intruder assessment on topics that are important to inhibiting contact with waste or ensuring that potential radiation exposures to an inadvertent intruder will meet the 10 CFR 61.42 performance objective.

The following sections provide guidance on the process for conducting site-specific inadvertent intrusion assessments. Section 4.2 provides guidance on selecting receptor behaviors for use in the assessments. Section 4.3 provides guidance on specific aspects of modeling radiological consequences for inadvertent human intrusion. Section 4.4 provides guidance on the information that licensees should include as part of the inadvertent intruder assessment to identify intruder barriers and describe a technical basis for the barriers' capabilities. Section 4.5 describes institutional controls in the context of the inadvertent intruder assessment. When reviewing a licensee's inadvertent intruder assessment, reviewers should consult Chapter 2.0 for guidance on general considerations regarding site scenario analysis, model abstraction, and performance assessment that are highlighted in the preceding paragraphs. Chapter 8.0 discusses other considerations related to developing waste acceptance criteria from the technical analyses (e.g., using the information from inadvertent intruder assessments to establish site-specific measures to limit radiological impacts if inadvertent intrusion occurs).

4.2.1 Generic Intruder Scenarios

The NRC used a limited number of generic intruder scenarios based on normal behaviors and activities to inform the development of 10 CFR Part 61 waste classification criteria (NRC, 1981a; NRC, 1982b; NRC, 1986a). Appendix F provides a summary of the assumptions and approaches used to develop the waste classification limits in 10 CFR 61.55. The intruder scenarios involve both direct contact with disposed waste and indirect contact through consumption of contaminated food, as well as intruder scenarios involving a single, acute exposure and those involving long-term, chronic exposure. The NRC used the direct-contact intruder scenarios to develop the 10 CFR Part 61 waste classification and segregation criteria (NRC, 1981a; NRC, 1982b) and later, to update the analysis (NRC, 1986a). The scenarios selected were hypothetical constructs intended to provide reasonable bounds on the exposure

of inadvertent intruders to radiation from the LLW for a reference disposal facility; they helped establish waste classification criteria to be applied at all licensed disposal sites. The regulatory basis for the original 10 CFR Part 61 assumed that inadvertent intrusion would occur after the end of an active institutional control period administered by the landowner or custodial agent. Institutional control of the disposal site was expected to continue beyond the active institutional control period but were not relied upon to ensure safety.

There are five generic intruder scenarios—three direct-contact intruder scenarios involve acute exposures, one direct-contact intruder scenario involves chronic exposures, and one groundwater intruder scenario involves chronic exposures as discussed below:

(1) intruder-construction, in which the intruder receives acute exposures while excavating into disposed waste during construction of a dwelling or building

(2) intruder-discovery, a variant of the intruder-construction scenario, in which the intruder recognizes the presence of waste during excavation and ceases activity

(3) intruder-drilling, in which the intruder receives an acute exposure while drilling through waste to install a well

(4) intruder-agriculture, in which an intruder receives chronic exposures following construction of a dwelling built in the intruder-construction scenario

(5) intruder-well, in which an intruder is exposed to contaminated groundwater while living on the disposal facility site

Table 4-2 summarizes these intruder scenarios and the pathways by which the intruder receives exposures. The following subsections discuss the details of each scenario.

4.2.1.1 Intruder-Construction and Intruder-Discovery Scenarios

The intruder-construction scenario involves the construction of a dwelling directly above the disposed waste. During construction activities, workers are assumed to come in contact with some of the waste (e.g., during excavation of a basement). Some waste is also assumed to be dispersed into the air by the excavation and emplaced onto the immediate area around the dwelling's foundation. Exposures are estimated for the pathways listed in Table 4-2, including inhalation of contaminated dust, exposure to direct gamma radiation from standing on contaminated soil and being immersed in a contaminated dust cloud, and ingestion of contaminated dust or food on which contaminated dust has been deposited. It is assumed that the exhumed waste is not recognizable and is dispersible with characteristics similar to natural materials.

Since this intruder scenario is limited to construction activities, the release and subsequent exposure occur over a period sufficient to complete construction activities for a typical dwelling (i.e., less than a year). The length of time that the intruder is exposed to radioactivity is a function of the stability of the waste encountered. If the waste is assumed to be degraded into an unrecognizable form, then it is possible that such construction activities could proceed following intrusion into the waste. However, if the waste is stabilized to the point that the waste is clearly distinguishable as something different than natural materials (e.g., soil), then it is likely that the inadvertent intruder would stop and investigate. In this case, a subset of the intruder-construction scenario is termed the intruder-discovery scenario. The exposures in the

intruder-discovery scenario are expected to occur over a very limited period of time (less than a day), since construction is unlikely to resume following the discovery.

Table 4-2 Exposure Pathways of Generic Intruder Scenarios

Intruder Scenario	Exposure Pathway							
	Inhalation[#]		Ingestion[†]			Direct/External[‡]		
	Air	Soil	Food (Air)	Food (Soil)	Food (Water)	Air	Soil Surface	Soil Volume
Acute Exposures								
Intruder-Construction	●		●			●		●
Intruder-Discovery	●		●			●		●
Intruder-Drilling	●		●			●		●
Chronic Exposures								
Intruder-Agriculture	●		●	●		●		●
Intruder-Well		●			●	●	●	

[#] "Inhalation" includes pathways originating from breathing contaminated air due to suspension of soil particles caused by human activity (air) and caused by natural suspension and volatilization of surface soil (soil).

[†] "Ingestion" includes pathways for plant-to-human, plant-to-animal-to-human, and plant-to-animal-to-product-to-human uptake. "Food (air)" considers food pathways originating from atmospheric deposition on plant surfaces and surrounding soil, leading to soil-to-root transfer. "Food (soil)" considers food pathways originating from soil-to-root transfer from contaminated soil. "Food (water)" considers food pathways originating from irrigation deposition on plant surfaces and the surrounding soil as well as uptake of radionuclides originating from ingestion of contaminated water (i.e., water-to-human; water-to-animal-to-human; and water-to-animal-to-product-to-human).

[‡] "Direct/External" includes exposure to gamma rays from standing in homogeneously contaminated air (air), standing on a homogeneously contaminated surface area (surface), and standing on homogeneously contaminated ground (volume).

4.2.1.2 Intruder-Drilling Scenario

The intruder-drilling scenario is a variant of the intruder-construction scenario that was developed in an update to the NRC's initial impacts analysis (NRC, 1986a). In this scenario, the intruder installs a well to secure an adequate water supply for living needs (e.g., for a dwelling or agriculture). If groundwater quality or quantity is inadequate at a particular disposal site, this scenario may not be possible (see Section 4.2.2.1.2 for details on screening scenarios based on site-specific groundwater quality or quantity).

During drilling activities, the drilling crew is assumed to inadvertently drill through the waste, bringing some waste to the land surface in the drill cuttings. If resistance is encountered (e.g., from resistant intruder barriers) during drilling, the crew is assumed to simply move a few yards horizontally to a new location and drill a new borehole.[2] Exposures are estimated for the pathways listed in Table 4-2, including exposure to direct gamma radiation from standing near the borehole (where drill cuttings collect) or near a mud pit (if drilling fluids are used). Because this intruder scenario is limited to drilling activities, the release and subsequent exposure occur over the very limited period of time needed to complete drilling activities (typically less than a day). Though a mud pit was evaluated in the initial analysis, current practices vary (e.g., drill cuttings are sometimes spread on the surface). The licensee should justify the assumed cuttings management practices.

The exposures pathways listed in Table 4-2 include inhalation of contaminated dust from drilling, exposure to direct gamma radiation from standing near the contaminated drill cuttings, immersion in a contaminated dust cloud, and ingestion of contaminated dust or food on which contaminated dust has been deposited. The primary difference between this scenario and the intruder-construction scenario, other than exposure time, is the volume of waste exhumed. The intruder-drilling scenario assumes that drilling is performed to supply water for living needs for a single dwelling or for agriculture. The volume of material exhumed is limited to the dimensions of the borehole rather than the dimensions of the dwelling footprint.

4.2.1.3 Intruder-Agriculture and Intruder-Well Scenarios

The intruder-agriculture scenario involves an individual or individuals living in the dwelling constructed in the intruder-construction scenario. Exposure pathways for the intruder-agriculture scenario, given in Table 4-2, include those considered for the intruder-construction scenario, as well as consumption of (1) food grown in the contaminated soil, (2) animals that consumed contaminated fodder, and (3) contaminated animal products (e.g., milk and eggs).

The intruder-agriculture scenario is assumed to be possible only if the waste has been degraded to a form that is indistinguishable from soil. The length of time that the individuals spend in the contaminated area is greater for the intruder-agriculture scenario than for the intruder-construction scenario because the latter scenario is only concerned with exposures to the intruder during construction activities.

The intruder-agriculture scenario used in developing the waste classification tables did not include consumption of water from an onsite well (as in the intruder-well scenario) because the

[2] The term "borehole" is used to refer to the wellbore drilled to install a well. It is not to be confused with the borehole disposal technology used for disposal of certain types of radioactive waste.

exposures from migration of radionuclides in groundwater are a function of site-specific environmental and hydrogeological conditions and total activity, rather than directly related to waste concentration. For radionuclides that are important from a migration standpoint, NUREG-0782 recommends that licensees establish site-specific inventory limits based on groundwater migration considerations. Licensees evaluating the generic intruder-agriculture scenario should also consider the use of contaminated groundwater from an onsite well.

Combining the intruder-well scenario with the intruder-agriculture scenario, ensures that licensees consider the consumption of food grown in contaminated soil, consumption of contaminated well water, and exposure to ground and plant surfaces that are irrigated from the contaminated well.

4.2.1.4 Criteria for Selecting Generic Intruder Scenarios

Licensees may adopt the generic intruder scenarios described in Section 4.2.1 to demonstrate compliance with the performance objective if the facility's design, operation, and site are suitable for their use. The scenario used to demonstrate compliance should be the one associated with the greatest reasonably foreseeable dose. Because the generic intruder scenarios are reasonably conservative, comprising a nearly comprehensive set of exposure pathways, they are anticipated to lead to higher estimated radiological exposures than site-specific intruder scenarios. Licensees who use the generic intruder scenarios may save time and effort by reducing the amount of site characterization, modeling, and reviews needed compared to using a site-specific intruder scenario.

Licensees should be aware that the use of the generic intruder scenarios may not be appropriate to demonstrate compliance for certain sites because of the following factors that may limit or alter the activities and exposure pathways for an inadvertent intruder:

- characteristics of the disposal site, such as the presence of adequate water
- facility design, particularly the expected long-term capabilities of engineered intruder barriers
- disposal practices, such as waste emplacement as a deterrent to intrusion
- waste characteristics, including migration behaviors of radionuclides and progeny

Licensees using a generic intruder scenario should demonstrate that it is reasonable or conservative for their particular site, facility design, and disposal practices. Section 4.2.2 contains guidance on using site-specific physical information to justify the use of a scenario in demonstrating that the inadvertent intruder performance objective is met. Examples 4.1 through 4.4 illustrate factors that licensees should consider in selecting generic intruder scenarios. Depending on the characteristics of the disposal facility and its environment, a licensee may need to or elect to consider other site-specific intruder scenarios (e.g., industrial, urban, or recreational) or modify the generic intruder scenarios to include additional exposure pathways (e.g., radon gas migration).

Reviewers should assess a licensee's analysis of whether the generic intruder scenarios are suitable, given actual site conditions and disposal practices (e.g., the spatial distribution of radionuclides). The reviewer should consult information provided to demonstrate compliance

with 10 CFR 61.12 to support the evaluation. NUREG-1199, Revision 2, provides guidance on information applicants should submit to demonstrate compliance with 10 CFR 61.12.

Example 4.1

A licensee proposes to dispose of all waste 10 m (33 ft) below the ground surface. The licensee performs an intruder assessment to demonstrate compliance with 10 CFR 61.42 using the generic intruder-construction (acute) and intruder-agriculture (chronic) scenarios.

Conclusion: The generic intruder scenarios identified above may not be suitable to demonstrate compliance because they assume residential construction, which typically does not involve activity at depths beyond approximately 3 m (10 ft). Therefore, the intruders will not come into direct contact with the disposed waste. To justify the suitability of the default scenarios, the licensee may evaluate other reasonably foreseeable generic scenarios (e.g., intruder-drilling, intruder-well) or site-specific scenarios in which the intruder comes into contact with the waste to demonstrate that the selected scenarios would be expected to result in the greatest exposure to radiation from the waste rather than reasonable alternatives. Alternatively, the licensee may justify why other plausible scenarios are not reasonably foreseeable. The licensee should consult Section 4.2.2 for guidance on screening scenarios based on site-specific information.

Example 4.2

A disposal facility is located in a geographic region that currently lacks an adequate groundwater source for drinking and irrigation. The licensee demonstrates that the inadvertent intruder performance objective is met using the intruder-agriculture scenario but excludes the exposure pathways resulting from the intruder-drilling and intruder-well scenarios.

Conclusion: Excluding exposure pathways from the intruder-well scenario may be appropriate for this disposal facility given the site conditions. The licensee should provide a justification to support exclusion of both groundwater-dependent exposure pathways and drilling as an activity. It may still be necessary to consider drilling for other resources (e.g., petroleum extraction), if those activities are occurring near the site. The licensee should demonstrate the expected persistence of the conditions used as a basis for excluding intruder activities and exposure pathways during the operational lifetime of the facility. For a more robust assessment, the licensee could also evaluate how future climate conditions could affect the site during the postclosure time period being evaluated. For example, evolution of the climate near the site could increase groundwater recharge, turning a presently inadequate source of groundwater into an adequate groundwater supply in the future. Section 4.2.2 provides guidance on using site-specific information to screen exposure pathways.

Example 4.3

A licensee proposes an engineered facility in which the waste is disposed of in reinforced concrete trenches (e.g., hot waste cells). The licensee uses the generic intruder-discovery scenario to demonstrate that 10 CFR 61.42 performance objective will be met.

Conclusion: The generic intruder-discovery scenario may be suitable for demonstrating that the performance objective will be met during the time period when the reinforced concrete trenches are distinguishable from the natural materials (e.g., rocks and sediments) that an intruder might encounter at the site. The licensee should provide a technical basis, including a consideration of cementitious degradation processes and local geology, to support its determination of the time period over which the hot waste cell would be distinguishable from natural materials. The licensee should demonstrate that other site-specific scenarios are not reasonably foreseeable, or that the intruder-discovery scenario results in a greater radiological exposure than other reasonably foreseeable site-specific scenarios. Section 4.2.2 provides guidance on screening scenarios based on site-specific information.

Example 4.4

A licensee proposes to dispose of significant quantities of depleted uranium at a disposal facility. The licensee demonstrates that the inadvertent intruder performance objective is met using the generic intruder scenario with the default exposure pathways that result in the greatest radiological exposure to the intruder.

Conclusion: In this example, the licensee should develop a scenario based on site-specific waste characteristics to demonstrate that the performance objective is met. The scenario should include credible exposure pathways consistent with the waste being disposed of. The generic intruder scenarios do not assess exposure to radon gas, a decay product in the uranium-238 decay chain that can migrate from the disposal cell to the surface of the site. The licensee should evaluate potential impacts from radon generation, migration to the surface, and potential exposures to an intruder for comparison with the performance objective. The licensee may need to consider implementing additional disposal requirements, such as minimum disposal depths or engineered intruder barriers, to provide reasonable assurance that an inadvertent intruder is protected.

NUREG-1200 provides guidance for reviewers on evaluating the sufficiency of the information submitted to demonstrate compliance with 10 CFR 61.12. In general, the generic intruder scenarios are acceptable for demonstrating compliance if licensees can demonstrate that the scenarios are suitable for the disposal facility and can reasonably be expected to result in greater exposure to radiation from waste than other reasonably foreseeable receptor scenarios. The reviewer should also assess the licensee's justification for the evolution of site conditions over the time periods of interest (i.e., the compliance period or the performance period), and should verify over what time the conditions supporting the licensee's determination can be expected to persist. For instance, if a license applicant, in proposing a new facility, relies on an intruder scenario that is consistent with activities anticipated to be occurring around the site at closure, which could be many decades away, the applicant should provide a basis to support future site conditions at the time of closure. Likewise, a licensee proposing to eliminate certain

pathways from an intruder scenario because of the persistence of an engineered barrier should provide a technical basis supporting the persistence of the engineered barrier over the time period of interest to justify eliminating the pathway.

Because of the uncertainties in predicting human behavior in the distant future, limitless speculation is possible on the types of activities an intruder may engage in. The generic intruder scenarios represent normal human activities (e.g., providing shelter, engaging in agriculture, and seeking natural resources such as water) and avoid excessive speculation about future activities. They can also be used to evaluate a nearly comprehensive set of exposure pathways that are expected to be sufficient to assess the need for additional measures to mitigate doses to inadvertent intruders. Licensees can use information on the likelihood of natural processes to support intruder scenario development.

Licensees can assess the effects of potential degradation of the capabilities of intruder barriers (e.g., concrete degradation or erosion of cover materials) to justify when one of the generic intruder scenarios could be initiated. However, they should avoid speculating on disruptive human activities taking place more than a few hundred years in the future, because there is no scientific basis for estimating the likelihood of human disruptive activity far into the future. Reviewers should evaluate the licensee's selected intruder scenarios for consistency with the activities associated with the generic intruder scenarios described in this section and with the activities expected in and around the disposal site at the time of closure for site-specific scenarios, which are discussed in more detail in Section 4.2.2. Reviewers should also evaluate the technical basis supporting the long-term capabilities of intruder barriers and the long-term evolution of the site environmental conditions relevant to intruder scenarios. For example, a licensee should appropriately consider long-term degradation processes such as corrosion or cement degradation in evaluating the longevity of intruder barrier capabilities.

Section 4.4.2 contains guidance on evaluating the longevity of intruder barrier capabilities. Section 2.2.4 discusses considerations for evaluating whether a licensee has provided adequate model support to provide confidence in the projection of long-term processes. Chapter 5.0 presents guidance on long-term site stability (e.g., erosion).

4.2.2 Site-Specific Intruder Scenarios

Site-specific intruder scenarios, which are developed by licensees, provide greater flexibility and enhance confidence that the inadvertent intruder assessment accounts for site-specific factors to demonstrate that the inadvertent intruder performance objective is met while also allowing licensees to more efficiently use disposal capacity. When developing a site-specific intruder scenario or modifying a generic intruder scenario using site-specific information, licensees should provide a technical basis to support the scenario. The intruder scenarios used to demonstrate that the performance objective will be met should be reasonably foreseeable scenarios for which the projected exposures to inadvertent intruders do not underestimate potential exposures; that is, other reasonably foreseeable scenarios should not result in higher doses to inadvertent intruders than the selected scenarios. This does not mean that the licensee should simply select the scenario resulting in the highest exposure; rather, the licensee should select the scenario that results in the highest exposure out of all the reasonably foreseeable intruder scenarios. In general, it is expected that inadvertent intruder scenarios will not occur, because passive controls will remain effective. However, since the effectiveness of passive controls cannot be guaranteed over long timeframes, the inadvertent intruder assessment is needed as a regulatory tool to ensure protection.

As described in Table 4-1, in identifying reasonably foreseeable intruder scenarios, licensees may consider intruder barrier capabilities, site characteristics, the likelihood of contact with certain types of waste, and trends in area land uses. Licensees may consider activities typical of the generic intruder scenarios that are plausible within the time period being evaluated (i.e., the compliance period or the performance period) assuming the loss of institutional controls. Consideration of the capabilities of intruder barriers and the natural evolution of site characteristics is also acceptable. Licensees may limit consideration of site-specific intruder activities to those consistent with activities occurring in and around the site at the time the inadvertent intruder assessment is developed to form reasonably foreseeable intruder scenarios. However, licensees who are early in a land disposal facility's life cycle may wish to consider reasonably foreseeable activities that may occur up to the closure of the land disposal facility to ensure that future revisions to the inadvertent intruder assessment would not result in significant changes to the intruder scenarios as land uses change over the lifetime of the land disposal facility.

The types of site-specific information that a licensee should use to justify its scenario selection can be broadly categorized as physical information and cultural information. Physical information includes the location, climate, topography, geology, soil types, and water availability of the site. It also includes features of the disposal facility, such as waste characteristics, disposal methods, and the use of intruder barriers. Cultural information is related to how the human population uses the land. The physical properties of the site may change over time, particularly for long time periods; however, such changes are expected to be slow compared to changes in the human use of the land. Because of the uncertainties in predicting future human disruptive activities, cultural information is anticipated to be uncertain over long time periods. Therefore, licensees should not rely on cultural information as a basis for using more permissive intruder scenarios beyond a few hundred years. Rather, licensees should limit consideration of cultural information to the operating lifetime of the disposal facility.

The justification and analysis that the licensee should provide depends on the site physical characteristics, the reasonably foreseeable land uses, and the length of time that the radiological hazard will persist at the site. Licensees modifying generic intruder scenarios or developing site-specific intruder scenarios should consider the performance of intruder barriers and the evolution of site characteristics over the duration of the radiological hazard. They should limit consideration of demographic information and potential future uses of the site to the period up to facility closure. Such considerations might include characteristics of the disposed waste, disposal practices, degradation processes impacting intruder barriers, estimates of the evolution of site characteristics, and, when appropriate, the prevailing and possible future uses of the land, within the operational lifetime of the facility, that could constrain use. Licensees may need to evaluate several potential intruder scenarios to determine which reasonably foreseeable scenario results in the greatest exposure. These scenarios could be based on different combinations of site-specific intruder scenarios developed from radiological characteristics, disposal practices, evolution of site physical characteristics, and expected land uses.

A licensee modifying a generic intruder scenario or developing a site-specific intruder scenario should provide a technical basis that considers the longevity of the hazard, in view of the following factors:

(1) waste characteristics and disposal practices

(2) the nature of the land and reasonable estimates based on physical and geologic characteristics

(3) societal uses of land based on past historical information, current uses, and what is reasonably foreseeable in the near future (i.e., from the time of development of the inadvertent intruder assessment until facility closure)

The reviewer should evaluate the licensee's justification for its selection of the intruder scenario(s) used to demonstrate that the inadvertent intruder performance objective will be met. The screening of alternative plausible intruder scenarios should consider the guidance in the following subsections.

4.2.2.1 Site-Specific Physical Information

Physical information about the site includes information related to waste characteristics and disposal practices as well as physical characteristics of the site (e.g., the presence, quality, and quantity of natural resources such as water; soil conditions; topography; and climate). Site-specific physical information can be used as a basis for modifying the generic intruder scenarios and associated pathways. Licensees should focus on the key factors that would affect the likelihood that an intruder engaging in normal activities (e.g., agriculture, dwelling construction, or drilling a well for water extraction and use), or in other reasonably foreseeable pursuits consistent with activities in and around the site at the time of development of the inadvertent intruder assessment, might be unknowingly exposed to radiation from the waste.

4.2.2.1.1 Waste Characteristics and Disposal Practices

Licensees may consider both waste characteristics and disposal practices when developing an appropriate intruder scenario. This site-specific information may affect what other physical or cultural information licensees need to consider. For example, the hazard from long-lived waste could require a consideration of the evolution of site characteristics, such as climate, for a longer period than for shorter-lived waste. Reviewers should consider the effects of this information on the selection of an appropriate intruder scenario.

A key waste characteristic is the longevity of the hazard from the disposed waste, which should be considered as a factor when using other site-specific information to develop intruder scenarios. For example, if a facility accepts only waste containing shorter-lived radionuclides (e.g., waste containing only radionuclides with half-lives on the order of those listed in Table 2 in 10 CFR 61.55), then the licensee can limit its consideration of the potential evolution of the site's physical characteristics to a shorter time period, commensurate with the time it takes for the shorter-lived waste to decay sufficiently to result in an acceptable exposure to the inadvertent intruder. For waste containing long-lived radionuclides, it may not be appropriate for the licensee to consider land use information to justify the selection of less conservative intruder scenarios, given the uncertainty in predicting human behavior in the distant future. Section 4.2.2.2 contains further discussion of land use information.

It is also important for licensees to consider disposal practices such as the use of intruder barriers (e.g., wasteforms designed to be recognizable for long time periods, or waste emplacement at depths intended to limit direct contact with the waste). Intruder barriers that are expected to be recognizable over a given timeframe may prevent inadvertent intrusion or limit its radiological consequences during that period. Licensees should provide an adequate technical basis for their determination of the time period over which a barrier's capabilities can be expected to limit direct contact with the waste or limit an inadvertent intruder's radiological exposure. Licensees should provide a technical basis to support the use of intruder barriers that are designed to be effective for more than a few hundred years. This technical basis should evaluate whether the barrier precludes or mitigates intruder exposures. It should also consider the evolution of the physical conditions to which the intruder barriers would be exposed over the duration of the hazard.

For example, a licensee may rely on depth of disposal to exclude intruder-construction scenarios from consideration (see Example 4.5). As part of its technical basis for such an exclusion, the licensee should assess the effects of erosion and other geomorphologic processes at the site to verify that the depth will be sufficient over the duration of the hazard from the disposed waste. Section 4.4 provides guidance for reviewers on evaluating the adequacy of the technical basis for intruder barrier capabilities.

4.2.2.1.2 Characteristics of the Disposal Site

Licensees may consider the physical characteristics of their disposal site (e.g., the presence, quality, and quantity of natural resources such as water; soil conditions; topography; and climate) in justifying an appropriate intruder scenario and associated behaviors and activities. The consideration of physical characteristics often results in modifications of intruder scenarios, rather than complete elimination of scenario types. For example, farming may not be supported (i.e., it may not be economically practical) because of poor soil quality, but residential gardening may still be reasonable. In its justification of any selections or modifications based on physical characteristics, the licensee should assess the physical characteristics of the site for the entire time period over which a significant hazard from the disposed waste will persist (see Example 4.6). Reviewers should evaluate the licensee's justification for modifying generic intruder scenarios or selecting site-specific intruder scenarios based on the site's physical characteristics.

The existence of natural resources may result in reasonably foreseeable exploratory activities at the site; however, it should be noted that 10 CFR 61.50(a)(5)(i) requires that areas that have known natural resources that, if exploited, would result in failure to meet the performance objectives of 10 CFR Part 61, Subpart C, must be avoided during siting. Because of the uncertainty in predicting future human activity, licensees do not need to speculate on a future society's interest after closure in known resources near the site that are not currently economically viable. When reviewing the technical basis involving the presence or absence of economically viable resources at the site, reviewers should limit speculation about natural resources that may be economically viable for future societies but are not currently viable. Reviewers should focus on natural resources for which exploration is currently being conducted in the region (i.e., within an 80-kilometer (50-mile) radius of the site). Natural resources not currently being extracted in the region of the site may still be considered if they are currently known to exist in economically viable quantities in the immediate vicinity of the site.

Example 4.5

A licensee proposes a new disposal cell in which long-lived waste is emplaced beneath an intruder barrier at least 5 m (16 ft) thick. Because of the thickness of the cover, the licensee eliminates the generic intruder scenarios from further consideration and proposes a site-specific scenario. The reviewer questions whether generic scenarios need to be included.

Conclusion: While the site-specific intruder scenario may be appropriate given the cover thickness, the reviewer should confirm that the licensee's exclusion of the generic intruder scenarios is justified over the time period for which the long-lived hazard will persist. The reviewer should confirm that the licensee has adequately evaluated the cover's longevity (e.g., the cover thickness) and its ability to limit contact with the waste during the generic intruder scenarios for the duration of the long-lived hazard. The evaluation should consider geomorphologic processes (e.g., erosion) including the impact of long-term evolution of the site characteristics (e.g., climate) on the rate of geomorphism at the site. If the licensee cannot adequately demonstrate that the cover will remain thick enough to limit contact with the waste when considering long-term evolution of intruder barrier capabilities, then the generic intruder scenarios should be evaluated to determine whether they result in greater exposures to radiation than the licensee's site-specific scenario. Section 4.2.2 contains guidance on using site-specific information to screen exposure pathways.

Example 4.6

A licensee proposes a site-specific intruder-residential scenario that excludes water pathways because there is currently no water suitable for drinking or irrigation at the site. The reviewer questions whether this exclusion is appropriate.

Conclusion: While the site-specific scenario may be appropriate given the current lack of potable water at the site, the reviewer should confirm that the exclusion of the water pathways is justified over the portion of the compliance period for which a significant hazard persists. The reviewer should confirm that the licensee has adequately evaluated the natural evolution of the climate over the compliance period, and that this expected evolution will not result in a change in the potability of the water sources available at the site or their availability for use in irrigation. The evaluation should consider potential changes in precipitation, evaporation, and vegetative cover and its impact on transpiration, as well as the effect of these changes on recharge and water potability or availability for irrigation at the site. If the licensee cannot adequately demonstrate that the water will remain nonpotable over the duration of the compliance period, then the intruder scenario should include the water pathways.

Justifications for limiting intruder scenarios and exposure pathways based on groundwater quality and quantity should be based on site conditions rather than local codes and should be based on classification systems used by the EPA or the State, as appropriate. Arguments involving depth to water table or well production capacity should have supporting documentation from either the U.S. Geological Survey (USGS), an appropriate State or local agency, or an independent consultant. It is important to understand that water usage and attempted water recovery are different scenarios. It is usually appropriate to consider an acute driller scenario,

because although a present-day driller may know that the water at the site is nonpotable or the location does not produce viable recovery rates, that knowledge may not persist in the future.

Licensees using soil quality to justify modifying intruder scenarios should provide supporting documentation from the U.S. Natural Resources Conservation Service, an appropriate State or local agency, or an independent consultant. Reviewers should carefully consider whether the state of the soil would reasonably preclude all activities or only certain activities. In most cases, poor soil quality can reasonably preclude activities such as crop production but could allow for grazing or development of small gardens.

Licensees using topography to justify modifying intruder scenarios should provide supporting documentation of the existing topography in the form of pictures, USGS or similar topographic maps, hand-drawn maps, or a detailed description of how the topography would limit activities. Licensees may need to conduct landform evolution modeling, possibly including expert elicitation, to provide support for reasonably foreseeable landform changes to local topography. Chapter 5.0 offers additional guidance on long-term landform evolution.

When reviewing justifications involving topography, the reviewer should limit speculation about future topographical changes from offsite civil engineering projects, but not topographical changes from the final closure design of the disposal facility itself. The reviewer should evaluate the reasonableness of the inadvertent intruder performing activities on the current topography (e.g., a slope), as well as after reasonable evolution of the site due to geomorphologic processes (e.g., fluvial, eolian, tectonic, biological). The reviewer may wish to perform a site visit to evaluate the current topography and assess how the disposal design may impact the current topography. For example, reviewers may wish to assess the effect of runoff from a licensee's cover system on erosion of the site.

4.2.2.2 Site-Specific Cultural Information

Licensees may consider cultural information in justifying an appropriate intruder scenario or scenarios to demonstrate that the performance objective will be met. Cultural information is related to how the human population uses the land; it describes the types of activities an inadvertent intruder is likely to engage in on the site. Information on land use can be based on past, current, or projected land uses. For intruder scenarios, a reasonable period of interest for future land use projections is the shorter of either the anticipated operational lifetime of the facility or 100 years, depending on the rate of change in the region and the peak exposure time. The 100-year timeframe stated here applies only to the identification of future land uses to justify the selection of an intruder scenario; the licensee must evaluate potential doses that could occur over the time periods specified in the regulations. However, because of the uncertainty in estimating future human behavior, cultural information projected beyond the operational lifetime of the disposal facility should not be used as justification for a receptor scenario. Licensees may rely on suitable physical information and should refer to Section 4.2.2.1 for guidance on its use in justifying scenario selection.

A licensee's assumptions about land use should focus on current practice in the region of concern, within a radius of up to 80 kilometers (50 miles) from the site. To narrow its focus, the licensee can use information on how land use has been changing in the region and should give more weight to land use practices either close to the site or in similar physical settings. Licensees projecting land uses over the lifetime of the land disposal facility should also evaluate land uses in environmental analog locations—that is, locations outside the region of concern

that share characteristics (e.g., temperature, precipitation, topography) expected in the region of concern during the period evaluated in a site-specific intruder scenario. Consideration of such locations may help the licensee identify whether present-day land uses have been driven by past socio-economic development. Land uses primarily resulting from socio-economic development are generally more uncertain over long time periods than land uses primarily resulting from physical conditions (e.g., climate).

Licensees should categorize potential land uses as reasonably foreseeable, less likely but plausible, or implausible. Land uses that are currently occurring at similar locations in the region, or that may occur there in the near future (e.g., within approximately 100 years or within the operational lifetime of the disposal facility), should be characterized as reasonably foreseeable. In determining the likelihood of potential land uses, licensees should consider trends and area land use plans. Less likely but plausible land uses are those that are plausible (generally because similar uses occurred historically in the region of interest or are currently occurring in regions with analogous environmental characteristics) but that are counter to current trends or regional experience. Licensees should provide either a quantitative analysis or a qualitative argument discounting the need to analyze intruder scenarios generated from the less likely but plausible land uses identified. If peak doses from a less likely but plausible land use are significant, the licensee should provide support for the claim that the corresponding intruder scenario is unlikely to occur. Implausible land uses are those that, because of physical limitations, could not occur. Because of the uncertainty in predicting future human behavior, land use information should be consistent with normal activities or other typical pursuits occurring in and around the site at the time the inadvertent intruder assessment is developed. However, licensees who are early in a land disposal facility's life cycle may wish to consider reasonably foreseeable activities that could occur up to the closure of the land disposal facility, so that land use changes over the facility lifetime will not necessitate significant changes to the intruder scenarios in future revisions of the inadvertent intruder assessment.

Reviewers should evaluate the justification provided for the selection of reasonably foreseeable intruder scenarios. Reviewers may wish to involve State and local land use planning agencies in discussions if the licensee has not already requested their involvement.

4.3 Model Abstraction

Model abstraction is the process of representing a conceptual model of a dynamic site in the physical world as a mathematical model governed by equations that is implemented within a numerical model. The model abstraction process tends to be simpler for inadvertent intruder assessments than for performance assessments. However, it is similar in both cases; therefore, the guidance in section 3.3 of NUREG-1573 is also generally applicable to inadvertent intruder assessments. The following sections discuss modeling issues that require additional guidance specific to the inadvertent intruder assessment.

Intruder scenarios involving direct contact with waste generally require simpler modeling approaches because the release of radionuclides from the disposal units is primarily determined by the human activities assumed. For example, for an inadvertent intruder assessment evaluating only direct contact with waste, model abstractions may be needed only for the degradation of intruder barriers, the source term, and the resulting biosphere exposures. Such calculations can often be performed in a spreadsheet. Intruder scenarios involving the release and migration of radionuclides through the site environment from natural processes may require modeling approaches that are more similar to those needed for performance assessments.

Chapter 2.0 discusses general considerations for performing and reviewing model abstractions for the various technical analyses. These considerations also apply to inadvertent intruder assessments. Guidance on selecting site-specific input parameters for models and providing a technical basis can also be found in NUREG-1757, Volume 2, Revision 2, Section I.6. Although, the guidance in NUREG-1757 is oriented toward decommissioning activities, the concepts presented are also relevant to the development of inadvertent intruder assessments for LLW disposal.

Reviewers should focus on verifying that the intruder abstraction is based on a reasonable representation of the disposal site. In particular, reviewers should verify that licensees understand the importance of the assumptions, models, data, and uncertainty in the intruder assessment. Reviewers should recognize that the licensee's models may range from highly complex process-level models to simplified models such as response surfaces or look-up tables. The reviewer should determine whether the intruder assessment appropriately accounts for model and parameter uncertainties. Sections 2.2.3 and 2.7.4 discuss general considerations for uncertainty and sensitivity analysis.

The numerical models used to implement mathematical equations are represented in software packages commonly referred to as "codes." Reviewers should verify that the intruder assessment codes and models and the supporting data used by the licensee are documented and have been verified in accordance with a QA/QC criterion that is acceptable to the NRC staff. Section 2.2.1 of this document provides guidance on QA for technical analyses. Additional guidance can be found in Chapter 9 of NUREG-1199, NUREG-1200, and NUREG-1293. The licensee should justify the conceptual model used in the intruder assessment. Reviewers should also review the source term models, transport models, exposure models, and biosphere models. Reviewers should assess the QA/QC documentation and the conservatism of alternate codes and models. Section I.5 of NUREG-1757, Volume 2, Revision 2 provides guidance on selecting codes and models, as well as on approaches to obtaining NRC acceptance of the codes and models, which is presented in the context of decommissioning activities but is generally relevant to intruder assessments.

The following sections provide guidance on developing model abstractions for inadvertent intruder assessments and discuss model development issues.

4.3.1 Intruder Barriers

The intent of an intruder barrier is to inhibit or prevent contact with waste and help ensure that an inadvertent intruder's radiation exposure will meet established limits, which provides reasonable assurance that the performance objectives can be met. A waste disposal facility may employ a variety of intruder barriers, depending on the nature of the waste, the facility design, and the site characteristics. Examples of intruder barriers include a sufficient depth of cover over the waste, as well as engineered structures that would protect inadvertent intruders (e.g., engineered covers, concrete vaults, engineered wasteforms, or waste containers). Each intruder barrier will perform its intended functions over a certain time period, which should be justified by the licensee. Section 4.4 provides additional guidance on developing the technical basis for intruder barrier capabilities.

The objective of a model abstraction of an intruder barrier is to establish model representations of the intruder barriers that are reasonably consistent with their intended capabilities and

expected behavior over time. A primary outcome of the intruder barrier model abstraction is an estimate of the time after disposal when intrusion could occur. This output is affected by the intruder scenario, the design of the intruder barriers, and the impact of natural processes on the longevity of the barriers' capabilities. Reviewers should apply the guidance in Section 2.7.2 when evaluating the model abstraction of intruder barriers. Reviewers should consider the extent to which the licensee relies on barrier capabilities to demonstrate the performance objective is met. If the licensee has demonstrated that intruder barriers have a minor impact on protection of the intruder, a simplified review should be sufficient.

The nature of the activities in which an inadvertent intruder might engage will affect the time at which they might unknowingly be exposed to radiation from the waste. Licensees should assess the capability of the intruder barriers to preclude contact with the waste or limit radiological exposures. For example, scenarios involving construction activities could be limited to the intruder-discovery scenario as long as wasteforms are expected to remain stable and, therefore, distinguishable from soil. Licensees should provide a technical basis for the barriers' ability to delay the time of initiation of intruder activities. The technical basis should provide detail commensurate with the delay time afforded; that is, barriers credited to provide substantial delay time require a more robust technical basis. The technical basis should consider beneficial as well as deleterious natural processes that may affect barrier capabilities. Reviewers should focus on those barriers that are expected to substantially delay the onset of intruder activities, especially activities that otherwise would result in direct contact with the waste or in other radiological exposures.

In developing models of intruder barriers, the licensee should describe the spatial relationships among the physical components (e.g., the layout and physical dimensions of a vault or cover system) and the physical distribution of the materials used in the barriers. A licensees' intruder assessment should identify the features of the intruder barriers that are most important to demonstrating the performance objective is met. Licensees should ensure that the model abstractions of intruder barriers are integrated with related model abstractions. For example, the conditions and assumptions used in analyzing the degradation of intruder barriers should be consistent with those used for other model abstractions (e.g., climate and infiltration).

Reviewers should examine the physical components that are important to demonstrating that the performance objective is met and should evaluate whether their model representations are consistent with their description, the technical basis supporting their capabilities, and other model abstractions. Reviewers should assess whether the licensee adequately describes the important design features, capabilities, and properties of the barriers (e.g., thickness, porosity, and saturation of a cover layer designed to limit gaseous fluxes to the surface). The modeling of the intruder barriers should reflect the quality that the licensee expects to achieve in the implementation of the design. Reviewers should assess whether the quality proposed can be attained, whether it is supported by an acceptable QA program, and whether it is adequately represented in the model abstraction.

Licensees should evaluate processes that may limit the effectiveness of the intruder barriers. Barriers may degrade from internal processes (e.g., interaction between incompatible materials, interaction with waste) or external processes (e.g., interaction with biota, erosion, leaching by infiltrating water, and processes such as seismically induced cracking). Barrier systems should be analyzed in an integrated manner, because of potential synergisms between degradation mechanisms. If the licensee's analysis assumes that the degradation mechanisms are independent, the reviewer should determine whether an adequate basis has been provided for

this assumption. For example, a licensee may demonstrate that the degradation analysis is reasonably conservative or that uncertainty has been adequately characterized and propagated in the model. Reviewers should verify that the degradation models are consistent with the environmental parameters, material properties, and assumptions used in the performance assessment.

The intruder assessment should account for relevant materials and conditions that could affect the ability of an inadvertent intruder to contact the waste over time. Licensees should consider interactions of the components and materials of the intruder barriers. Factors that may need to be considered include:

- compatibility of materials
- how construction (e.g., construction joints, changes in geometry, penetrations) may affect system behavior
- how failure of a design feature or some portion of an intruder barrier would affect the overall ability of the intruder barrier system to limit or prevent contact with the waste
- how the degradation of material properties will affect barrier performance over time

The temporal performance of an intruder barrier may be represented in three phases: (1) the service life, during which the barrier effectively inhibits or prevents contact with the waste and helps ensure that radiation exposures will meet the performance objective, (2) a period of decreasing performance due to ongoing processes of degradation (which may be quite short for some barriers), and (3) the period of complete degradation, in which the barrier no longer limits contact with the waste. Barriers that are designed to ensure that radiation exposures will meet the performance objective, rather than to inhibit or prevent contact with waste, may provide some diminished level of performance after degradation (see Example 4.7). The service lives of different barriers may vary significantly because of the inherent diversity and variability of barrier components. The intruder assessment should account for such variability in service life.

Licensees should provide information to support the model estimates of intruder barrier performance. Section 2.2.4 of this document provides guidance on model support. Model support can include laboratory experiments, field measurements, previous experience with similar systems, process modeling of barrier performance (e.g., detailed models of landform evolution to estimate cover erosion), natural or industrial analogs, independent peer review (NRC, 1988a), or other sources of relevant information. Reviewers should confirm that the evidence supporting the modeling is based on environmental parameters, material factors, assumptions, and approximations that are similar, or appropriately analogous, to those of the facility, and that the models are not likely to underestimate barrier degradation or failure. Reviewers should examine the procedures that the licensee used to construct and test its mathematical and numerical models. Section 4.4.1 provides guidance on a risk-informed approach to model support for engineered barrier capabilities.

Licensees should assess the uncertainty in model estimates for the longevity of intruder barrier capabilities and should evaluate the sensitivity of the potential inadvertent intruder exposures to uncertainties in the support for the model estimates. Sections 2.2.3 and 2.7.4 provide guidance on uncertainty analysis.

Example 4.7

A licensee proposes a reinforced concrete vault as an intruder barrier for the near-surface disposal of depleted uranium. The licensee describes the capabilities of the vault to both inhibit contact with the depleted uranium and limit release of radon gas to the land surface over the expected service life.

Conclusion: It would be appropriate for a model representation to assume that the vault will limit an inadvertent intruder's contact with the waste during its service life, as long as the licensee provides an adequate technical basis for the service life. Reviewers should therefore evaluate the adequacy of the technical basis for the service life. Furthermore, the licensee may conservatively assume that when the service life has ended, the vault will return to the constituent sand and gravel aggregates; in this case, the licensee will take no credit for the ability of the degraded vault to either inhibit contact with the waste or limit radon diffusion to the surface. Alternatively, the licensee may choose to assume that the degraded vault will continue to reduce radon diffusion to the surface, thereby limiting radon exposures to an inadvertent intruder. In this case, the reviewer should evaluate the licensee's basis for the continued ability of the degraded vault to reduce radon diffusion, including the licensee's analysis of the degraded material properties (e.g., effective diffusivity) to continue delaying radon transport to the surface.

4.3.2 Source Term

The objective of the source term abstraction in an intruder assessment is to estimate the radionuclide concentrations accessible to the inadvertent intruder. The modeling of the source term can be one of the most important determinants of the estimated exposures to an inadvertent intruder. Typically, source term modeling includes the inventory of radionuclides, physical and chemical characteristics, and other properties used to estimate release rates.

For the intruder assessment, radionuclides are expected to enter the environment accessible to the intruder primarily through direct contact of the intruder with the waste, though entry may also be possible through the use of contaminated surface water or groundwater on the site, or as a result of gaseous releases from the disposed waste to the atmosphere above the site. In direct-contact intruder scenarios, source term modeling is used to estimate the concentrations of radionuclides contacted and exhumed to the biosphere that the inadvertent intruder inhabits. These concentrations are used as input in the biosphere (dose assessment) modeling. In indirect-contact intruder scenarios, source term modeling is used to estimate releases occurring by advective or diffusive mechanisms, although other mechanisms may be possible (e.g., biointrusion, erosion). The release rates from the disposal unit are estimated and used as input to transport modeling.

4.3.2.1 Inventory

The characteristics of the radionuclide inventory can significantly affect the selection of models to represent the source term. Radionuclide inventories need to be addressed on a facility-specific basis. The anticipated distribution of specific radionuclides in the disposal facility inventory should be estimated by waste class, waste type, wasteform, waste stream, and waste container type, as appropriate. This information provides a basis for selecting an approach for source term modeling. The level of detail needed in the information will vary. For example, a direct-contact intruder scenario in which a discrete volume of waste is exhumed may require a simpler source term model than other scenarios involving release from the disposal units.

Licensees should describe the radionuclide inventory in the disposal facility, including each radionuclide's volume, concentration, and location within the disposal system. The radionuclide inventory used in the intruder assessment should be consistent with the waste acceptance criteria and the inventory used in the performance assessment. However, in some cases, licensees may use conservative estimates of inventory in the intruder assessment and more realistic estimates in the performance assessments. For instance, licensees may wish to represent a larger inventory in the intruder assessment than their waste acceptance criteria because the disposal facility provides sufficient margin for demonstrating compliance with 10 CFR 61.42, but the margin may be smaller for demonstrating compliance with 10 CFR 61.41. Regardless, the waste acceptance criteria, as described in Section 8.0, needs to be consistent with the results of the technical analyses.

Reviewers should evaluate the methods used to estimate the radionuclide inventory and the concentrations of radionuclides not measured, to verify that the source term model appropriately accounts for uncertainty and variability.

Licensees may use a screening approach to determine which radionuclides in the facility inventory need to be considered in the inadvertent intruder assessment. Licensees may:

- Eliminate radionuclides with half-lives less than 5 years that are not present in significant activity levels and do not have long-lived daughter products.

- Perform an intruder dose calculation using the generic intruder scenarios discussed in Section 4.2.1.

- Assume that inadvertent intrusion occurs into the spatially averaged peak waste concentration and that intruder barriers are ineffective in inhibiting contact with the waste.

- Assume radionuclides are available for the relevant exposure pathways of each generic intruder scenario. The licensee may eliminate from consideration any radionuclides with an estimated peak dose in each of the intruder scenarios that is less than 10 percent of the performance objective, provided that the sum of their doses is accounted for in demonstrating that the performance objective is met. If a radionuclide has an estimated peak dose greater than or equal to 10 percent of the performance objective in any of the generic intruder scenarios, then the radionuclide should be retained in the intruder assessment. Computer models such as RESRAD (Yu et al., 2001) may be useful in this type of screening. To avoid inadvertently screening out important radionuclides, the licensee should confirm that the dominant pathways (i.e., those contributing most to

estimated dose) in the screening calculations are consistent with those in the intruder assessment.

Reviewers should examine the licensee's description of the screening assessment to verify that the licensee has used appropriate parameters and has provided an appropriate basis for the screening out of any radionuclides.

For intruder scenarios in which direct contact with the waste occurs, source term modeling can be as simple as distributing a concentration of radionuclides in the biosphere for exposure modeling. For intruder scenarios with indirect releases from the disposal cells, such as through advective-diffusive release mechanisms, licensees may develop source term abstractions similar to those developed for the performance assessment (see Section 3.2). Licensees may also use estimated concentrations from the performance assessment at an appropriate location and point in time to assess exposures to the inadvertent intruder in the biosphere.

Reviewers should verify that licensees using radionuclide concentrations from the performance assessment have not overestimated releases in the performance assessment to conservatively estimate doses to the public, which could result in underestimation of radionuclide concentrations in the intruder assessment.

Licensees should consider the radioactive decay before the time of intrusion or may assume no decay, if such an assumption is conservative. In either case, however, licensees should consider the impact of significant progeny on the intruder. Radioactive decay can result in significant ingrowth of progeny at future times. For example, activities from depleted uranium may increase for more than 1 million years due to ingrowth of decay products. Given that the regulations specify that active institutional controls cannot be relied upon for more than 100 years, radioactive decay is expected to be more important in direct-contact intruder scenarios for shorter-lived radionuclides such as cesium (Cs)-137. Reviewers should verify that the source term modeling appropriately accounts for radioactive decay.

Licensees may account for the time of waste emplacement in the intruder assessment, as long as they provide justification (for example, describing the projected operating lifetime of the disposal facility). Technical analyses completed to support facility closure can then be updated to reflect actual waste emplacement. If a licensee assumes in its analyses that all waste is emplaced in the final year of operation, then minimal justification would is needed. A technical basis is expected to support earlier times of waste emplacement that account for decay during operations.

4.3.2.2 Radionuclide Concentrations for Direct Contact Intruder Scenarios

To develop the source term abstraction, licensees should estimate the concentration of radionuclides in the facility at times after which the direct-contact intruder scenarios would be expected to occur. These times depend on the assumed institutional control period (see Section 4.5), intruder activities envisioned in the scenario, and the capabilities of intruder barriers to preclude intrusion (see Section 4.3.1).

Licensees should consider the following when generating concentrations of radionuclides for the intruder dose assessment:

- Licensees are permitted to account for dilution of the waste following disposal due to mixing with uncontaminated materials in the disposal cells.
- The amount of mixing should account for methods of waste emplacement such as disposal depth.
- The basis for mixing may rely on site characteristics.
- The justification may include reasonable assumptions about the intruder activities.

Licensees are permitted to account for dilution of the waste following disposal due to mixing with uncontaminated materials in the disposal cells. Licensees should provide a technical basis for dilution of the waste mixed with uncontaminated materials. The modeling of the mixing should be consistent with the facility design, waste emplacement, site characteristics, and assumptions about intruder activities for the given scenario. For example, the facility design may incorporate uncontaminated materials within the disposal cells that would reduce radionuclide concentrations in the waste. Design features may include intruder barriers and stabilization materials such as backfill or grout. For example, waste may be emplaced beneath an engineered cover with uncontaminated backfill material to help maintain the stability of the disposal cells. The uncontaminated cover and backfill would likely reduce radionuclide concentrations in the waste when mixed during excavation, as considered in an intruder-construction scenario.

The estimate of the amount of mixing should account for methods of waste emplacement. For example, in the intruder-construction scenario, it is typically assumed that the intruder only excavates to a depth necessary to construct a dwelling. Waste and backfill placed beneath this depth should not be considered in the mixing assessment. The justification for mixing may rely on site characteristics. For example, in estimating dilution for an intruder-driller scenario, the licensee may consider the thickness of uncontaminated soil between the bottom of the disposal cells and the underlying aquifer or resource. To justify mixing based on deeper aquifers or resources, licensees should explain why it is more appropriate to assume mixing with the uncontaminated soils between the bottom of the disposal cells and the depth of the deeper aquifers or resources, than to assume a smaller mixing volume based on shallower aquifers or resources. Licensees should limit the mixing volume to the shallowest aquifer or resource that would be reasonable for humans to access to obtain water or natural resources. Licensees should also consider the impact of variability in the mixing volume associated with the use of shallower aquifers or resources, if it is plausible that an inadvertent intruder might access them.

It may be reasonable to assume that an intruder who does not know that the site is a disposal facility will excavate at random locations. In this case, the excavation could exhume material that was contained within multiple waste packages or disposal cells. Therefore, the waste concentration for estimating intruder impacts from excavation can be averaged over the area of the site where waste is disposed of, including uncontaminated regions within the disposal cells (e.g., backfill). The areal average should not include portions of the site beyond the disposal cell footprints, such as administrative areas or the buffer zone. If the excavation area is larger than the disposal cell area, dilution with uncontaminated soil surrounding the disposal cell may be considered.

For intruder-drilling scenarios, the waste concentrations may be based on the average concentration in the waste containers (for containerized waste) or the average concentration of the waste (for bulk waste). Uncontaminated material above and below the waste may be credited in the intruder-drilling waste concentration calculations. However, uncontaminated material within the disposal cell adjacent to the assumed drilling location (e.g., backfill) should not be credited because the driller is assumed to disturb a small, discrete area. Operational limits, such as waste emplacement strategies, may be necessary to limit the impact from wastes with significant heterogeneity.

If used, licensees should provide a basis for the random-access assumption. For example, at an aboveground disposal facility characterized by mounded disposal cells that create a "ridge-and-valley" topography (in which the disposal mounds are expansive with gentle terrain and are separated by steep and narrow "valleys" that are uncontaminated), licensees may argue that an intruder would prefer to excavate a dwelling foundation on the expansive "ridges" rather than in the steeper, narrower "valleys." In this case, random access may not be an appropriate assumption for an intruder-construction scenario.

Reviewers should evaluate the technical basis supporting the dilution of waste concentrations, considering the facility design, waste emplacement, site characteristics, and scenario assumptions. Generally, the more mixing that is assumed, the more robust the technical basis should be to support the dilution factor.

Radionuclides that are released offsite through natural processes before intrusion are generally not available to be exhumed by an inadvertent intruder. Licensees may consider radionuclide migration before the start of intruder activities and may use information from the performance assessment to estimate radionuclide releases. However, licensees should exercise caution with this approach if the performance assessment overestimates releases in order to conservatively assess exposures to the public. This may result in an underestimation of the waste concentrations remaining for the intruder to access. Ideally, the performance assessment and intruder assessment should be internally consistent.

Reviewers should assess the consistency between estimates of radionuclide releases in the intruder assessment and the performance assessment, verifying that pessimistic assumptions in the performance assessment do not result in overly optimistic assumptions in the intruder assessment.

The spatial extent over which exhumed waste is deposited at the surface will depend on the activities assumed to occur in the intruder scenario (e.g., drill cuttings management practices) and physical arguments about the exhumed volumes (i.e., how widely can a volume of exhumed waste physically be spread?). The area over which exhumed material is spread may be an important parameter because it can limit certain exposure pathways. For example, too small an area may make certain agricultural pathways unviable, while too large an area may unreasonably dilute the concentration of the waste exhumed. For the spatial distribution of the exhumed radionuclides, licensees should use reasonable physical assumptions consistent with the scenario assumptions. For example, a small volume of drill cuttings is not likely to be spread over a large area of the site. Licensees may consider Esh et al (2020) for additional information.

Reviewers should evaluate the licensee's basis for the extent of contaminated material assumed in the biosphere. Reviewers should assess the consistency between the assumptions for the spatial distribution of exhumed radionuclides and the assumed activities in the intruder scenario.

4.3.2.3 Radionuclide Concentrations for Transport Scenarios

Reviewers should assess the adequacy of the radionuclide release and transport models in the intruder assessment consistent with the guidance in Sections 3.2 and 3.3. Radionuclide concentrations in the environment to which intruders may be exposed are estimated after release and transport through the disposal site.

A performance assessment and an intruder assessment differ in several ways. First, in the intruder assessment, access points for the intruder to the contaminated environmental media should all be onsite, whereas the performance assessment involves offsite exposures to members of the public to demonstrate that the performance objective for protection of the general population is met. Therefore, the receptor scenarios for the two assessments are mutually exclusive and the performance assessment does not evaluate offsite exposures resulting from contamination that arises from inadvertent intruder activities. Second, intruder scenarios are typically independent. For example, the intruder who contacts the waste during construction of a dwelling may not be the same individual who lives in the dwelling. The radiological exposures from these distinct intruder scenarios are generally mutually exclusive and not additive. However, in some cases it may be appropriate for intruder scenarios to be dependent. For example, it would be reasonable to assume that a well is installed to supply water to a dwelling that is constructed on the site and that the intruder residing in the dwelling is also exposed to contaminated groundwater from the well. When dependent intruder scenarios are used and one is a direct-contact scenario and the other is a migration scenario, the licensee may provide a basis for including the removal of some of the waste from the disposal cell at the time of the intrusion event to calculate releases from the buried waste for the transport scenario. Licensees should also assess the transport of waste that the intruder accessed (i.e., the transport of radionuclides from waste that the intruder excavated) in the inadvertent intruder assessment.

Reviewers should verify that the source term abstraction is consistent with and appropriately integrated with other model abstractions (e.g., climate and infiltration for advective-diffusive release mechanisms). Reviewers should consult Section 3.2 for guidance on reviewing advective-diffusive release mechanisms. Reviewers should evaluate the licensee's assessment of concentrations for the direct-contact and transport scenarios to verify that assumptions in one of the scenarios do not result in unreasonably optimistic estimates in the other scenario.

4.3.3 Transport

Radionuclides directly contacted by an inadvertent intruder or released from disposal units may be transported in the environment by groundwater, surface water, air, and biota (e.g., rodents). Model abstraction for transport processes in an intruder assessment is generally similar to that in a performance assessment. Usually, the main exposure pathways for an intruder are from the direct disturbance of buried waste (e.g., from drill cuttings). However, transport of exhumed material can occur. Therefore, the guidance in Section 3.3 is generally applicable to an inadvertent intruder assessment. One notable exception is that an intruder assessment is focused on exposures onsite. Therefore, for an inadvertent intruder assessment, model

abstraction of transport processes should be focused on onsite transport processes. These may include gaseous diffusion of radionuclides from the disposal unit through the overburden to an intruder at the surface, as well as leaching from the disposal units and transport through groundwater to an intruder well, if applicable. Radionuclides released from disposal units that are transported beyond the site boundary (i.e., offsite*)* are generally evaluated in a performance assessment and do not typically need to be considered in an inadvertent intruder assessment.

4.3.4 Dose

The objective of the dose (exposure pathway) modeling in an intruder assessment is to estimate potential radiological impacts to an inadvertent intruder, in terms of the average member of the critical group, from direct contact with disposed waste or onsite releases from disposal units after the period of active institutional controls. Therefore, dose modeling integrates the information from the other model abstractions. In general, dose modeling in an intruder assessment is similar to dose modeling in a performance assessment. It consists of converting radionuclide concentrations in environmental media to dose through the various exposure pathways associated with an intruder scenario. Guidance on developing receptor scenarios is discussed in Section 2.5.4.2, and guidance on the dosimetry parameters is provided in Section 3.4.2. The paragraphs below offer additional guidance on modeling exposure pathways in intruder assessments.

Intruder exposure pathway models (or abstractions) are used to link the radiological source, transport of radionuclides within environmental media, intruder location, and intruder behaviors that lead to the exposure of an intruder to radionuclides through a combination of direct exposure, inhalation, and ingestion of contaminated water, soil, plants, and animal products, as appropriate.

Reviewers should evaluate the licensee's description of the human behaviors leading to or controlling the doses to intruders. Reviewers should confirm that models describing human behaviors that control radiological impacts are consistent with the intruder scenario and site conditions. Therefore, the occupational, behavioral, and metabolic parameters describing these models should be reviewed.

In some cases, either the location of the disposed waste, the physical characteristics of the site, or the facility design may make the generic exposure pathways inappropriate or inconsistent with an intruder scenario. The licensee should provide a transparent and traceable description of the development of the intruder scenarios used in the inadvertent intruder assessment, starting with potential land uses and site-specific conditions. Development and review of alternate intruder scenarios may be iterative. For example, the licensee may (1) develop a generic list of exposure pathways, (2) develop the site conceptual model to screen the generic list. (3) aggregate or reduce the remaining exposure pathways to the major exposure pathways, and (4) reevaluate the conceptual model to verify that the necessary processes are included.

The NRC recommends the following process for analyzing exposure pathways:

(1) Compile a list of exposure pathways. Several existing sources of information can be used for this, including NUREG/CR-5512, Volume 1; NUREG-1757, Volume 2, Revision 2; NUREG/CR-5453, "Background Information for the Development of a Low-Level Waste Performance Assessment Methodology," Volumes 1 and 2, issued December

1989 (NRC, 1989b; NRC, 1989c); and the International List of Features, Events, and Processes (SSI, 1996).

(2) Categorize the sources of radioactivity at the site (e.g., mixed in sediment or soil, groundwater).

(3) Screen out pathways for each source of radioactivity that do not apply.

(4) Identify the physical processes pertinent to the remaining pathways.

(5) Separate the list of exposure pathways into unique pairs of exposure media (e.g., source to groundwater). Determine the physical processes relevant to each exposure media pair and combine the processes with pathway links.

(6) Reassemble exposure pathways for each source type, using the exposure media pairs as building blocks, thus associating the physical processes identified for the individual pairs with the pathways.

The licensee should provide transparent and traceable documentation of its decisions to include or exclude pathways. If any single pathway or combination of pathways evaluated is found to contribute less than 10 percent of the total dose limit in 10 CFR 61.42, it does not need to be included. However, the sum of the doses from all the pathways excluded should not exceed 10 percent of the total dose. If the projected dose to an intruder from a given combination of pathways is within 10 percent of the limit, then the intruder assessment must include these pathways. If there are multiple relevant intruder scenarios, then the significance of the exposure pathways needs to be screened and analyzed separately for each scenario, because pathways that are insignificant in one scenario may not be insignificant in others.

4.4 Inadvertent Intrusion Barriers

Intruder barriers are designed to inhibit or prevent the contact of inadvertent intruders with waste that could present a hazard to them (e.g., Class C or GTCC waste). The barriers help ensure that radiation exposures to an inadvertent intruder will be within the limits of the performance objective specified at 10 CFR 61.42. Intruder barriers are distinct from and complementary to institutional controls, providing defense-in-depth. An intruder barrier is defined as either (1) a sufficient depth of cover over the waste that inhibits contact with the waste and helps to ensure that radiation exposures to an inadvertent intruder will meet the performance objective, or (2) engineered structures that provide equivalent protection to the inadvertent intruder. According to 10 CFR 61.52(a)(2), Class C waste must be disposed of at sufficient depth (i.e., minimum of 5 m (16 ft) below the top surface of the cover) or must be disposed of with engineered structures (also referred to as “structural intruder barriers”) designed to protect for at least 500 years. The regulation at 10 CFR 61.52(a)(3) requires that GTCC waste must be disposed of both at sufficient depth (i.e., minimum of 5 m (16 ft) below the land surface)) **and** must be disposed of with structural intruder barriers designed to protect for at least 500 years. Licensees that elect to develop waste acceptance criteria from the results of their technical analyses (i.e., under 10 CFR 61.13) may also need to identify certain waste streams that require intruder barriers to ensure protection of inadvertent intruders and demonstrate compliance with 10 CFR 61.42. The inadvertent intruder assessment for near-surface disposal should be used to demonstrate that a structural intruder barrier (e.g., reinforced concrete vaults) can limit exposures for at least 500 years when used for Class C and GTCC

waste, or to demonstrate that an intruder barrier can limit exposure as long as necessary to an inadvertent intruder to the dose limit prescribed in 10 CFR 61.42. For GTCC disposal in a specialized land disposal facility, the inadvertent intruder assessment should be used to demonstrate that a combination of the depth of disposal, natural site features, and engineered intrusion barriers provide reasonable assurance that an inadvertent intruder will not disrupt or contact the waste at any point in the compliance period when the waste presents a radiological hazard, and that any exposure from onsite releases of radioactivity (e.g., radon emanation) will not exceed the dose limit prescribed in 10 CFR 61.42. In either type of inadvertent intrusion assessment, a licensee should look at site-specific conditions, such as the length of time over which the disposed waste will present a significant hazard to inadvertent intruders. A licensee may use the intruder assessment to identify the need for adequate barriers to demonstrate protection of inadvertent intruders from disposal of other waste (e.g., Class A and B waste) based on site-specific conditions. Additional information about intrusion analyses for GTCC waste disposal is provided in Appendix G.

Although both intruder barriers and institutional controls can limit contact with waste, they use different mechanisms to do so. Institutional controls limit access to or use of the site after its transfer to the owner. Under 10 CFR 61.59, they cannot be relied on for more than 100 years. An institutional control program includes legal mechanisms (e.g., land use restrictions), environmental monitoring, periodic surveillance, minor custodial care, and other requirements as determined by the Commission, as well as the administration of funds to cover the costs of these activities. By contrast, intruder barriers are passive features of the disposal facility and site that are intended to enhance protection of inadvertent intruders who engage in normal (e.g., residential or agricultural) activities and other reasonably foreseeable pursuits that are consistent with activities in and around the site at the time the intruder assessment is developed.

Under 10 CFR 61.12(b), the applicant's specific technical information describing the land disposal facility must include design features related to inadvertent intrusion and intruder barriers. Applicants should identify intruder barriers and describe their capabilities in terms of inhibiting or preventing contact with disposed waste and ensuring that radiation exposures will meet the performance objective at 10 CFR 61.42. Because of the wide range of radioactive materials that may be disposed, the regulations and this guidance are not prescriptive as to the acceptability of site-specific intruder barriers. Because of this flexibility, it is important for licensees to clearly and completely document the description of the barriers' capabilities and the supporting technical bases.

The information supporting barrier capabilities should be provided for the time period over which each barrier is expected to perform its intended function, including changes to the barrier that would significantly affect its ability to protect inadvertent intruders. Barrier capabilities, including uncertainties, should be consistent with the performance attributed to the barriers in the intruder assessment. Reviewers should use information from the intruder assessment, independent calculations, and other appropriate quantitative analyses to confirm each barrier's capabilities. Reviewers should confirm that the technical bases for barrier capabilities are consistent with the technical bases for the intruder assessment. In some cases, the actual barrier capabilities may differ from those assumed in the intruder assessment; for example, a licensee may have conservatively ignored the benefits of a barrier in the intruder assessment.

The functionality and robustness of the barriers will be determined using the approach described in Section 4.4.1 and will be evaluated on a site-specific basis for each license

application. The general framework that a licensee should consider will not vary from licensee to licensee; only the depth and breadth of information supplied to demonstrate the capabilities of the intruder barriers may vary. The guidance that follows provides the framework a licensee should consider for developing and providing a basis for intruder barriers.

4.4.1 Risk-Informed Approach to Evaluating Intruder Barriers

Licensees are encouraged to use a risk-informed approach to select, design, and provide a technical basis for the intruder barriers at their site, considering the hazard level and likelihood of hazard occurrence. Licensees should provide more robust barriers and more detailed bases for barrier capabilities for higher-risk disposal facilities than for lower risk facilities. The use of the term "risk" with respect to intruder barriers means the potential for risk; the inadvertent intruder assessment and resultant actions should ensure that high risks are not realized by inadvertent intruders. The NRC considers as high risk any disposal facilities where the potential for intruder hazards may be large, such as if the barrier were to fail. For instance, a disposal facility with unmitigated doses greater than 50 mSv/yr (5,000 mrem/yr) following the institutional control period is considered high risk.

Licensees should assess the significance of intruder barriers in mitigating exposures to inadvertent intruders. An intruder barrier that reduces unmitigated doses less than or equal to 5 mSv/yr (500 mrem/yr) to smaller values is low significance. A barrier that reduces unmitigated doses ranging between 5 and 50 mSv/yr (500 to 5,000 mrem/yr) to less than 5 mSv/yr (500 mrem/yr), achieving a relative reduction of 50 percent or more, is moderate significance. A barrier that reduces unmitigated doses greater than 50 mSv/yr (5,000 mrem/yr) to less than 5 mSv/yr (500 mrem/yr) is high significance. Licensees should provide a technical basis for the capabilities of the barriers commensurate with the barrier's significance in mitigating inadvertent intruder exposures. The technical basis should evaluate the time period over which each barrier performs its intended function, including changes taking place during the various periods of performance (i.e., the compliance and performance periods).

Reviewers should use insights from a licensee's understanding of each barrier's capabilities, as well as independent analyses, to focus reviews on the technical bases supporting the capabilities of significant barriers to limit or prevent contact with the waste. Section 4.3.1 discusses guidance on modeling intruder barriers in the intruder assessment. Section 4.4.2 contains guidance on justifying the capabilities of intruder barriers modeled in the intruder assessment.

4.4.2 Technical Basis for Structural Intruder Barrier Capabilities

There is significant uncertainty in estimating the service life and long-term degradation of structural intruder barriers. The following information should be provided to support the licensee's technical basis for its evaluation of structural intruder barrier capabilities:

- design, features, and functionality of the structural intruder barriers
- site and environmental conditions to which structural intruder barriers would be exposed to
- potential degradation mechanisms, including combined and synergistic effects resulting from the service environment expected for the barriers

- suitability of selected numerical models, if used, for the estimation of structural intruder barrier capabilities
- uncertainty in parameters and models used in the assessment of barrier capabilities and the design of structural intruder barriers
- parametric or component sensitivity analysis to identify how much degradation of the structural intruder barrier is needed for noncompliance to occur
- model support for the structural intruder barrier's performance (e.g., analogs, experiments, simple engineering calculations to demonstrate the reasonableness of the results)
- QA and QC for the design, analysis, and implementation of structural intruder barriers

The capabilities of some structural intruder barriers may not be amenable to validation in the traditional scientific sense because of the long time periods involved. Therefore, licensees should provide multiple lines of evidence particularly, though not exclusively, to support barrier capabilities beyond 500 years in the future. As discussed in Section 2.2.4, there are many forms of model support, including but not limited to analogs, laboratory experiments, field experiments, formal and informal expert judgment, and engineering calculations to demonstrate the reasonableness of the results (e.g., hand calculations when numerical models are used). The detail and comprehensiveness of model support provided should be commensurate with the relative significance of the barriers in protecting the inadvertent intruder. NRC-sponsored research on the potential performance of structural intruder barriers is available (CNWRA, 2020a). As insights are developed, this guidance will be updated and refined, if necessary.

During the operational period, postclosure observation and maintenance period, and institutional control period, monitoring may be needed to verify the capabilities of structural intruder barriers. This monitoring should encompass both the environmental system surrounding the disposal facility and the performance of the facility itself. Nondestructive monitoring technologies, including designed and emplaced sensors, are preferrable to conventional post-failure monitoring. To the extent practicable, structural intruder barriers should be designed to support and simplify monitoring and maintenance.

Reviewers should use information from the licensee's description of structural intruder barrier capabilities to focus their review of the adequacy of the technical bases. Reviewers should confirm that the technical bases provide detail commensurate with the significance of each barrier's capabilities and the associated uncertainties. Based on the reviews of the intruder assessment, reviewers should confirm (1) the consistency of the technical basis for structural intruder barrier performance with the intruder assessment and (2) the quality and completeness of the technical bases for barrier capabilities.

4.4.3 Use of Structural Intruder Barriers

If a disposal facility uses structural intruder barriers, the intruder assessment should credit only the barriers' passive performance (i.e., performance in the absence of monitoring, inspection, and maintenance) to inhibit or prevent contact with waste and limit radiological exposures. The assessment of barrier capabilities should consider the reasonableness of an inadvertent breach as the barrier becomes indistinguishable from native materials, as well as the potential

degradation of the barrier over time, since monitoring and maintenance cannot be relied on beyond the period of institutional control.

The following documents provide more information about intruder barriers and domestic water well drilling practices:

- NUREG-1757, Revision 2, Volume 2, Section 3.5.4 and Appendix P. These sections discuss degradation mechanisms and capabilities of common engineered barriers. Though focused on decommissioning, the information is generally applicable to engineered barriers for land disposal facilities.

- NUREG-1757, Revision 2, Volume 2, Section 3.5.5. This section summarizes existing guidance and references that may be relevant to structural intruder barriers at disposal facilities.

- NUREG/CP-0195, "Proceedings of the Workshop on Engineered Barrier Performance Related to Low-Level Radioactive Waste, Decommissioning, and Uranium Mill Tailings Facilities," issued August 2011 (NRC, 2011a). The appendices to this document provide reference material and electronic information sources.

- The Center for Nuclear Waste Regulatory Analyses (CNWRA) report "Implications of Domestic Water Well Drilling Practices for Inadvertent Intruder Scenarios," issued December 2020 (CNWRA, 2020b). This report documents a study of drilling practices near Idaho National Laboratory and the Savannah River Site.

Licensees should evaluate the consequences of other reasonably foreseeable disruptive events caused by humans or by natural events and processes, as well as the uncertainty in projecting passive barrier performance into the future (see Example 4.8). For example, waste and disposal facilities may be subject to instability because of waste characteristics (e.g., differential settling caused by voids in the waste) or facility design (e.g., long-term physical instability of covers). Subsidence could cause the formation of small depressions on an engineered surface cover, differential settlement leading to an uneven cover surface, the formation of stress cracks on the cover surface, or the formation of open voids in the cover, each of which could reduce the depth of cover protecting an intruder.

Engineered surface covers often rely on vegetation to resist erosion. The probability that the vegetation cover will deteriorate, because of future drought or disease, depends on the hardiness and diversity of plants established on it. Fires, severe storms, and ecological succession due to changing temperature and precipitation could influence the ability of the vegetation to resist wind and water erosion and maintain a sufficient depth of cover over the waste. Even with reliance on vegetation, over long periods of time, consistent erosion even at a minimal rate may lead to a substantial loss of cover thickness (e.g., erosion at a consistent 0.5 millimeter per year rate will reduce a cover's thickness by half a meter in 1,000 years).

Example 4.8

A licensee proposes to use a reinforced concrete vault to limit access to waste by an inadvertent intruder drilling into the waste. The concrete vault would function as an intruder barrier for as long as it maintains its integrity.

Conclusion: The licensee should consider how reasonably foreseeable processes could degrade the vault's mechanical properties that would be relied on to limit access to the waste. These processes, which are site-specific, could include seismic activity, cementitious material degradation such as sulfate attack, carbonation, leaching of the cement, biological activity, and corrosion of the reinforcing steel. The licensee should also consider local drilling practices and estimate when currently used technology would likely penetrate the vault, given the estimated degradation of its mechanical properties. For instance, a concrete vault would more likely be breached by drilling in regions where hard-rock drilling is common than in regions where it is not common. Over long periods of time, it may not be appropriate to assume that only drilling practices that are in use locally at the time of development of the assessment will be relevant. At a minimum, licensees should describe the results if drilling were to occur, even if drilling is not anticipated based on locally used technology. The licensee should also consider current and projected population density, as well as availability of resources, when evaluating the likelihood that a structural intruder barrier will be breached.

4.5 Institutional Controls

An institutional control program includes legal mechanisms (e.g., land use restrictions), environmental monitoring, periodic surveillance, minor custodial care, and other requirements as determined by the Commission, as well as the administration of funds to cover the costs for these activities. Further, the institutional control period will be determined by the Commission. During the institutional control period (also referred to as the "active period"), which cannot be relied on for more than 100 years unless otherwise authorized by the Commission, the site owner is assumed to be performing monitoring, surveillance, and custodial activities. Licensees may assume in the inadvertent intruder assessment that the institutional controls are durable and will preclude intrusion throughout the institutional control period, but that inadvertent intrusion is possible after 100 years, consistent with 10 CFR 61.59.

If the Commission approves an institutional control period shorter than 100 years, then the inadvertent intruder assessment should assume that inadvertent intrusion could occur following the Commission-approved institutional control period. If the Commission approves an institutional control period longer than 100 years, 10 CFR 61.59 still requires that licensees may not rely on institutional controls to preclude inadvertent intrusion in the inadvertent intruder assessment beyond 100 years after transfer of control of the disposal site to the owner. However, the Commission could approve an exemption under 10 CFR 61.6 to this requirement. Many international programs have implemented an institutional control period of 300 years following IAEA guidance (IAEA, 2009b).

5.0 SITE STABILITY ASSESSMENT

The regulations at 10 CFR 61.44 include a performance objective for disposal site stability after closure. It states that the disposal facility must be sited, designed, used, operated, and closed to achieve long-term stability of the disposal site. For the institutional control period, a licensee must eliminate, to the extent practicable, the need for ongoing active maintenance of the disposal site following closure. The regulations at 10 CFR 61.50 require that LLW disposal sites should not be susceptible to erosion, flooding, seismicity, or other disruptive events or processes to such a degree or frequency that compliance with 10 CFR Part 61 performance objectives cannot be demonstrated with reasonable assurance. This section provides guidance on the process for developing a site stability assessment, including technical information that a licensee should provide and that a reviewer should evaluate relating to the potential effects of erosion, flooding, seismicity, and other disruptive processes. This section also addresses stability of the wasteform as well as other engineered features that might be used at an LLW disposal facility. Licensees that meet the criteria in 10 CFR 61.1(b) need not comply with 10 CFR 61.50(a) and (b) but must instead comply with 10 CFR 61.50(c).

Stability must be achieved for both short- and long-lived waste. Long-term stability of the disposal site includes the ability of the site to maintain structural stability from within the disposal units and to maintain stability of the surface of the site. The guidance in this section covers important stability issues associated with the design, operation, and closure of disposal facilities for all types of waste (i.e., short- and long-lived). Table 5-1 summarizes the FEPs that may be considered with respect to time and distance in a site stability assessment. For most disposal facilities that will be used for the disposal of commercial LLW that does not contain significant quantities of long-lived radionuclides, a licensee can focus the site stability assessment on those processes and issues that were the source of many of the early performance challenges at LLW disposal facilities in operation before the issuance of 10 CFR Part 61. These challenges arose from water management during operations and a lack of waste conditioning. Waste acceptance criteria as well as conditioning and stabilization of waste are extremely important to ensure that the performance objective associated with stability can be achieved. Minimization of void space combined with proper geotechnical engineering can help prevent future subsidence and differential settlement.

Stability is the capability of the wasteform, disposal containers, disposal site, and disposal facility to maintain their shape and properties to an extent that the disposal action will meet 10 CFR 61.41 and 10 CFR 61.42 performance objectives. “Stability” is defined in 10 CFR 61.2 as follows:

> the capability of the disposal site (e.g., wasteform, disposal containers, and disposal units) to maintain its shape and properties to an extent that will not prohibit the demonstration that the land disposal facility will meet the performance objectives in §§ 61.41 and 61.42 of this part and will, to the extent practical, eliminate the need for active maintenance after site closure and for maintenance in any form after license termination.

However, the impact of instability is not solely the change in shape and properties of the system. The change in shape and properties of the system may affect other FEPs associated with safety.

Table 5-1 Summary of Spatial and Temporal Considerations for Site Stability Assessment

	Short term[1]	Long term
Internal to Disposal Site	Subsidence Differential settlement Surface water management Biodegradable waste Gas generation	Differential rates of degradation of engineered barriers Radiation damage Wasteform degradation (e.g., leaching) Plant and animal intrusion
External to Disposal Site	Generally, not applicable if 10 CFR 61.50 requirements are met	Landform evolution Land use Cover naturalization and degradation Disruptive events Erosion protection Slope stability

[1] "Short term" refers to the period of operations, postclosure observation and maintenance, and the institutional control period.

Stability is important to do the following:

- Limit changes to the disposal system that may result in increased releases of radioactivity to the environment from increased infiltration.
- Limit erosion and similar processes (10 CFR 61.50).
- Ensure waste is recognizable by an inadvertent intruder.
- Protect an inadvertent intruder by maintaining an appropriate overburden over the waste.

The NRC strategy for waste disposal is to "concentrate and contain" the LLW for as long as it remains hazardous. A key feature of that regulatory strategy is that components of the LLW disposal system must maintain stability. Stability helps ensure that once waste is emplaced and covered, access to the waste by water, biota, or humans is minimized, reducing the potential exposure to the public.

Many of the early problems associated with LLW disposal arose from site stability issues associated with water (NRC, 2007b). To address these problems, practical regulatory requirements, such as waste segregation requirements, were introduced in the regulations to obviate or limit future problems associated with waste stability (NRC, 1982a). Disposal of long-lived waste can increase the challenge associated with ensuring waste stability, relative to short-lived LLW. For long-lived waste, it may be difficult to demonstrate the performance of the engineered features used to ensure waste stability, because the service life of those features is relatively short compared to the time those wastes remain hazardous. Site stability, in addition to the physical and chemical stability of the waste itself, can affect the performance of the waste disposal facility. Internal or external phenomena can initiate instability.

NUREG-1200 provides details about site stability focused on waste stability, erosion protection, and geotechnical issues (slope stability, settlement, subsidence). The guidance in this document is not a substitute for the information in NUREG-1200; it can be used to supplement NUREG-1200 for the site stability analyses associated with long-lived LLW. This guidance document uses the term "site stability" to refer to the overall stability of the LLW disposal system, which includes stability of the waste, disposal site, and disposal facility and surrounding environment, as applicable (because processes external to the disposal site may impact site stability). As shown in Table 5-1, for the operational, postclosure observation and maintenance, and institutional control periods, a licensee should emphasize achieving stability consistent with those items discussed in NUREG-1200. Recent advances have been made in tools to analyze near-term stability issues such as subsidence (Foye et al, 2018). Probabilistic methods have been applied to analyze differential settlement and other processes associated with surficial stability.

Most of the guidance in this section focuses on the stability assessment for the disposal of significant quantities of long-lived radionuclides. It discusses disruptive processes (i.e., hazards), technical assessment, and engineered barriers. After licensees identify and characterize potential disruptive processes, they can use different technical assessment methods to understand the impact of the disruptive processes on the ability of a disposal site to meet the performance objectives. Licensees can use technical assessment to evaluate whether the hazards can be reduced or mitigated. The technical assessment for site stability does not need to be complex (see Section 5.2.2), especially for the disposal of short-lived waste. Screening analyses may be appropriate. In addition, it may be possible for a licensee to mitigate the impact of some hazards using engineered barriers. Long-term evaluation and monitoring of the site can be used to confirm whether site stability can be achieved. If the monitoring calls into question whether site stability can be achieved, licensees can propose a mitigating engineering action to enhance stability. The technical assessment may also determine that the candidate site is not suitable for the disposal of certain concentrations and quantities of specific radionuclides. Consequently, it may be necessary for a licensee to impose limits (see Chapter 8.0) on the concentrations and quantities of LLW (e.g., GTCC waste) suitable for disposal at the proposed site. Similarly, the material itself may not be suitable for near-surface disposal under 10 CFR Part 61 regulations. Figure 5-1 shows an overview of the main elements of site stability analyses.

Stability is interrelated with the requirement for a minimum disposal depth for certain wastes. Significant quantities of uranium must be disposed so that the top of the waste is a minimum of 5 m (16 ft) below the land surface. The land surface is the minimum of either (1) the preexisting surface at the disposal facility location immediately prior to development of the facility, or (2) the land surface after development of the facility but prior to disposal of waste. Decay of uranium can produce radon that diffuses to the land surface. Traditional methods for protecting from radon emissions are to use radon barriers, typically a clay. Research has indicated that unless the waste is buried deep enough to avoid freeze-thaw cycles, root intrusion, burrowing animals, and other pedogenetic processes, the clay will be altered over time. The depth requirement is specified with respect to the land surface. This implies that erosion protection will be implemented to preserve the 5 m (16 ft) depth.

Stability and uncertainty are interrelated topics. Site stability analyses should be developed and evaluated in a risk-informed manner. The regulation at 10 CFR 61.44 requires that site stability must be demonstrated for the long-term. As timeframes increase, the uncertainty associated with site stability increases. In addition, the scope and diversity of phenomena that may impact

the stability of the disposal site increase. As previously noted, stability is to be applied in the context of meeting the performance objectives for 10 CFR 61.41 and 10 CFR 61.42. However, site stability is an independent performance objective (10 CFR 61.44).

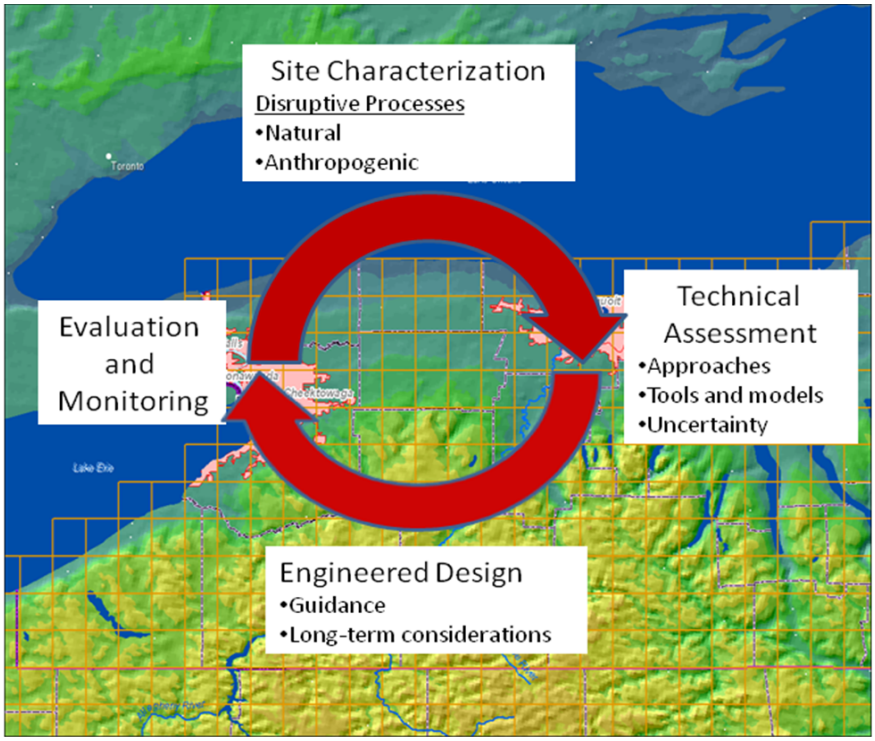


Figure 5-1 Main Elements of Site Stability Analysis

Uncertainty associated with instability may limit the ability of a licensee to demonstrate that 10 CFR 61.41 and 10 CFR 61.42 can be met. Some impacts of instability may be difficult to quantify in terms of potential dose impacts. Inventory limits may be used to mitigate the uncertainties associated with potential instability, or to prohibit disposal of waste that remains hazardous when the instability is projected to occur.

In 10 CFR 61.50[1], the NRC provides the disposal site suitability requirements used in the site selection and evaluation process. Licensees can use site selection to eliminate or greatly reduce the number of disruptive processes and events that may need to be considered in the site stability assessment. For example, alluvial fans that form at the base of mountain ranges tend to be particularly unstable landforms over the long-term (NRC, 1983b). The impact of some events may be impossible to design against, given current technology and understanding (e.g., volcanism), and therefore should be avoided. Other processes and events may be unavoidable (e.g., erosion, large precipitation events) and may be mitigated through site selection and engineered design.

[1] The terminology used in 10 CFR Part 61 is that waste is emplaced in a "*disposal unit*." The disposal units, area between units, plus the surrounding buffer zone make up the "*disposal site*." The disposal site plus the other buildings, structures, equipment, and land used to complete disposal operations is the "*land disposal facility*" or "disposal facility." In this guidance document, the term "disposal system" is used to refer to the disposal facility and surrounding environment.

Most, if not all, of the requirements listed in 10 CFR 61.50 were developed to ensure site stability, as well as to allow licensees and reviewers to evaluate the performance of the site with reasonable assurance (NRC, 1982a). Site suitability requirements are separated into hydrological characteristics (10 CFR 61.50(a)(2) and 10 CFR 61.50(a)(3)) and other characteristics (10 CFR 61.50(a)(5)). In 10 CFR 61.50(a)(2), the NRC specifies hydrological characteristics that a disposal site must have for a 500-year timeframe following closure of the disposal facility (e.g., the site must not be in a 100-year flood plain). The regulation at 10 CFR 61.50(a)(3) specifies that after the 500-year timeframe following closure of the disposal facility, if any of the negative disposal site hydrological characteristics listed in 10 CFR 61.50(a)(2) are present (e.g., groundwater discharge to the surface), that they shall not significantly affect the ability of the disposal facility to meet the performance objectives in 10 CFR Part 61, Subpart C. This is a risk-informed implementation of the site characteristics requirements.

The other site suitability requirements in 10 CFR 61.50(a)(5) specify characteristics of the site that should not be present such that they significantly affect the ability of the disposal site to meet the performance objectives of 10 CFR Part 61, Subpart C (e.g., surface geologic processes, tectonic processes). The requirements in 10 CFR 61.50(a)(5) can exclude a potential disposal site location only if the FEP in question (e.g., tectonic processes; see Section 2.5.3) may prevent the performance objectives from being met, or if defensible modeling is precluded. In addition, 10 CFR 61.50(a)(4) requires that projected population growth and future developments in the vicinity of the land disposal facility are not likely to affect the performance objectives from being met. For FEPs that result in severe disruption, defensible modeling may be precluded, and licensees, in concert with their regulators, may need to establish conservative inventory limits to mitigate the associated uncertainties. Practical solutions, other than numerical modeling, to limiting the impact of uncertainties associated with the disposal of long-lived waste may be necessary. In 10 CFR 61.50, requirements associated with specific FEPs associated with site stability include, but are not limited to, the following:

- Areas must be avoided where tectonic processes such as faulting, folding, seismic activity, or volcanism may occur with such frequency and extent to significantly affect the ability of the disposal site to meet the performance objectives of 10 CFR Part 61, Subpart C, or may preclude defensible modeling results due to large uncertainties. [10 CFR 61.50(a)(5)(ii)].

- Areas must be avoided where surface geologic processes such as mass wasting, erosion, slumping, land sliding, or weathering occur with such frequency and extent to significantly affect the ability of the disposal site to meet the performance objectives of 10 CFR Part 61, Subpart C, or may preclude defensible modeling results due to large uncertainties. [10 CFR 61.50(a)(5)(iii)].

A site stability assessment should be tailored to the types of waste disposed. A facility designed for what might be regarded as traditional LLW streams (i.e., short-lived waste and limited quantities of long-lived waste) will have a comparatively less complex site stability assessment than a facility designed for significant quantities of long-lived waste. For example, higher quality rock and more detailed testing and assessment of the durability of rock used for erosion protection will be needed for long-lived waste (compared to short-lived waste). Erosion protection is necessary for all wastes, but the durability and longevity of materials used may differ for different wastes. A site stability assessment has three areas of focus: stability of the

wasteform, stability of the engineered disposal facility, and geologic/geomorphic stability of the disposal site. For disposal of traditional LLW, a site stability assessment will likely focus on the first two areas. For disposal of significant quantities of long-lived waste, the focus will likely be on the latter two areas. The type of waste to be disposed will strongly influence the areal extent of the site stability assessment. Stability of wasteforms, disposal units, engineered barriers (such as cover systems), disposal site, disposal facility, and disposal system may be within the scope of the site stability analysis.

The scope of the analyses needed for the site stability assessment will be defined primarily by the type of waste and secondarily by the disposal system. The type of waste (e.g., short-lived, long-lived) will determine the timeframe appropriate for the site stability assessment and, therefore, the types of FEPs a licensee should evaluate. Waste disposal systems are generally reliant on passive features of the site geology and hydrology to limit long-term releases of radioactivity. To limit short-term releases, waste disposal systems may strongly rely on the performance of engineered systems. Passive systems, in general, will be used more frequently for long-term performance than active systems. Figure 5-2 provides an integrated view of the NRC staff's perspective on the temporal and spatial scales, as well as the relative influence of various processes on site stability for LLW disposal.

The NRC staff developed Figure 5-2 based on the experience with assessments of LLW disposal facilities and complex decommissioning sites. Waste disposal facilities have not experienced frequent unlikely natural events (which are defined relative to the operating experience base) because the experience base is limited to the last half century. There are not enough data to quantitatively define the relative importance of the relevant processes. Figure 5-2 is a generalization; the relative importance of a particular process at a site may differ from the conceptualization shown in the figure primarily because of the location of the site. For instance, based on historical data, glaciation in a southern State would be expected to be less important than glaciation in a northern State. Individual processes may also be dynamic such that the relative importance is not constant over time. The relative importance of climate could change based on the magnitude of effects of anthropogenic processes on climate change. With these types of caveats, the general expectation of the relative influences of various processes on site stability is as follows:

- In the near term (left hand side of the figure), erosion is expected to be the dominant process impacting stability. Secondary processes are biologic and climate. Igneous and tectonic phenomena are expected to be insignificant for a properly selected site. Glaciation is not anticipated in the near term.

- In the medium term (middle of the figure), erosion is still of primary importance, but changes to climate that can impact erosion or result in glaciation at some sites become more significant.

- In the long term (right hand side of the figure), most processes are expected to be important to evaluate in the site stability analysis. As time increases, the likelihood of a large and low-frequency seismic or volcanic event increases.

- The overall significance of various processes affecting site stability increases as time progresses. Over long timeframes, there is a higher overall likelihood that site stability will be affected by one or more of the processes.

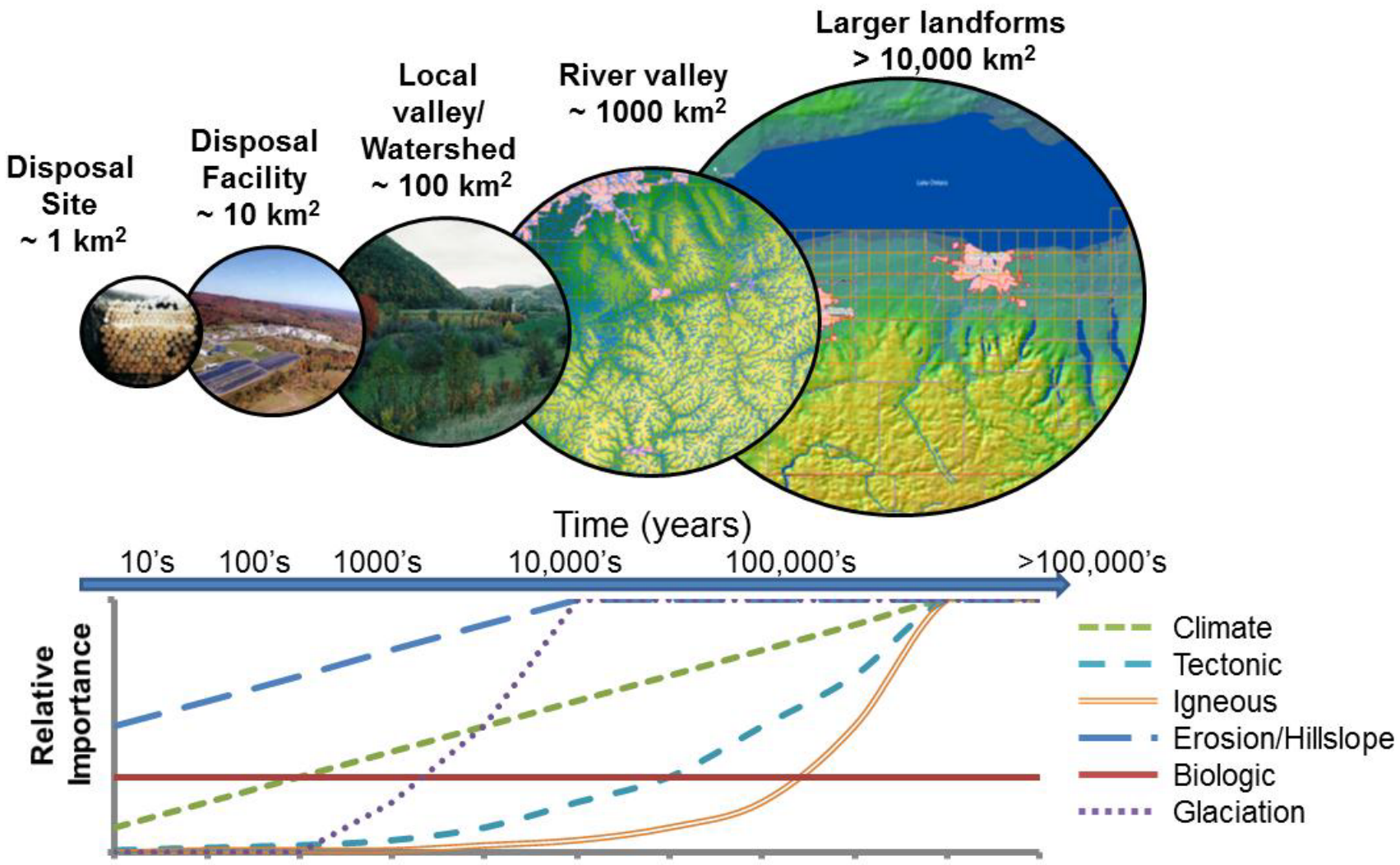


Figure 5-2 Spatial and Temporal Scales and NRC Staff Perspective on the Relative Importance of Various Geomorphological Processes Relevant to LLW Disposal

Note: The time on the figure is comparable to the persistence of the hazard of the waste

A variety of resources are available to facilitate the development or review of a site stability assessment. This section attempts to consolidate and summarize some of these resources but does not attempt to replicate them. The reader of this document should consult the original documents for more detailed information.

5.1 Disruptive Processes and Events

Disruptive natural and anthropogenic processes and events could impact the long-term stability of a waste disposal system. Section 2.5 discusses the FEP screening process that licensees can use to identify disruptive processes and events that need to be evaluated in the site stability assessment. This section describes the information that a licensee should provide, and a reviewer should evaluate with respect to identifying potential disruptive processes and events. The NRC staff expects the disruptive processes important to the estimation of the stability of an LLW disposal site to be site-specific.

The following questions are associated with the evaluation of disruptive phenomena:

- Which disruptive phenomena need to be evaluated?
  - What screening process will be used?
  - What criteria will be used for screening?
- How may the phenomena influence each other (i.e., are they correlated, dependent, independent)?
- Can the frequency and consequences of the events or processes impact the demonstration of compliance with the performance objectives of 10 CFR Part 61, Subpart C?

Licensees should describe in the site stability analysis the screening process that will be used to evaluate which disruptive phenomena to consider and the criteria used for the screening. Some natural phenomena will be expected to occur, while others will be unlikely or very unlikely. The likelihood that a phenomenon occurs will be related to the timeframe of the evaluation; as explained in Section 2.3.2, the timeframe evaluated in the analysis is dictated by the type of waste that is disposed. Longer timeframes will result in higher likelihood that low frequency events may occur over the timeframe considered.

The NRC staff recommends that licensees evaluate reasonably foreseeable disruptive conditions from natural events or processes in the site stability analysis. Reasonably foreseeable disruptive conditions include both those expected to occur and events that are moderately unlikely to occur (approximately 10 percent or greater chance of occurring) over the analysis period, not to exceed 1 million years. The reasonably foreseeable disruptive conditions should be consistent with the characteristics of the waste. In other words, if the radiological inventory of the disposal site is primarily short-lived with low concentrations of long-lived radionuclides, it would not be necessary to look at reasonably foreseeable disruptive events based on a 10,000-year compliance period. A licensee could justify a shorter period, consistent with the characteristics of the radiological inventory. However, lower frequency events will need to be considered for higher concentrations of long-lived radionuclides. NUREG/CR-3964, "Techniques for Determining Probabilities of Events and Processes Affecting the Performance of Geologic Repositories: Literature Review," Volume 1, issued June 1989 (NRC, 1989d),

provides examples of techniques that licensees can use to investigate and then estimate the likelihood of natural events or processes that might be disruptive to the performance of an LLW disposal facility.

The intruder assessment required by 10 CFR Part 61 (see Chapter 4.0) often mitigates the need to consider low frequency disruptive events by bounding the projected dose impacts that could result from disruptive events. An intruder assessment typically bounds the dose from a disruptive event because less dilution and dispersion are expected in an LLW intruder assessment, and the resultant contact with disturbed waste is more direct than what would be expected for low-frequency natural events, which are typically highly energetic (e.g., a volcanic event). Thus, in many cases, the dose to the intruder may bound the impact from unlikely disruptive events. However, if licensees do not estimate the impact to intruders from the generic scenarios (i.e., they use site-specific scenarios), then they will need to evaluate the impact from unlikely disruptive events, which may drive overall performance. Site-specific intruder scenarios that have limited to no direct disturbance of the waste or limited exposure times will commonly produce lower doses to intruders than those that result from disruptive processes or the generic intruder scenarios. The argument that disruptive processes and events do not need to be considered because an intruder assessment was performed may be invalid when site-specific intruder scenarios are used.

Licensees should use a screening process to determine which processes and events need to be considered in the site stability assessment. Processes and events can be screened out of the assessment based on probability, consequence, risk-based criteria (i.e., the combination of probability and consequence), or regulation (see Section 2.5.3.1.2). Screening based on probability typically establishes a cutoff frequency below which disruptive events are not considered further. This cutoff frequency is expected to be higher for LLW analyses compared to HLW analyses because the consequences of disruptive events are likely to be less severe for LLW than for HLW. Screening of disruptive processes and events based on consequence or risk should consider the integrated effect of the process or event. For example, the growth of trees and eventual tree fall on a cover may not significantly impact the stability of the cover from the event itself, but it may create a depression to initiate gullying. In many cases, there is limited experience and, therefore, limited understanding of potential combined and synergistic phenomena. Although the uncertainty associated with combined phenomena cannot be reduced, the use of multiple independent peer reviewers with diverse backgrounds can reduce the likelihood of omitting important combined effects.

5.1.1 Natural Processes

The near-surface environment is continually evolving, influenced by processes as well as by discrete events. The processes and events may interact, compounding or reducing the effects observed from the individual phenomena. The systems are dynamic and may include positive or negative feedback effects. The phenomena most likely to impact the stability of a disposal facility and, therefore, the demonstration of compliance with the performance objectives of 10 CFR Part 61, Subpart C, will be specific to the site and design. This section covers different types of natural processes that can impact the stability of a disposal facility. Geomorphic (e.g., mass wasting, erosion, slumping, land sliding, or weathering), tectonic (e.g., faulting, folding, seismic activity, or volcanism), biologic (e.g., animal or plant intrusion), or other processes and events (e.g., flood, fire, or extreme weather) may impact the future stability of waste disposal facilities. Processes and events may impact stability either directly (e.g., a large flood) or indirectly (e.g., a fire that disturbs vegetation, leading to increased erosion). In general,

geomorphic processes will be the most likely stressors of long-term performance for most facilities.

Geomorphology is multidisciplinary, combining aspects of hydrology, climatology, ecology, and geology. Landform evolution is the sequence of processes and events that shape a given landscape. Landforms are built through tectonic, volcanic, and sedimentation processes and are reduced through processes such as erosion and mass wasting. Evolution of landforms is associated with the balance between additive and subtractive processes. For long-term assessments, a site stability assessment of geomorphological processes will likely entail integration of the effects of many of the processes described in the following sections.

Site characterization and selection play an integral role in assessing natural processes that may impact the stability of the disposal facility. For new sites, applicants should characterize the site to define the geomorphic, tectonic, and other hazards that may significantly impact site stability. For existing sites that want to accept new waste, additional site characterization may be needed if the waste is materially different (e.g., more long lived) from previously accepted waste. Selection of a site with limited disruptive processes and events can greatly simplify the scope of a site stability assessment.

5.1.1.1 Geomorphic Processes

Geomorphic processes, such as mass wasting, slumping, land sliding, erosion, sinkhole formation, and weathering, may impact the stability of an LLW disposal facility. A description of the geomorphology of the site, including USGS topographic maps that emphasize pertinent local geomorphic features, may be useful to identify processes, such as erosion, that may affect long-term site stability. Erosion and weathering are especially applicable to LLW disposal. Fluvial (at more humid sites) and eolian (at more arid sites) erosion of engineered covers are common disturbance mechanisms for disposal of long-lived waste. To mitigate certain geologic processes, licensees may achieve long-term erosion protection with robust engineered designs using a system of erosion controls and durable, appropriately sized rock (NRC, 2002b). Long-term erosion protection lasts for potentially thousands to a few tens of thousands of years. To achieve long-term protection, the erosion controls should be independent and redundant, and the rock used for erosion protection should be resistant to weathering. Avoidance of steep slopes and the use of low-relief designs can reduce the impact of some geologic processes (NRC, 2002b). As discussed in Section 5.3, licensees may consider engineered barriers to mitigate the impact of geologic processes.

The NRC staff expects erosive processes (fluvial and eolian) to be the most likely disruptive processes to impact the long-term stability of most disposal facilities. Therefore, the staff recommends that licensees develop robust erosion control designs using durable materials, as discussed in Section 5.3. Robust erosion control designs are usually developed based on the consideration of low-probability events, such as the PMP and corresponding PMF (NRC, 2002b). The PMF is defined as the hypothetical flood that is the most severe flood reasonably possible, based on (1) comprehensive hydrometeorological application of the PMP and (2) other hydrologic factors favorable for maximum flood runoff, such as sequential storms and snowmelt. The return period of the PMP has been debated in technical literature because there are few direct observational data. A comparison of PMF peaks with historic floods in the United States yielded nine floods that exceeded 80 percent of the PMF but none that exceeded the estimated PMF (Bullard, 1986). Similar data compiled for approximately 20,000 gauging-station-years did not yield a flood that exceeded the calculated PMF

(Crippen and Bue, 1977). Paleo-flood data may be available to provide confidence that the computed PMF estimate is appropriately conservative. The return period associated with the PMP has been estimated from a statistical analysis of historical data at approximately 60,000 years (Koutsoyiannis, 1999). Koutsoyiannis (1999) cites the work of others that estimated the return period of the PMP to be from 200,000 to 1 x 10^9 years. The NRC staff considers the PMP to be a very unlikely event with respect to LLW disposal and believes that the PMP is appropriate to use to produce a conservative design.

The approach used by a licensee to assess geomorphic processes will likely involve three steps: (1) identification of past geomorphic processes and estimation of their rates from stratigraphic and geomorphic records, (2) identification of present geomorphic processes and estimation of their rates from historic records and field observations (roughly the last 100 years), and (3) prediction of future processes and rates, accounting for uncertainty. NUREG/CR-3276RW, "Geomorphic Controls on the Management of Nuclear Waste," issued 1983 (NRC, 1983c), identifies over 25 geomorphic processes that can create hazards at waste disposal sites. Typically, only a few of the hazards may be of concern at a particular site. A geomorphic hazard assessment will typically include the site and the surrounding region. An assessment of the surrounding region may provide information on processes that are occurring beyond the boundaries of the site but that may eventually extend onto the site. For example, the migration of a river channel near a disposal site may indicate that the river channel may eventually migrate onto the disposal site. In addition, the presence of a dam upstream from a facility may influence the frequency or likelihood of flooding or result in flooding if the dam were to fail. Therefore, licensees should use information from the surrounding region to help estimate rates and magnitudes of processes that are not currently occurring at the disposal site.

A site does not necessarily need to experience mass loss; a site may be in an accreting environment. Accreting environments are more favorable to long-term stability as nature enhances the isolation of the waste from the environment by covering the disposal facility over time. However, certain depositional processes may degrade the performance of some types of erosion controls (such as diversion channels). Licensees may use diversion channels at a site in an accreting environment to limit fluvial erosion and ensure site stability. Even in a net depositional environment, the site may experience periods of both deposition and erosion.

5.1.1.2 Tectonic Processes

Tectonic processes that can lead to long-term processes or events such as faulting, seismicity, or volcanism, may impact the stability of an LLW disposal facility. A variety of factors should make tectonic processes generally less significant than geomorphic processes. First, licensees should use the site selection and characterization process to avoid areas of high tectonic activity. Second, LLW facilities are generally passive buried systems that are more susceptible to gradual deterioration from degradation and weathering than to discrete failure due to unlikely tectonic events. An exception would be facilities that rely significantly on engineered barriers to meet the performance objectives, especially resistive engineered barriers. A resistive engineered barrier is one that is reliant on physical properties to prevent liquid flow or transport, such as a geomembrane or cementitious barrier. A significant seismic event could result in cracking or tearing that would impact the ability of the engineered barrier to limit water flow. The durability of the engineered barriers would need to be assessed, as discussed in Section 5.3.

Licensees should assess the potential for subsurface geologic processes, such as faulting or seismic activity, to affect the site and proposed disposal facility. They should review historically

recorded seismic information including recurrence intervals, magnitudes, and durations, as well as factors that contribute to peak ground acceleration, such as underlying geologic structures (e.g., active faults) and the stratigraphy and lithology of the site. Licensees should evaluate the predicted effects of seismic events on waste isolation. Reviewers should evaluate features of the disposal system designed to mitigate the effects of seismic events. NUREG-1804 contains additional guidance on reviewing information related to seismic events in waste disposal.

5.1.1.3 Other Disruptive Processes

Other natural processes and events can significantly impact the stability of an LLW disposal facility. These may include, but are not limited to, climatic processes and climate change, biological processes, and fires.

Climate and climate change have the potential to impact the stability of the disposal site. This section discusses climate change in terms of the natural cycling of climate, not human-induced climate change. Section 5.1.2 discusses anthropogenic impacts on climate. The impact of natural cycling of the climate could range from minimal to severe, depending on the location of the disposal facility. The type of waste that is disposed can influence how disruption from climate and climate change should be evaluated in the assessment. Climate change can result in changes to the magnitude and duration of precipitation events, modification of the number of freeze-thaw cycles, or other effects.

For conventional LLW inventory, licensees should limit unnecessary speculation about severe climate change, such as glaciation. This type of event is envisioned as broadly disrupting the disposal site region. For conventional LLW, the hazard from the inventory remaining after 500 years is expected to be relatively low. The dispersion and dilution associated with the glaciation would further reduce doses. The impact from LLW that has been disturbed by severe climate change is likely to be small compared to the impacts on humans from the event itself (i.e., the non-radiological consequences should outweigh the radiological consequences). However, it may be useful to communicate the sensitivity of the LLW disposal facility performance to less severe changes in climate, such as changes to annual average precipitation and temperatures. Robustness and resiliency of the disposal system design are important considerations in the safety case.

For LLW inventories containing long-lived isotopes, the hazard from the inventory remaining after 500 years may be significant. Therefore, licensees may need to evaluate the impact of climate change on the disposal site with reasonably conservative site scenarios. Climate change may not be gradual and continuous, as evidenced by a wide variety of published information (Alley, 2000; Fricke et al., 1998; Jenkyns, 2003). The rate of change may be as important to establishing the risk as the magnitude of change. As previously discussed in Section 2.5.3, the site suitability requirements do not allow for selection of sites where instability precludes defensible modeling or assessment of long-term impacts.

Licensees should examine plausible site scenarios for site evolution in the site stability assessment. For example, the erosion rate under a future, wetter climate may be significantly greater than at the present day. It should be noted that even small erosion rates can amount to substantial thicknesses of material eroded in the long term (e.g., a 0.5 millimeter/yr (0.02 inch/yr) rate will reduce a cover's thickness by half a meter in 1,000 years). The NRC staff recommends that a licensee not change the magnitude of the PMP and PMF for a wetter climate, based on the conservatisms associated with the estimation of the PMP and the PMF.

However, some sensitivity analyses could be performed for relatively large precipitation events occurring during a reasonably foreseeable wetter climate state because of their potential to cause significant erosion and to gain insights into potential future impacts from erosion.

Biological processes can impact the stability of the waste and the stability of the disposal system. Biodegradation can affect the stability of waste containers and result in deterioration of the wasteform, which can result in instability of the disposal system. The BTP on wasteform test methods describes test procedures and criteria to evaluate wasteform stability (NRC, 1991b). Biointrusion into the waste, or into engineered covers over the waste, can increase infiltration to the waste and contribute to the instability of the disposal system. Plants and animals may create pathways in resistive barriers. The disruptive effects of biological processes tend to be gradual but can be significant in some systems. Evapotranspiration or water balance covers can depend on vegetation to eliminate water. Therefore, licensees should consider the impact of invasive species that have different root depths or growing patterns and water usage on the site stability and performance assessments. Climate change may affect pedogenesis (i.e., soil development processes) and the type of cover vegetation. Fires, severe storms, and ecological succession due to changing temperature and precipitation could influence the ability of the vegetation to resist wind and water erosion and maintain a sufficient depth of cover over the waste.

The performance of LLW facilities is not usually sensitive to the impact of fires, such as brush fires. However, as mentioned above, vegetation can play an important role in reducing soil loss and maintaining site stability for slopes and covers. Under extreme fire conditions, fires may change the physical properties of the site soils and other cover materials. Strong fires have been shown to impact the durability of select rock types (e.g., silica-based minerals deteriorate after exposure to high temperatures) (Dorn, 2003). However, researchers at the DOE's Hanford Site directly evaluated the impact of fire on an engineered cover system and found the impact to be minimal for their design (Ward et al., 2009).

5.1.2 Anthropogenic Processes

Human activity may influence site stability both directly and indirectly. An example of a direct influence is construction of a dwelling on the disposal facility. An example of an indirect influence is acid rain generation, which could impact the durability of erosion control materials. A licensee uses the intruder assessment to examine the direct impact to an inadvertent intruder who unknowingly disturbs the waste disposal facility. A licensee does not need to evaluate indirect impacts associated with human activity at the disposal site. The direct impacts to the inadvertent intruder should bound the impacts to the public (offsite) from anthropogenic disruption of the disposal facility because there is comparatively less dispersion and dilution in the assessment of impacts to the intruder. However, if site-specific intruder scenarios are used that result in relatively limited exposures, then indirect processes may dominate the risk and should be examined. A licensee does not need to consider anthropogenic climate change but should consider natural cycling of climates if necessitated by the timeframe for their waste disposal actions. Chapter 4.0 provides guidance for performing the intruder assessment.

5.1.3 Subsidence and Differential Settlement

Subsidence and differential settlement are very important because instability of the disposal site can result in unacceptable releases, regardless of the overall stability of the disposal facility. Differential settlement was a key failure mode in earlier LLW disposal sites, before the

promulgation of 10 CFR Part 61 (NRC, 2007b). Subsidence and differential settlement can lead to unacceptable releases because the processes can adversely affect barrier performance (e.g., cracking of an engineered cover, creating a depression in the cover resulting in localized ponding).

Subsidence and differential settlement can result from a variety of different processes and events such as the following:

- excessive void space in the waste or backfill
- lack of compaction or improper compaction of waste, backfill, engineered or natural materials
- degradation of wasteforms, engineered barriers, and other structures
- improper waste emplacement
- alteration of natural materials
- excessive or uneven loading of the disposal site or engineered cover
- interaction of the disposal system with water

Some designs may be more susceptible to subsidence and differential settlement than others (e.g., designs that combine robust cementitious barriers with unconsolidated and uncompacted natural materials may be particularly sensitive). NUREG-1199 and NUREG-1200 cover subsidence and differential settlement in detail and, therefore, additional guidance is not provided in this document. These documents detail relevant site characteristics, construction and operations phase data, experiment and test data, modeling, and remedial actions.

5.2 Technical Analyses

Technical analyses will be used to complete the site stability assessment. The site stability assessment should evaluate the stability of the wasteform, the disposal facility including waste containers and other engineered barriers, and the site. The specific waste, the complexity of the site and facility, and the hazard being mitigated will dictate the level of detail and type of assessment used. Licensees may use either probabilistic or deterministic analyses as long as uncertainty and variability are assessed.

5.2.1 Available Tools and Codes

Tools and codes[2] are available to licensees to support development of a site stability assessment. Recent advances have resulted in an array of technologies to facilitate analysis of the near-surface environment. Licensees are encouraged to use the best available information for their assessments. The field of geomorphology has evolved significantly over the last half century, and numerous technologies are available today to facilitate completion of a site stability assessment (Kondolf and Piégay, 2005). Although most tools are focused on near-term and small-scale evaluations, tools and techniques have been developed to look at the longer-term and a larger scale. Tools such as optical dating, image analysis, pollen analysis, paleomagnetic dating, oxygen isotopes, lithologic analysis, vegetation surveys, tree-ring analysis, archives, and

[2] The NRC does not endorse or recommend any specific code or model for site stability analysis. Licensees are free to select and justify the code or model for their application. Regardless of the code selected, it is important that the code meet all applicable QA requirements.

radioisotopic dating have been used to study the evolution of the near-surface environment (Anderson and Anderson, 2010).

Walter and Dubreuilh (2007) evaluated computational approaches used to simulate engineered cover performance and degradation. They evaluated 21 computer codes, which they categorized as hydrologic codes, erosion codes, and miscellaneous codes. The erosion codes were further categorized as generalized erosion codes, localized erosion codes, and mass wasting codes. The generalized erosion codes generate estimates of the average soil loss due to water or wind erosion from a plot of land. The localized erosion codes simulate soil loss at specific locations, and some can simulate landscape evolution that may be important to long-term site stability. Table 5-2 summarizes the erosion and mass wasting computer codes, including their applicability and limitations for long-term site stability analysis (Walter and Dubreuilh, 2007). The table does not represent an exhaustive list of available computer codes; however, it may be useful to licensees to identify the types of tools available and their limitations.

The University of Colorado and partners have created the Community Surface Dynamics Modeling System (CSDMS), which provides (1) a modular modeling environment capable of significantly advancing fundamental earth-system science and (2) fully functional and useful repositories for models, supporting data, and other products for educational and knowledge transfer use (Syvitski et al., 2011). The CSDMS repository contains more than 160 models and tools. The system includes 66 codes and 41 tools associated with terrestrial modeling. Landscape evolution (CHILD, SIBERIA, Caesar, Erode, GOLEM, MARSSIM, and WILSIM), eolian transport (Eolian Dune Model), and cryosphere (GC2D, ISGR, Ice Ages) codes are included.

5.2.2 Approaches for Analyses

Licensees may use a variety of technical analyses to perform a site stability assessment. Because of the site-, facility-, and waste-specific nature of a site stability assessment, the tools and techniques used may differ considerably from site to site.

The approaches may be design-based, model-based, or a combination of the two. Modeling may play a more significant role in analysis of longer timeframes, because of the limited data to support long-term performance of stability designs. Iterative analysis may be necessary. It is important that licensees provide a technical basis for the site stability assessment. The technical basis will likely be more extensive for analysis of longer timeframes than for shorter timeframes. The technical basis should be provided for the performance of engineered barriers. Model support is essential for the site stability assessment. Licensees should develop model support throughout the process, although the largest effort (comparable to model validation) is usually at the end of the process. Model validation in the traditional usage is not possible for analyses over long timeframes.

Table 5-2 Summary of Applicability and Limitations of Erosion and Mass Wasting Codes for Long-Term Performance Assessment

Code	Process Representation	Calculation of Cover Loss and Topographic Change
RUSLE	Empirically based using correlations from test plots, requires empirical data to simulate future cover conditions	Computes areally averaged soil loss, independent calculations required to compute changes in cover thickness
EPIC	Empirically based using correlations from test plots, requires empirical data to simulate future cover conditions, but more versatile than RUSLE by calculating soil water balance	Computes areally averaged soil loss, independent calculations required to compute changes in cover thickness
WEPP	Physics-based approach to erosion allows flexibility in representing future soil cover conditions based on fundamental properties	Computes areally averaged soil loss, for multiple hill slopes, independent calculations required to compute changes in cover thickness
EUROSEM	Physics-based approach applied over two-dimensional topographic surface allows representation of complex topography including future changes in topography	Computes distributed soil loss within model domain with a fixed topography, independent calculations required to compute change in surface elevation and cover thickness
LISEM-Gullies	Empirically based approach to soil loss and gully formation over two-dimensional topographic surface but limited to single storm events	Calculates gully formation and associated changes in topography during single storm events, multiple simulations required to evaluate long-term performance
SIBERIA/CHILD	Physics-based approach applied over two-dimensional topographic surface allows representation of complex topography including future changes in topography	Calculates changes in elevation and topography based on soil loss and deposition, including soil creep and slumping
WESS	Empirically based wind erosion code with limited process documentation	Computes areally averaged soil loss, independent calculations required to compute changes in cover thickness
WEPS	Physics-based wind erosion code with ability to represent topographic and wind-break effects	Computes areally distributed soil loss, independent calculations required to compute changes in cover thickness
LISA	Simple, physics-based calculation of sliding potential with stochastic output	Computes sliding potential, but not topographic change
DLISA	Simple, deterministic, physics-based calculation of sliding potential	Computes sliding potential, but not topographic change

The authors of the codes listed can be found in Walter and Dubreuilh (2007).

A licensee's site stability assessment should include the following steps:

(1) Site description — Describe the features of the site, facility, and waste associated with stability. At a minimum, licensees must provide information required by 10 CFR Part 61, Subpart D, "Technical Requirements for Land Disposal Facilities." As previously discussed, 10 CFR 61.50(a)(5)(ii) and 10 CFR 61.50(a)(5)(iii) provide specific requirements with respect to tectonic and geologic processes. The site description may include geologic and geomorphic characteristics that contribute to site stability. The site description may also include geologic, geomorphic, and tectonic hazard assessments.

(2) Initial radiological risk screening — It may be possible to demonstrate with a conservative evaluation that release of radioactive material due to potential instability will not result in sufficient radiological risk (i.e., 10 CFR 61.41 and 10 CFR 61.42 performance objectives can still be met). If so, a site stability assessment may be limited to providing a screening assessment. In other words, a licensee can satisfy 10 CFR 61.44 if its screening assessment provides reasonable assurance that 10 CFR 61.41 and 10 CFR 61.42 will be met under disturbed conditions. Licensees should select receptors and site scenarios consistent with the guidance in Sections 2.5.4.2 and 4.2. The screening assessment should be sufficiently conservative to account for uncertainty.

(3) Process and event screening — Perform screening of disruptive processes and events (consistent with guidance in Section 2.5). Processes and events can be screened out of the assessment based on probability, consequence (as indicated in Step 2), or risk-based criteria. Risk-based criteria combine probability and consequence. Screening of disruptive processes and events should consider the integrated effects of the processes or events, as discussed in Section 5.1.

(4) Define scope of the assessment — The FEPs that cannot be screened out from the site stability assessment should be used to define the scope. Licensees can compile into site scenarios the processes and events that cannot be screened out. They can inform the scope by considering the operating experience of analogous sites and facilities. If there is uncertainty about whether a process or event should be included, the NRC staff recommends erring on the side of inclusion. It is generally more difficult to add processes and events later in the analysis process than it is to remove them from the assessment.

(5) Characterize information — Use the site description to determine what data are available to complete the assessment and to identify the significant sources of uncertainty. The type of information needed is likely to depend on the timeframe being analyzed. Examination of the available information (e.g., the characteristics of the waste, disposal facility, and site), will inform the approach that should be used for the assessment (e.g., model-based or design-based). The purpose of this step is to determine how much information is available to support the assessment completed in Step 6.

(6) Perform assessment — The assessment may be performed with a model-based approach (generally for long-lived wastes), a design-based approach (for short-lived or long-lived wastes), or a combination of the two. The NRC staff expects that licensees

may use modeling to assess and develop designs. The steps provided below may be completed in a different sequence. However, regardless of the order, most assessments will include the following steps:

a. Design-Based:

i. Define the design objectives.
ii. Develop or select the design.
iii. Document and provide the basis for assumptions.
iv. Characterize or parameterize the design.
v. Assess the expected performance of the design.
vi. Provide support (technical basis) for the design.
vii. Iterate, if necessary.

b. Model-Based:

i. Define the model objectives.
ii. Develop or select the conceptual model.
iii. Document and provide the basis for assumptions.
iv. Develop the numerical model.
v. Parameterize the model.
vi. Calibrate the model.
vii. Verify the model.
viii. Characterize uncertainty.
ix. Provide model support.
x. Iterate, if necessary.

The elements listed above for design-based or model-based approaches are not unique to site stability assessment for LLW disposal. Additional guidance is available, such as Chapter 2.0 of this document and NUREG-1573. Licensees may mix elements from design-based and model-based approaches. For example, Appendix P to NUREG-1757 describes how to risk-inform the design-based approach. Risk-informing the design-based approach involves performing technical analyses, such as sensitivity analyses, to test the robustness of the design to unanticipated events and processes. Technical basis for engineered barrier performance must be provided in either the design-based or model-based approach.

(7) Integrate — The site stability assessment will likely need to be integrated with the performance assessment required by 10 CFR 61.41 and the intruder assessment required by 10 CFR 61.42. If stability can be ensured, then the need for integration is minimal. However, if instability is projected, then licensees will need to evaluate the significance of that instability in the context of the performance and intruder assessments. For example, degradation of an engineered cover from erosion may lead to increased water infiltration to waste in a disposal cell. The releases estimated in the performance assessment should include nominal performance, as well as releases attributed to instability of the waste, the disposal site, and the disposal facility. When the site stability assessment is based on both design and modeling, it is also important for a

licensee to document that the important features of the design have been integrated into the stability modeling.

(8) Iterate (if necessary) — It is likely that some amount of iteration will be necessary in the site stability assessment. There are likely to be many sources of uncertainty, including some that were unanticipated during the initial design or modeling process.

5.2.2.1 Design-Based Approach

A licensee may use a design-based approach to demonstrate site stability. The design-based approach has been used successfully for the management of uranium mill tailings, though the experience base is only tens of decades. Section 5.3.2 presents information on analogs that suggests longer-term performance should be achieved. Appendix E provides an example of the use of the design-based approach to develop erosion protection measures for the management of uranium mill tailings. Section 5.3 contains guidance on the use of engineered barriers for the site stability assessment. A licensee may use the guidance in Section 5.3, or alternatively, may develop its own approach. Designs may use risk information to mitigate long-term uncertainties. Conservative, robust designs may be beneficial irrespective of projected risk information. The design-based approach may be supplemented with sensitivity analyses to better characterize and understand uncertainties (see NUREG-1757, Volume 2, Revision 2, Sections 3.5.2 and 3.5.3). The following steps outline the design-based approach:

(1) *Define the design objectives*—The primary design objectives are the time over which stability must be achieved and what instability, if any, is tolerable to ensure compliance with performance objectives in 10 CFR Part 61, Subpart C. Waste characteristics and the risk to the public, including an inadvertent intruder, will be important considerations. For example, a licensee that wishes to dispose of short-lived waste with limited quantities of long-lived waste may achieve protection of public health and safety by ensuring waste stability for 500 years. The licensee may be able to demonstrate that insufficient radioactivity remains in the disposal system after 500 years to pose an unacceptable risk to a member of the public.

(2) *Develop or select the design*—After defining the design objectives, the licensee should develop the design or select a design that has previously been demonstrated to achieve the objectives. Previously used designs are likely to require modifications for site-specific application. Section 5.3 presents information on rock durability, for example, which is strongly influenced by local environmental conditions. Section 5.3 provides an acceptable process for selecting erosion protection materials.

(3) *Document and provide the basis for assumptions*—Assumptions are a common part of design and analysis. Licensees should document and provide a basis for assumptions that they have used in developing or selecting the design. Regulators should review the documentation and determine whether the technical bases for the assumptions are adequate. Assumptions should be internally consistent.

(4) *Characterize or parameterize the design*—After a licensee has developed or selected a design, it may need information to characterize the features of the design or provide parameters that will represent the performance of the design. For example, a design may use natural and engineered materials. The licensee may need to consider the weathering or degradation rate of the materials. The licensee may need to perform and

document experiments used to develop the design information. Documentation should be sufficient to allow for independent verification of the results. If a licensee uses information from generic sources to parameterize the design, it should demonstrate that the information is relevant to the site-specific application.

(5) *Assess the expected performance of the design*—Verification of the design involves technical assessment of the performance of the design over the range of disruptive processes and events expected to influence the design. A licensee should clearly identify over what range of conditions the design is expected to perform and over what range of conditions the design may fail. This information is useful to regulators and other stakeholders if future conditions occur that may have been unforeseen during the design process. A licensee or regulator may use independent confirmatory analyses or modeling to verify the design.

(6) *Provide support for the design*—A licensee must provide adequate support for the design before its implementation. Types of support for engineered designs are similar to support provided for models, as discussed in Section 2.2.4 (e.g., tests, experiments, field studies, and analogs). In addition, support for designs may also include the experience base for the design, such as the number of locations where it has been used and for how long. Unique or novel designs, which may be necessary, are not to be discouraged but they will have additional uncertainty as to whether they will achieve the design goals. A licensee should provide additional support for the performance of a novel design compared to what would be provided for a standard design. The support for the design at the time of implementation will not have information from the actual performance of the design. Therefore, the design process may need to be iterative as monitoring and performance confirmation data are collected.

NUREG-1623, which was developed to provide methods, guidelines, and procedures that the NRC staff considers to be acceptable for designing erosion protection at uranium mill tailings sites, describes an example of a design-based approach to erosion protection. In addition, Appendix P to NUREG-1757 describes how to risk-inform the design-based approach. These design approaches are based on technical procedures and design parameters that are widely used in the engineering community and by other Federal agencies. They have been applied at disposal sites.

5.2.2.2 Model-Based Approach

A licensee may also use a model-based approach for performing the site stability assessment. The model-based approach is used to evaluate the FEPs that may affect the stability of the site and determine if site stability requirements can be met. The staff expects that the model-based approach may be iterative, and that the understanding of the significance of FEPs may evolve over time.

The steps involved in the model-based approach to site stability assessment in Section 5.2.2 are similar to the steps to develop a performance assessment. Therefore, a licensee should review the guidance in Chapter 2.0 when using a model-based approach to site stability assessment. Appendix E provides an example of using a model-based approach to site stability assessment. As discussed in Section 2.2.4 and Section 5.3.2, model support is essential to any type of model-based approach. The NRC staff strongly recommends using natural analogs and other forms of evidence of the long-term stability of the site.

Tools and codes are available to licensees to support site stability modeling. The field of geomorphology has evolved considerably over the last half century and many technologies are available today to facilitate site stability analysis, including various programs such as hydrologic codes, erosion codes, and landscape evolution codes (e.g., CHILD and SIBERIA). Although the uncertainties associated with landscape modeling are often large, licensees can use modeling to gain insights and perform site stability assessments.

Licensees should consider the following technical information when developing the scope of a model-based approach to site stability assessment:

- Changes to nearby stream beds and river channels in unconsolidated material may impact landform evolution.
- Given sufficient precipitation, large rain events, and time, the watershed of an area can change considerably. Specific parts of a facility may end up close to stream channels with an increased gradient, leading to accelerated erosion.
- Drainage patterns may change. For example, if variations in climate circulation patterns are great enough, regional drainage patterns may change.
- Erosion and deposition rates may vary spatially and temporally across the site.
- Long-term erosion often concentrates in gullies, which do not uniformly erode over their entire length. Peak erosion depth may translate into a total breach of a portion of the disposal facility.
- Evidence of the past natural history of the area, or from an appropriate natural analog, may be an indicator of future surface geomorphic processes.
- Pedogenic processes, biotic activities, and bioturbation may impede or accelerate surface geomorphic processes. Thick root systems from certain plants are known to greatly reduce erosion; however, it is difficult to rely on the continuous presence of a specific plant that might be needed to ensure stability over longer times.
- Different factors could influence flora, such as drought, fire, disease, fungi, and insects. Observed erosion rates may be tied to the presence of flora.
- Changes in the water chemistry and other environmental conditions may alter the properties of the rock responsible for physical and chemical stability.

5.2.3 Uncertainty

Uncertainty is inherent in the site stability assessment and should be accounted for. Probabilistic assessment techniques are generally more amenable to accounting for uncertainty (see Section 2.7.4), although the NRC staff does not prescribe a particular technique. The site stability assessment should consider the types of uncertainty (e.g., data, model) described in Section 2.2.3. A variety of techniques are available for handling uncertainty in multimedia environmental modeling (NRC, 2004c).

Site stability assessment may involve uncertainties that can be challenging to characterize and understand. Some near-surface processes can display sensitivity to initial conditions, such as the sensitivity of erosion rates to initial topography. There can also be feedback, both positive and negative, that can result in complex responses (Pelletier, 2008). The complex responses in turn translate into uncertainty in interpreting characterization and observational data. The complex responses can be difficult to understand; therefore, they can be difficult to implement in numerical modeling, which results in model uncertainty. Environmental systems can also exhibit sensitivity to the initial conditions as well as the pathway taken to arrive at the initial conditions (i.e., hysteretic phenomena). As discussed in Section 2.2.3, one approach to mitigating the impact of uncertainty is to use conservatism. For example, using the PMP/PMF for the design-based approach to erosion protection can result in a robust design.

Incorporation of information at variable temporal and spatial scales is a challenge in a site stability assessment. Some processes or events may span orders of magnitude in temporal and spatial scales. The key to understanding complex systems often lies in understanding how the processes on different scales influence each other. Licensees may need to coarsen or upscale detailed information at finer scales, to perform numerical modeling at coarser scales. If upscaling is used, it is important to perform physical or numerical experiments that demonstrate that essential information is preserved in the upscaling process. In addition, it is important that data are representative of the scale and conditions being simulated. For example, a single point measurement of soil moisture content may not be representative of (1) a sitewide value, (2) a more global value, or (3) the distribution of local values needed in a site stability assessment.

Licensees may take a variety of actions if site stability is not demonstrated. They could make design changes to achieve stability, or they may conclude that the site does not have adequate stability to be suitable for the type of waste proposed for disposal at the site. Inventory limits may be used to mitigate the risks and uncertainties associated with site instability. Or licensees may refine their modeling (if technically justifiable), and the refined model may demonstrate that FEPs that were previously believed to be significant do not have significant impacts.

5.3 Engineered Barriers for Site Stability

This section describes the information that a licensee should provide and a reviewer should evaluate with respect to the use of engineered barriers for site stability. Engineered barriers are likely to be used for erosion control or may be used for other reasons such as to mitigate the impact of disruptive processes and events.

An engineered cover is frequently used for physical stabilization of the site (10 CFR 61.44). The engineered barrier systems may include liners, covers or caps, or lateral barriers or walls, alone or in combination. These systems may use a variety of natural material such as aggregates, soil, or clay, and synthetic, cementitious, and bituminous materials including polyethylene, fabrics, mortar, and asphalt. The requirements and the type of engineered barrier system that a licensee chooses depend on the waste type. An engineered surface barrier, or cover, is the most used barrier and is often considered one of the most important components. The NRC staff expects that the main type of engineered barrier used for LLW disposal will continue to be engineered covers. Engineered covers can be further classified as conventional (resistive) covers (Section 5.3.2.1) and evapotranspiration (water balance) covers (Section 5.3.2.2). Each of these types of covers may have erosion control functions. Engineered surface covers can be significant barriers because they may contribute to a licensee demonstrating that one or more performance objectives will be met. In addition to providing the stability required in

10 CFR 61.44, engineered covers may contribute to disposal facility performance by minimizing infiltration and slowing degradation of a stabilized wasteform (10 CFR 61.41) and by providing an intruder deterrent (10 CFR 61.42).

5.3.1 Existing Guidance

Existing guidance developed for analogous programs (e.g., uranium mill tailings, decommissioning) applies to the design of engineered barriers for the stability of LLW disposal facilities. The processes and events that may disrupt LLW facilities are essentially identical to those that may impact a uranium mill tailings disposal facility or a decommissioned site. The main exception is that some LLW disposal facilities could contain higher quantities of long-lived waste. Section 5.3.2 discusses considerations for the design of engineered barriers for long-lived waste. Existing guidance focuses on cover design, particularly material durability. The focus of site stability assessment for waste disposal has been on erosion protection.

Engineered barriers are distinct and separate from institutional controls. An engineered barrier could be one of a broad range of barriers with varying degrees of durability, robustness, and isolation capability. Generally, engineered barriers are passive, man-made structures or devices intended to enhance a facility's ability to meet the performance objectives. Engineered barriers are usually designed to inhibit water from contacting waste, limit releases of radionuclides (e.g., through groundwater, biointrusion, or erosion), or mitigate doses to inadvertent intruders. Engineered barriers can serve a variety of functions; therefore, they have markedly different technical considerations and designs.

Section 3.5 of NUREG-1757, Volume 2, provides guidance on a risk-informed, graded approach to the design and evaluation of engineered barriers. NUREG-1573 contains guidance on engineered barriers for application to LLW disposal and a bibliography organized by different topics, including Appendix C on engineered and natural barriers. A broad range of engineered barriers may be used to achieve stability, with varying degrees of durability and robustness. Engineered barriers used for stability should be passive (i.e., they should perform without reliance on active monitoring and maintenance). Licensees should supply an appropriate technical basis demonstrating that engineered barriers provide significant performance.

The following are the main steps in developing and implementing engineered barriers, as described in NUREG-1757:

- Describe the design, features, and functionality of the engineered barriers.
- Provide the technical basis for the performance of the barriers that will allow a licensee to demonstrate that the performance objective will be met, considering the degradation mechanisms and the combined and synergistic effects resulting from the real-world conditions expected for the barriers.
- Describe uncertainty in parameters and models used in the assessment of barrier performance and the design of engineered barriers.
- Demonstrate the suitability of numerical models for the estimation of engineered barrier performance.

- Perform parametric or component sensitivity analysis to identify how much degradation of the engineered barrier would result in noncompliance.

- Provide model support for the engineered barrier performance (e.g., analogs, experiments, engineering calculations to demonstrate reasonableness of the results).

- Provide QA/QC for the data collection, design, construction, and analysis of engineered barriers.

These steps apply to engineered barriers used to support demonstration of compliance with 10 CFR 61.41 and 10 CFR 61.42, as well as for site stability assessment. Model support for engineered barrier performance is essential. Model support, as discussed in Section 2.2.4, can come in many different forms, including but not limited to analogs, laboratory experiments, field experiments, and formal and informal expert judgment. Knowing the basis for expectations that a barrier will perform the desired function is essential to give the licensee and other stakeholders confidence in the future performance of the barrier. For engineered barriers that are estimated to have long-term performance (e.g., thousands of years), licensees should consider natural analogs to provide confidence that the estimates are reasonable. Extrapolating short-term observations to estimate long-term performance can result in a significant uncertainty in long-term barrier performance. Standard approaches implicitly assume that the initial conditions persist; however, the actual application of a barrier may be more appropriately viewed as an evolving component of a larger dynamic system (Waugh and Richardson, 1997). In addition, inaccurate conceptualization of degradation mechanisms and their interactions can be a source of significant error. Adequate model support can help reduce the impact of these uncertainties. NUREG-1757 provides examples of analogs for cement performance (wasteform stability), durability of earthen covers, and riprap durability (site stability). NUREG-1757 also summarizes the degradation mechanisms of common engineered barriers, including engineered covers such as resistive covers, evapotranspiration covers, and erosion control covers. Section 3.5.5 of NUREG-1757 contains a summary of reference information on engineered cover design and performance.

One of the barriers used to ensure stability in waste disposal facilities, especially for near-surface disposal of long-lived waste, is an engineered surface cover designed for erosion protection. As mentioned in Section 5.2.2.1, NUREG-1623 may be useful to licensees because it provides methods, guidelines, and procedures that the NRC staff considers to be acceptable for designing erosion protection at uranium mill tailings sites. The main elements to developing an engineered barrier for erosion protection in waste disposal are the following:

- selection of proper rainfall and flooding events

- selection of appropriate parameters for determining flood discharges

- computation of flood discharges using appropriate and/or conservative methods

- computation of appropriate flood levels and flood forces associated with the design flood discharge

- use of appropriate methods for determining erosion protection needed to resist the design discharge

- selection of a rock type for the riprap layer that will be durable and capable of providing the required erosion protection

- placement of riprap layers in accordance with accepted engineering practice and in accordance with appropriate testing and QA controls

Guidance for the design of erosion protection systems for management of uranium mill tailings is applicable to the stability of LLW disposal facilities and may be used for nearly all types of erosion protection designs. However, for significant quantities of long-lived waste (as discussed in Section 5.3.2), licensees should place greater emphasis on rock durability, and additional considerations may be necessary.

Erosion protection designs may take many different forms. Many designs use a durable rock cover; however, some combine vegetation or employ multiple layers of differing materials (i.e., a composite cover) or rock mulches. The design considerations for an erosion protection cover using rock can be separated into two main areas: (1) appropriate sizing of the rock, and (2) assurance of the durability of the cover materials. NUREG-1623 provides methods for sizing rock for an erosion protection cover (for a design based on the PMP). Licensees should size the rock to ensure that it will stay where it is placed; they should then ensure that the rock will not degrade significantly (i.e., it will stay close to its initial size). To maintain its design size, appropriately sized rock should not experience significant mass loss from weathering or experience significant cracking.

Rock durability is defined as the ability of a material to withstand forces of weathering. Factors that affect rock durability are (1) chemical reactions with water, (2) saturation time, (3) the temperature of the water, (4) scour by sediments, (5) windblown scour, (6) wetting and drying, and (7) freezing and thawing. Chemical weathering and mechanical weathering may reduce the effectiveness of a rock cover for erosion protection. Chemical weathering is generally a slow, continuous process that usually occurs in the presence of water. Mechanical weathering can lead to deterioration of the rock without chemical alteration. The most prevalent mechanical weathering processes are (1) frost action and freeze-thaw activity, (2) salt crystallization, migration, and hydration, (3) water sorption, (4) mineral hydration, (5) wetting and drying cycles, (6) abrasion by wind, water, and mechanical means, and (7) temperature-induced expansion and contraction of mineral grains (NRC, 1982c). Comparing the latter processes with the factors that affect rock durability demonstrates that mechanical weathering is a dominant degradation process. The individual weathering mechanisms are likely to be specific to the rock type selected and the weathering processes at the site. Weathering processes and rates are strongly influenced by climatic conditions. Figure 5-3 shows the macroscale relationship of climatic variables, environments, and rock weathering agents.

As discussed in Appendix P to NUREG-1757, a licensee should evaluate rock durability to provide multiple and complementary lines of evidence and greater confidence in rock durability. The evaluations are for (1) rock durability testing and scoring, (2) absence of adverse minerals and heterogeneities, and (3) evidence of resistance to weathering.

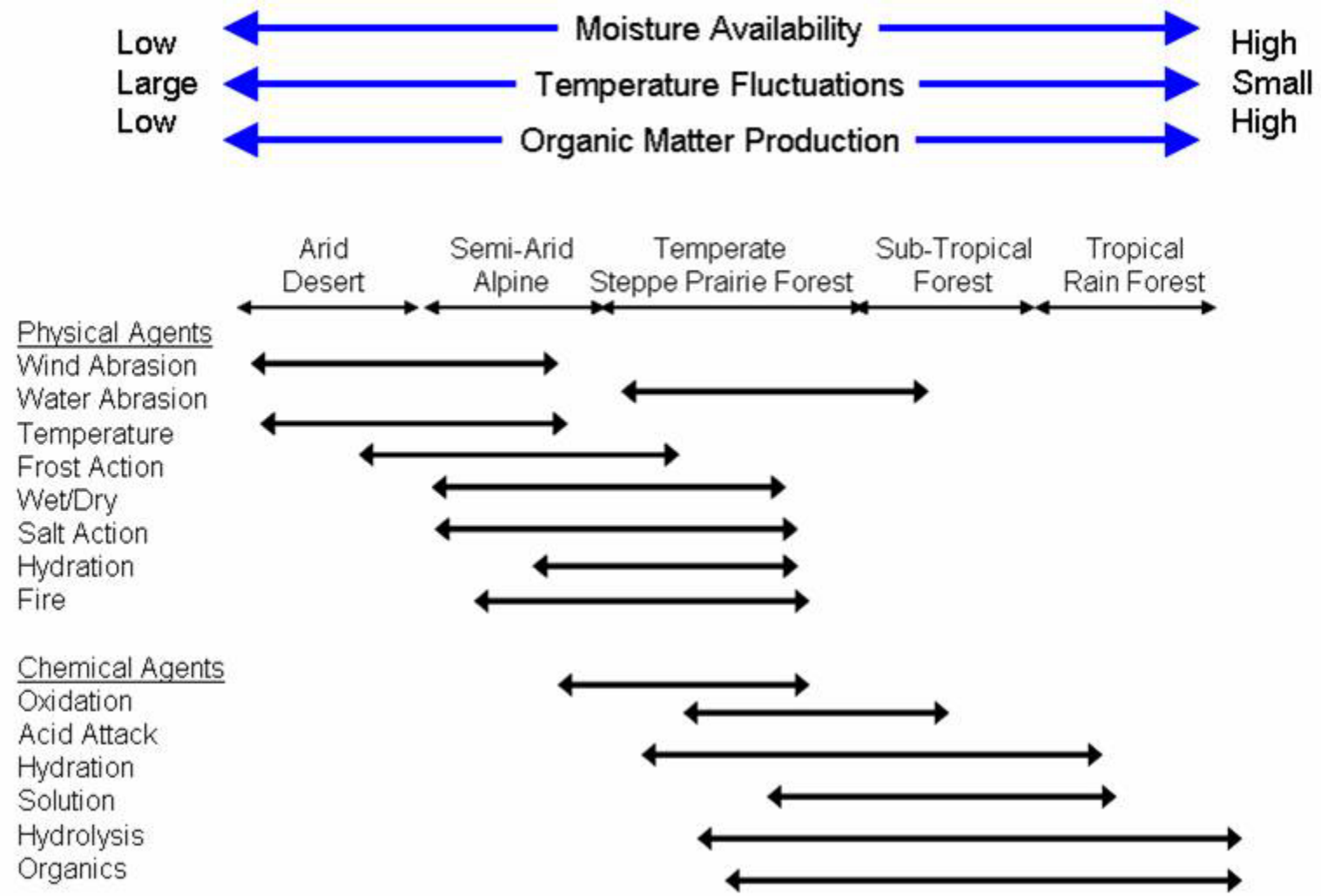


Figure 5-3 Macroscale Relationship between Climatic Variables, Environments, and Rock Weathering Agents: Occurrence of Weathering as a Function of Climate (NUREG/CR-2642)

First, licensees should conduct rock durability testing and scoring. They should test and evaluate potential rock sources to ensure that rock used for erosion protection remains effective for the required timeframe. NUREG-1623 presents a procedure for determining the acceptability of a rock source. Rock durability testing is usually performed using standardized test procedures, such as those developed by ASTM and published and updated in ASTM's *Annual Book of ASTM Standards.* The scoring procedure indicates that rock scores of 80 percent or greater indicate a high-quality rock for most applications (e.g., decommissioning sites for 1,000 years). Long-lived waste would generally require very high-quality rock, higher than necessary for most decommissioning applications.

Second, the absence of adverse minerals and heterogeneities is essential. Licensees should use analyses (such as petrographic analyses) to establish the absence of adverse minerals that could cause rapid degradation of the rock, such as clays, olivine, or calcite cement. If adverse minerals are present, licensees should evaluate their potential effect on the weathering of the rock and on weathering rates.

Third, licensees should provide evidence of resistance to weathering, using direct evidence from the selected rock source whenever possible. For example, weathering rind thickness and alteration of minerals and rock properties from exposures of the weathered rock source can provide insights into the extent and nature of future weathering. Table 5-3 shows characteristics of rocks related to weathering resistance (NRC, 1982c). Some characteristics are favorable with respect to chemical weathering and unfavorable with respect to mechanical weathering, and vice versa. Indirect evidence from natural and archeological analogs should also be used, as discussed in Section 5.3.2.

An engineered cover for LLW disposal may have more than one design goal. For example, in addition to providing site stability, the engineered cover may be used to reduce infiltration. Licensee should ensure that all design goals are achieved, while realizing that some decisions may have competing influences on the design goal. For instance, using large rock may promote site stability but may increase infiltration compared to other erosion protection designs, such as using rock mulch.

5.3.2 Long-Term Considerations

Disposal of long-lived waste introduces complexity and uncertainty with respect to site stability. A disposal facility containing long-lived waste may be exposed to more extreme conditions as well as more cycles of extreme conditions. The performance of the disposal facility will have greater uncertainty because of more limited performance and observational data. A licensee may experience greater difficulty in obtaining relevant information. Because of the increased uncertainty inherent in longer-term predictions, engineered barriers for the stability of long-lived waste will generally be designed with conservative assumptions.

Table 5-3 Weathering Resistance and Susceptibility Related to Weathering Type and Rock Properties

	Chemical Weathering		Physical Weathering	
	Durable	**Nondurable**	**Durable**	**Nondurable**
Mineral Composition	- Uniform mineral composition - High silica content - Low metal ion content (Fe-Mg), low biotite - High orthoclase, Na feldspars - High aluminum ion content	- Mixed/variable mineral comp. - High $CaCO_3$ content - Unstable primary igneous minerals - Low quartz content - High calcic plagioclase - High olivine	- High feldspar content - Calcium plagioclase - low quartz content - $CaCO_3$ - homogeneous composition	- High quartz content - Na plagioclase - Heterogeneous composition
Texture	- Fine grained dense rock - Uniform texture - Crystalline - Clastics - Gneissic	- Coarse-grained igneous - Variable textural features - Schistose	- Fine grained (general) - Uniform texture - Crystalline, tightly packed clastics - Gneissic,Fine grained silicates	- Coarse grained - Variable texture - Schistose - Coarse-grained silicates
Porosity	- Large pore size, low permeability - Free draining - Low internal surface area	- Small pore size, high permeability - Poorly draining - High internal surface area	- Low porosity, freely draining - Low internal surface area - large pore diameter free draining	- High porosity, poorly draining - High internal surface area - Small pore diameter
Bulk Properties	- Low adsorption - High compressive and tensile strength - Fresh rock - Hard	- High adsorption - Low strength - Partially weathered rock - Soft	- Low adsorption - High strength with good elastic properties - Fresh rock - Hard	- High adsorption - Low strength - Partially weathered rock - Soft
Structure	- Strongly cemented, dense grain packing - Siliceous cement - Massive	- Poorly cemented - Calcerous cement - Thin bedded - Fractured cracked - Mixed soluble and insoluble minerals	- Minimal foliation - Clastics - Massive formations - Thick bedded sediments	- Foliated - Fractured, cracked - Thin bedded - Mixed soluble and insoluble minerals
Representative Rock	- Igneous varieties (acidic) - Metamorphics (other than marbles) - Crystalline rocks - Rhyolite, granite, quartzite, gneisses - Granitic gneiss	- Calcerous sedimentary - Poorly cemented sandstone - Slates - Limestones, basic igneous, clay-carbonates - Marble, dolomites - Carbonates, schists	- Fine-grained granites - Some limestones - Diabases, gabbros, some coarse-grained granites - Quartzite (metamorphic) - Strongly cemented sandstone - Slates, granitic gneiss	- Coarse-grained granites - Poorly cemented sandstone - Many basalts - Dolomites, marbles - Soft sedimentary - Schists

Licensees should design engineered barriers for disposal of long-lived waste using the following:

- multiple, independent, and redundant[3] barriers
- conservative approaches
- very high quality and durable materials

5.3.2.1 Conventional (Resistive) Covers

Licensees can use conventional (resistive) covers. One common use of a conventional cover is for erosion protection. Designs for sites with relatively short-lived waste are expected to be simpler than designs for sites with waste that will remain hazardous until or beyond the end of the compliance period. Designs with required performance for the duration of the compliance period and for the performance period should include the use of oversized rock and over-thickened rock layers, use of the PMF, and consideration of more stable structures such as low slopes and blending with local topography. By using multiple, independent, and redundant barriers, licensees can reduce the impact from the unanticipated failure of a single barrier.

The use of multiple barriers for an erosion control system could include one or more of the following types of erosion controls:

- The riprap layer for the top slopes and side slopes of the closure covers could be designed to resist the PMP and PMF.
- The top of the covers could be sloped to drain entirely in one direction, minimizing the flows that would enter gullies that form in areas that are not designed for drainage downstream of the covers.
- The site could be optimally graded to enhance drainage, and diversion channels could be constructed to convey runoff to noncritical locations.
- Downstream gullies could be armored with very large rock to prevent further gullying and nickpoint migration.
- Diversion channels could also be constructed upstream of the covers to divert flows away from the covers or from potential critical locations (e.g., where gullies are more likely to form).

Licensees may need to consider the natural cycling of climates in the design for sites intended for long-lived waste disposal (e.g., DU, GTCC). In some locations (e.g., more northern), glacial and interglacial cycles may result in glacier development and migration over an LLW disposal facility. It is beyond current technology and understanding to design a near-surface facility to withstand such forces. In more northern locations, the assessment should focus on the risks following disruption of the design. Licensees may consider natural analogs to estimate the

[3] Redundancy with respect to long-term engineered barriers is not meant to infer duplicative systems, such as multiple drainage layers in an engineered barrier. Redundancy is meant to refer to redundancy in the protective function that is desired, such as limiting the rate of erosion. For example, low relief design with proper slope configuration can limit erosion rates as well as use of a durable and appropriately sized riprap layer.

amount of waste dispersion associated with those processes, and therefore, the risk. Stylized, conditional dose assessments may also be useful to understand potential performance but not as a measure of future impacts to human health. In more southern geographic locations, the impact of natural cycling of climates may be less severe and may be confined to effects such as increased precipitation and cooler temperatures (e.g., more freeze-thaw cycling).

Licensees should examine the durability of erosion control materials over the range of projected future climate states. The use of the PMP and PMF for erosion protection design may mitigate the need to consider future climate states because these parameters represent maximum events. The PMP approach approximates the maximum rainfall that is physically possible, and the PMF is a hypothetical flood that is the most severe reasonably possible (NUREG-1623, p. 10). Climates that are arid now and more likely to be arid in the future are typically preferable to wetter climates. Low amounts of water reduce chemical and mechanical weathering.

Licensees should consider the following when selecting and justifying the long-term durability of erosion protection materials:

- Selection of highly durable rock—Select only a highly durable rock type with a mineral that is most resistant to chemical weathering, such as quartz. This would favor a metamorphic quartzite or a sedimentary orthoquartzite with a high percentage of quartz grains (99 percent) cemented by quartz. Rock types that can easily alter to clay over the timeframe considered, such as feldspars, should not be used. This may eliminate many rock types. Locally available, highly durable rock is preferable because it has a higher likelihood of being in equilibrium with the disposal environment.

- Selection of a homogeneous rock source—Select a rock unit that will result in riprap pieces that are homogeneous and free of heterogeneities, such as bedding planes, thin shale layers, or joints. Heterogeneities that can allow access of water can contribute significantly to mechanical weathering, such as freeze-thaw.

- Reliance on natural analogs and weathering rate studies—Evaluate natural analogs and obtain applicable weathering rate studies, if available, to justify the durability of the selected rock.

  - Weathering rate studies: Obtain weathering rate study data over relevant timeframes, to the extent practicable, recognizing uncertainties in the studies. Weathering rate data may be estimated from laboratory data complemented with field observations. If laboratory data are generated, a licensee should use caution in extrapolating the data to field performance because mineral weathering does exhibit scale dependence (Drever and Clow, 1995). Weathering rates can also decrease exponentially with time, as the most vigorous attack usually occurs early in the weathering process. Comparative data on natural materials may help with material selection, such as that shown in Table 5-4 (Brookins, 1984) or in Table 6.7 of NUREG/CR-2642, "Long-Term Survivability of Riprap for Armoring Uranium Mill Tailings and Covers: A Literature Review," issued 1982 (NRC, 1982c). Comparative data show the relative durability of different materials in a common test. Various compilations of research on weathering rates have been developed (e.g., Colman and Dethier, 1986). A licensee should use weathering rate data that are representative of the estimated future exposure conditions because weathering rates

can be very sensitive to exposure conditions. For example, one of Cleopatra's Needles (a granite obelisk) survived very well in over 3,000 years of exposure in arid conditions in Egypt but weathered heavily in under 100 years after being moved to New York City.

- Natural analogs: Use natural analogs to provide confidence in the long-term durability of the materials selected. A licensee should use natural analogs for the specific rock type proposed and from the region of the disposal facility, instead of more distant examples, because local materials with demonstrated long-term durability will provide the most direct link to long-term performance. However, more distant examples may be useful to address the durability of the materials under a more diverse range of exposure conditions. Quaternary glacial striations on quartzites and dating of very old rock surfaces may also be useful. Research is ongoing using cosmogenic dating to estimate the ages of natural materials that have not weathered significantly over very long time periods. For long-term stability, engineered systems should mimic durable natural systems as much as practicable. A licensee's comparison of engineered systems to natural systems should address material properties as well as how the materials are emplaced and distributed to achieve stability.

  For a variety of reasons, certain Quaternary glacial features have the potential to be a very good source of natural analogs. The features of greatest potential value include glacial striations, polished rock surfaces, and glacial erratics. Glacial striations (i.e., fine scratches on a bedrock surface that can be less than a millimeter in depth) and polished rock surfaces are delicate features that could easily be removed by weathering. Preservation of such vulnerable features over long time periods demonstrates the rock's significant resistance to weathering. Glacial features (found in various climates today) are also common in a range of different rock types, which makes them available for use as analogs. Glacial features have been exposed to a range of climates during the past thousands of years. Licensees do not need to determine the precise age of the features in order to use them as a natural analog for an LLW facility. In the absence of dating studies, a general assumption can usually be made that the features are the result of the last glacial period, which ended about 10,000 years ago. Fortuitously, the general 10,000-year age assumption coincides with the 10,000-year compliance period for an LLW facility with significant quantities of long-lived radionuclides.

- Historic analogs: Preliminary analyses and observations by the NRC staff and consultants have indicated that many man-made sites exist that demonstrate the long-term stability of both man-made structures and naturally occurring features. Many of these analog sites are in the United States and consist of structures such as Native American earthen burial mounds, Native American ruins, and rock features. Many of these sites have been dated and shown to have remained intact for thousands of years. Licensees can use studies of the long-term survivability of such features to demonstrate the potential for man-made sites, such as LLW waste disposal facilities, to remain intact for very long periods of time without the need for ongoing active maintenance. For example, the Sarsen stones of Stonehenge may provide data for orthoquartzites.

Table 5-4 Comparison of Chemical Durability (Soxhlet Test) of Waste Glass and Common Minerals

Minerals	Wt% Leached[1]
Quartz crystals	0.41
Milky quartz	0.50
Dolomite	0.55
HLW glass	0.70
Garnet	0.73
Corundum	0.77
Orthoclase	0.90
Granite	1.10
Quartzite	1.20
Felsite	2.10
HLW glass (devitrified)	2.50
Marble (dolomite)	2.90
Calcite	5.80
Basalt	6.10

Source: Brookins, 1984

[1] Represents mass loss in the Soxhlet test

The benefits of historic or archeological analogs are that the ages of rock carvings, monuments, or buildings are usually well known and can demonstrate preservation under known climates and time periods. Furthermore, many potential analogs might be available for a range of rock types. While these analogs can demonstrate preservation for hundreds or thousands of years, the timeframe is less than the 10,000-year compliance period for significant quantities of long-lived radionuclides. However, if available, historical or archeological analogs can complement natural analogs by providing evidence that increases the confidence in the ability of a rock type to resist significant weathering over long time periods.

Although direct evidence of material durability from the site or site region is preferable, licensees can use indirect evidence from other locations where the rock type is similar to the selected rock source, considering differences in environmental conditions. In some cases, the durability may be sensitive to the exposure conditions, while in other cases, it may be less sensitive or insensitive. For example, licensees could use evidence of durability from a diabase igneous rock-type found in Europe to provide insights on a diabase rock-type source in Maryland because the general mineralogy of diabase is similar, regardless of the location. This approach allows the use of datable natural or archaeological and historical rock sites, which could provide general evidence of rock weathering rates or time periods during which rock types have remained resistant to weathering. For example, a licensee could use many datable archaeological sites, such as Stonehenge (constructed about 4,000 years ago of diabase and silica-cemented sandstone), Hadrian's Wall (constructed by the Romans over 2,000 years ago of primarily diabase), and numerous buildings, monuments, and megoliths in Europe, to demonstrate that these rock types have been resistant to weathering over time periods that exceed thousands of years.

Historical evidence can provide useful insights on the durability of certain rock types. One example is the comparison of dated Civil War photographs of diabase outcrops in Devil's Den at the Gettysburg National Military Park to present-day conditions of the same outcrop. A licensee could demonstrate that this diabase has been resistant to weathering for about 150 years.

Similarly, dated grave markers or historical buildings made from the selected rock source or a similar rock type can also provide evidence of resistance to weathering for 100–200 years.

A test wall of building stones was constructed in 1948 in Washington, DC and in 1977 it was moved to the National Institute of Standards and Technology in Gaithersburg, Maryland (Stutzman and Clifton, 1997). This test wall provides the opportunity to study the effects of weathering on different types of stones under identical exposure conditions, and the durability of different materials. Imaging and petrological studies have been performed to characterize texture and mineralogy. Correlation of mineralogical and microstructural features to stone performance provides information for estimating the long-term durability of natural materials. The wall contains 2,352 individual samples of stone: 2,032 domestic stones from 47 States and 320 stones from 16 foreign countries. Over 30 distinct types of stones are represented, including marble, limestone, sandstone, and granite. Data from this project are limited at this time and are likely to be uncertain but may provide licensees another line of evidence for rock durability. Appendix A to NUREG/CR-2642 provides additional information on rock weathering, durability, and examples of analogs that offer insights on general weathering rates of various rock types.

5.3.2.2 Evapotranspiration Covers

Evapotranspiration (ET) covers use the natural processes of evaporation and transpiration to remove water. Although evaporation and transpiration can remove water from the conventional covers discussed in Section 5.3.2.1, it is primarily the layers of low permeable material present in the conventional covers that are intended to limit water from reaching the waste. In contrast, ET covers rely more heavily on evaporation and transpiration to limit water from reaching the waste. The performance of ET covers depends on many factors, especially the climate, soil hydrology, fauna, and plant ecology at a site. ET covers may be used in a variety of settings but may be most effective in arid or semi-arid climates with high potential ET.

Licensees should develop a design for an ET cover that is effective over the range of expected natural and ecological conditions. Natural and ecological conditions are inherently variable over the timeframe of most LLW disposal analyses. With effective design and development, ET covers may be very effective, especially in arid and semi-arid climates. In humid climates, ET covers may be effective at managing a substantial fraction of the infiltration but may not achieve design goals. Infiltration may exceed ET in humid climates or in colder climates, where a large fraction of infiltration may occur as snowmelt when ET is low. Therefore, one of the major lessons learned, albeit not related to physical degradation, is that design of an ET cover must consider natural and ecological variability over the timeframes of the analyses (Benson et al, 2011).

Licensees should be aware that physical, biological, and chemical processes can induce changes in the structure, physical, and biological characteristics of covers that are intrinsic to their functioning as barrier systems. Degradation processes occur over a broad range of time scales and include, but are not limited to, climatic variability, plant succession, geomorphic processes, pedogenesis, anthropogenic impacts, erosion, microbial processes that affect barrier materials and drains (e.g., biofouling), and geochemical processes. An example of unanticipated ecological consequences is the development of deeper-rooted plant species that result in pathways for moisture that are below the design zone for moisture storage and removal.

Engineered systems evolve towards a natural equilibrium. Licensees should recognize that surface soil properties may change quickly and, therefore, should minimize the consequences of these changes by designing and constructing covers that mimic longer-term conditions that are congruent with nature. Cover degradation attributable to pedogenesis and ecological change should be recognized as an inevitable, predictable, natural succession. In some cases, natural pedogenesis and ecological succession can lead to improved system performance over time. Therefore, performance will be steadier over time when licensees design engineered soil layers and vegetation to resemble more closely the characteristics of natural systems. Licensees should develop techniques to understand the magnitude and direction of anticipated natural changes. One approach is to evaluate natural analogs. A licensee may reduce the negative impacts of degradation processes by doing the following:

- providing attention to construction QA; QA is especially important to the successful short-term performance of the cover
- identifying the phenomena that have the greatest impact on total system performance
- analyzing the components of the cover within the system context

To increase confidence in the long-term stability of the site, the licensee should do the following:

- Use natural analogs to better understand and evaluate long-term degradation processes, including both spatial heterogeneity and temporal trajectories of change.
- Design covers that mimic the favorable attributes of natural analogs.
- Evaluate the effects of soil development and ecological change.
- Evaluate the effects of waste subsidence on long-term cover performance.
- Predict and incorporate landform changes in cover and disposal cell designs (NRC, 2011b).

The performance of an ET cover can be particularly sensitive to temporal and spatial variability in precipitation and other processes.

5.3.3 Monitoring of Engineered Barriers

Most waste containment facilities require monitoring to verify performance and support predictive modeling (NAS/NRC, 2007). Environmental monitoring of an LLW disposal facility is required for the duration of the institutional control period. The design of monitoring systems for engineered barriers has proven difficult because of technological challenges and complex goals; however, progress is being made. Observing the effects of degradation processes is critical to understanding long-term performance. Performance monitoring of engineered covers gives a licensee confidence that the cover is functioning as predicted in the performance assessment, intruder assessment, or site stability assessment.

Monitoring of engineered covers is conducted by direct non-destructive performance monitoring and direct or indirect interpretive monitoring. Performance monitoring consists of directly and continuously monitoring the primary performance variable (e.g., the flux of water through a cover) using an in-situ device. Interpretive monitoring consists of measuring secondary

variables (e.g., water content) related to the primary performance variable that can be used to understand or interpret data obtained from primary performance monitoring (NRC, 2011a). Interpretive monitoring can be conducted directly using embedded sensors or indirectly using remote sensing methods such as ground penetrating radar or airborne radar systems. Water content and temperature are the two commonly measured secondary variables employed for interpretive monitoring. Interpretive monitoring currently is conducted almost exclusively using direct methods. However, indirect remote sensing methods likely will become more important in the future, especially for long-term monitoring from remote locations (NRC, 2011a).

During the period of institutional controls, licensees must perform environmental monitoring to ensure continued satisfactory disposal system performance and to develop confidence that the observed performance is likely to persist after the institutional control period. Licensees should also perform physical surveillance to restrict access to the site and conduct minor custodial activities during the institutional control period. Short-term performance of the disposal site can be monitored with various types of onsite instrumentation or by remote sensing. Licensees can use monitoring to detect early releases of contaminants and verify the validity of assumptions made and the accuracy of the results of predictive modeling, thereby reducing uncertainty. The NRC staff recommends that licensees conduct interpretive monitoring because it can provide observations of performance problems that are a precursor to releases of radioactivity into the environment.

Licensees should include in their monitoring plan assumptions, parameters, and features that have a large influence on the disposal facility performance and have relatively large uncertainties. For example, monitoring plant processes or more generally ecological processes, can add greatly to understanding cover stability and performance. Even carefully designed cover systems begin a process of change immediately following construction. These changes can affect containment system performance both directly and indirectly and should be monitored. Additional information gained can reduce uncertainties and support previous predictive modeling. Monitoring is important in obtaining confidence that barrier components are performing as intended and is a vital tool in detecting early signs of degradation of a disposal system. Because of increased understanding of potential shortcomings with engineered surface barriers, monitoring of engineered systems is being recognized as a powerful tool that has the potential to yield valuable data. A well-conceived monitoring system for engineered surface barriers could provide information to assess barrier performance including degradation.

Airborne and satellite-based remote monitoring techniques can efficiently monitor engineered covers. For example, remote sensing may detect vegetative change that depends on characteristics of water flow. Linear features of heavier vegetation may be indicative of cracks or other structural features allowing increased contact of water with the waste and may be indirect signs that the overall stability of a barrier may be decreasing. Sensor development has rapidly advanced so that sensors are becoming not only quicker, more reliable, and longer lasting, but also smaller, more automated, wireless, and more sophisticated. Licensees can observe changes in vegetation, soil water content and temperature through multispectral imaging. Ground penetrating radar, LiDAR (or Light Detection and Ranging technology), and other remote-sensing techniques may detect stabilization problems at the very early stages through their high-resolution output (NAS/NRC, 2007). Licensees may someday be able to place automated sensors throughout the cover to monitor those features and processes demonstrated to be significant.

Licensees should not use monitoring as a substitute for the development of adequate performance data before implementing their system, but rather to support the previous determination of adequacy considering uncertainty. When there is uncertainty associated with the waste disposal system, monitoring data can be used to maintain confidence in the performance demonstration. If environmental monitoring of an LLW disposal facility is limited for the duration of the institutional control period, maintaining this confidence will be possible for performance over decades and centuries (assuming no disruptive events); however, confidence will be harder to maintain for time periods longer than this.

Monitoring and modeling activities are complementary. Modeling can serve to focus monitoring efforts by identifying key processes and parameters or disconnects between field observations and model results. Similarly, the results of monitoring provide feedback to refine models and improve the understanding of the system. Licensees should design their monitoring systems to understand processes and events and identify early indicators of performance problems.

6.0 PERFORMANCE PERIOD ANALYSES

The regulation at 10 CFR 61.13(e) requires performance period analyses to do the following:

(1) Assess how the disposal site limits the potential long-term radiological impacts consistent with available data and current scientific understanding.

(2) Identify and describe disposal site design features and natural site characteristics relied on to demonstrate compliance with the applicable performance objectives set forth in 10 CFR 61.41(b) and 10 CFR 61.42(b).

(3) Evaluate those processes likely to occur during the performance period, including degradation, deterioration, and alteration processes that affect performance.

The primary purpose of the performance period analyses is to provide information that demonstrates that releases of long-lived[1] radionuclides from an LLW disposal facility are effectively managed. The performance period is defined in the regulations to be the time following the compliance period. Performance period analyses are required only if the disposal facility is accepting significant quantities of long-lived radionuclides and the licensee has used a 10,000-year or longer compliance period. Table 6-1 lists long-lived isotopes that may be present in LLW inventories.

The disposal system consists of the disposal units, disposal site, land disposal facility, and surrounding environment. The analyses should evaluate natural and engineered characteristics of the disposal system and describe how those characteristics will reduce long-term impacts. Over the long timeframes of regulatory concern, performance of the disposal system is likely to be driven by the features of the natural system rather than by man-made engineered barriers. The licensee should calculate the expected concentrations of long-lived waste remaining in the disposal site after the compliance period to risk-inform the longer-term performance period analyses. The resources and effort devoted to the assessment for the performance period will increase in proportion to the magnitude of the longer-lived radioactive waste inventory present, considering both the initial inventory and ingrowth. Table 6-2 provides examples of how performance period analyses may be risk-informed, accounting for the longevity of the inventory and the duration of the hazard.

Licensees should provide model support for the performance period analyses; however, that support will likely be less quantitative and involve more expert judgment than model support that would be required for the compliance period.

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[1] Long-lived radionuclides means radionuclides where (1) more than 10 percent of the initial activity of the radionuclide remains after 1,000 years, (2) the peak activity from progeny occurs after 1,000 years, or (3) more than 10 percent of the peak activity of a radionuclide (including progeny) that occurs within 1,000 years remains after 1,000 years. The first part of the definition represents radionuclides with approximately a 300-year or longer half-life. Table 6-1 lists examples of isotopes that are short lived but produce long-lived progeny (e.g., Am-241, curium (Cm)-242).

Table 6-1 Long-lived Isotopes Potentially Present in LLW Inventories

Isotope	*Half-life (yr)*	*Long-lived*		*LLW PA Inventory[1]*	*Isotope*	*Half-life (yr)*	*Long-lived*		*LLW PA Inventory[1]*
		Parent	*Progeny[2]*				*Parent*	*Progeny[2]*	
Al-26	7.17×10^5	X			U-233	1.59×10^5	X	Th-229	Yes
C-14	5,730	X		Yes	U-234	2.45×10^5	X	Th-230	Yes
Cl-36	3.01×10^5	X		Yes	U-235	7.038×10^8	X	Pa-231	Yes
K-40	1.3×10^9	X			U-236	2.342×10^6	X	Th-232	Yes
Ni-59	7.5×10^4	X		Yes	U-238	4.468×10^9		U-234	Yes
Se-79	1.1×10^6	X		Yes	Np-237	2.14×10^6	X	U-233	Yes
Zr-93	1.53×10^6	X			Pu-238	87.7		U-234	Yes
Nb-94	2.0×10^4	X			Pu-239	2.41×10^4	X	U-235	Yes
Tc-99	2.14×10^5	X		Yes	Pu-240	6.54×10^3	X	U-236	Yes
Pd-107	6.56×10^6	X			Pu-241	14.4		Np-237	Yes
Sn-126	1×10^5	X		Yes	Pu-242	3.76×10^5	X	U-238	Yes
I-129	1.6×10^7	X		Yes	Pu-244	8.26×10^7	X	Pu-240	
Cs-135	3×10^6	X			Am-241	432		Np-237	Yes
Sm-146	1×10^8	X			Am-242m	16 hr		U-234	Yes
Pm-147	2.62		Sm-147		Am-243	7.38×10^3	X	Pu-239	Yes
Sm-147	1.06×10^{11}	X			Cm-242	0.446		U-234	
Eu-152	13.3		Gd-152		Cm-243	28.5		Am-243	
Gd-152	1.08×10^{14}	X			Cm-244	18.1		Pu-240	
Ra-226	1,600	X		Yes	Cm-245	8.5×10^3	X	Np-237	
Th-229	7.3×10^3	X		Yes	Cm-247	1.56×10^7	X	Am-243	
Th-230	7.7×10^4	X	Ra-226	Yes	Cm-248	3.39×10^5	X	Pu-244	
Th-232	1.41×10^{10}	X		Yes	Cf-249	351		Cm-245	
Pa-231	3.28×10^4	X			Cf-251	898		Am-243	
U-233	1.59×10^5	X	Th-229	Yes	Cf-252	2.64		Cm-248	

1 Any isotope that is to be disposed of should be considered as part of the LLW performance assessment (PA) inventory, but not all isotopes may be included in the technical analyses. The isotopes with "Yes" are expected to more commonly be a significant isotope in an LLW PA inventory based on past analyses as of the date of this publication. Technical analyses should consider progeny important for the radiological dose calculations. For example, Rn-222 is an important short-lived progeny of radium (Ra)-226.

2 Only the first long-lived progeny encountered in decay chains are listed in this column.

Table 6-2 Expected Level of Review Effort for Performance Period Analyses

Radiation Hazard and Duration[1]	Level of Review Effort	Example
Short-lived, any concentrations of radionuclides	NA	Long-term performance period analyses are not required.
Short-lived and low concentrations of long-lived or limited quantities of concentrated long-lived radionuclides	Low	Using undisturbed concentrations during the performance period, provide analyses showing that the dose to inadvertent intruders meets the compliance period performance objective (e.g., assume no dilution and perform an intruder assessment).
Moderate concentrations of long-lived or moderate quantities of concentrated long-lived radionuclides	Moderate	Provide analyses showing that the disposal system will limit releases from natural processes and plausible disruptive events[2]. Estimate the range of doses that may result to intruders and members of the public and demonstrate that they are reduced to the extent reasonably achievable[3]. Include uncertainty and variability. Formal peer review of the analyses and results should be considered.
High concentrations and quantities of long-lived radionuclides	High	Provide analyses showing that the disposal system will limit releases from natural processes and plausible disruptive events. Estimate the range of doses that may result to members of the public and demonstrate that they are reduced to the extent reasonably achievable. Include uncertainty and variability. Support for the range of impacts should include model support, such as that derived from natural analogs of long-term site evolution. Independent, formal peer review of the analyses, results, and model support should be performed.

[1] The terms "low," "moderate," and "high" in Table 6-2 refer to *relative* concentrations when waste was already determined to have significant quantities of long-lived radionuclides necessitating the performance period analyses.

[2] Discussed in Section 6.2.

[3] Discussed in Section 6.3.1.2.

Review methods other than those suggested by the examples in the table may be suitable. The review effort should be greater when risks for the performance period are larger. If possible, simple, conservative analyses should be used, especially when projected risks are low. For higher hazards and longer-lived wastes, the analyses should address central scenarios as well as less likely, but plausible, disruptive site scenarios (discussed in Section 6.2). Licensees should provide model support for analyses, as discussed in Section 2.2.4, especially for higher hazard and longer-lived wastes. To determine what hazard is posed, it may be useful for licensees to estimate the doses to intruders and public receptors with conservative site scenarios and compare those estimates to their central scenarios. A hazard may or may not translate into risk, but high potential long-term hazards should have more support and independent review relative to low potential long-term hazards.

6.1 Disposal Site Characteristics that Enhance Long-Term Isolation

The regulatory requirements do not prescribe the type of performance period analyses that a licensee must perform. Licensees may use analyses (e.g., screening, quantitative probabilistic) considered sufficient to demonstrate that the regulatory requirements will be met. However, there are limits to the analyses of projected performance over very long timeframes (e.g., tens of thousands of years) and how much confidence a licensee should place in the results of the analyses. Disposal sites and near-surface disposal facility designs that have more disposal site characteristics that enhance long-term isolation may be more likely to achieve a higher degree of long-term isolation of waste from the accessible environment.

Table 6-3 shows the characteristics that will enhance isolation at most disposal sites. Individual characteristics may not apply at a specific site. For example, low porosity in a cementitious wasteform generally reduces the potential for subsidence and reduces leaching. However, use of a low-porosity cementitious wasteform in a cold climate may result in freeze-thaw damage that could contribute to release over the long term. While use of robust, low-porosity wasteforms is generally favorable, in this specific example it may not be. Some of the characteristics listed in Table 6-3 apply to the design of disposal facilities, while others apply to the site characteristics.

6.2 Scope of the Performance Period Analyses

The performance period analyses should provide information about the performance of the disposal system under a range of conditions that represent central scenarios, as well as less likely, but plausible, site scenarios that may have significant consequences. Licensees should consider the range of conditions consistent with the site stability analyses (described in Chapter 5.0), including the FEP analysis (Section 2.5). Less likely, but plausible site scenarios include those that are unlikely to be observed (e.g., as low as a 10 percent chance of occurrence over the analysis timeframe), as well as those that are expected to be observed over the analysis timeframe. Performance period analyses for various sites may have different timeframes owing to differences in radionuclide inventories, as well as differences in geologic settings. Therefore, a single event frequency (i.e., 10^{-5}/yr) cannot be defined for the purposes of this regulatory guidance.

Table 6-3 Disposal Site Characteristics that Could Enhance Long-Term Isolation

Characteristic[1]	Description
Simple, passive designs	Simple designs are less likely to experience unforeseen failure mechanisms; passive designs do not rely on active monitoring and maintenance
Designs that mimic natural features	Stable natural features may indicate design characteristics that may help achieve long-term isolation
Low-relief designs	Designs with low relief (e.g., buried) will experience lower rates of erosion
Low water contact with waste	Release and instability are generally associated with mass transfer. Limited water contact reduces rates of aqueous phase mass transfer
Robust, low-porosity wasteforms	Durable, low porosity wasteforms enhance long-term stability by limiting consolidation and subsidence
Geochemical compatibility of the waste and disposal environment	Waste that is geochemically compatible with the disposal environment is less likely to experience significant release into the environment
Stable disposal environment conditions	Physically and chemically unstable environmental conditions contribute to long-term instability. For example, leaching from waste can be highest in zones of water table fluctuation
Accreting environments	Disposal systems that are gaining mass over time contribute to waste isolation by working with the natural processes instead of against them
Large distance to water table and homogeneous natural materials	The unsaturated zone can provide a significant barrier to releases to an aquifer, especially if the natural materials are relatively uniform which contributes to confidence in the performance of sorptive materials
Deep disposal	Many disruptive processes are more dynamic, complex, and more likely for shallow disposal compared to deeper disposal
Limited natural resources	A disposal system with limited natural resources decreases the likelihood of anthropogenic processes or events impacting the disposal facility or site
Stable climate	Disposal systems located in a more stable climate are less likely to experience impacts from climate variation
Low frequency of geologic and tectonic events	Over the long term, disposal systems that are in areas of low geologic and tectonic activity are more likely to achieve waste isolation

[1] Highlighted characteristics are associated with 10 CFR 61.50 site suitability characteristics.

FEPs defining the natural and engineered systems as well as assumptions about future human behavior, will be needed for the performance period analyses. The challenge for performance period analyses is to provide credible assessments of the future evolution of the disposal system while avoiding open-ended speculation. Near-surface disposal introduces specific challenges over the long term because environmental processes can have complex, dynamic, and nonlinear responses. The NRC staff believes that the approach recommended in the following sections is suitable for defining the scope of the performance period analyses for land disposal. The goal of the long-term analyses is to understand the safety implications of the type of waste being disposed and not to precisely estimate the future evolution of the surface of the earth. The goal is to provide a perspective on how the hazard may evolve over time (e.g., persistence of long-lived radionuclides and potential for ingrowth of risk-significant daughters).

Disposal of high-specific-activity waste, if improperly managed by a licensee, poses the greatest radiological risk to public health and safety. This is because release of relatively small amounts of mass may result in significant impacts. It would be very difficult for a licensee to demonstrate with reasonable assurance that public health and safety are protected from the disposal of the high-specific-activity waste at a site where the waste is improperly managed because the margin for error is small. For example, accidental release of a relatively small quantity of strontium (Sr)-90 into an aquifer at the West Valley Demonstration Project resulted in a significant groundwater plume requiring remediation (NYSDEC, 2008). Only a small amount of high-specific-activity waste released into the environment can cause significant environmental impacts.

For the performance period, the margin for error is not as small because the fraction of the waste with high specific activity has decayed in the disposal units. The specific activity of the material remaining in the disposal site is much lower compared to the waste when first disposed. Furthermore, because of the long timeframes involved, a licensee may consider the performance objectives in 10 CFR Part 61, Subpart C, when evaluating whether its site meets the site suitability requirements in 10 CFR 61.50 for the performance period. For example, a disposal site might have some projected seismic activity (10 CFR 61.50(a)(5)(ii)) sometime after the 10,000-year compliance period. However, future seismic activity would disqualify the site only if the licensee was unable to demonstrate that the performance objectives would be met. Therefore, it is acceptable for the performance period for a licensee to evaluate the significance of the site characteristics using technical analyses.

6.2.1 Features, Events, and Processes

The objective of the performance period analyses is to provide information to decision makers about disposal system performance under various site scenarios. Licensees should assess the uncertainties and present the results. Reviewers should not hold the performance period analyses to a standard that is not attainable. In comparison with the compliance period, the performance period analyses will be more susceptible to bias because objective supporting information will be more limited.

Different LLW land disposal facilities may have significantly different characteristics and may contain different wastes. The FEPs for one disposal site may be substantially different from those at a different site. Identification of the FEPs relevant to the performance period will be site-specific. Section 2.5 describes the FEP process that a licensee may use to develop the scope of the technical analyses (e.g., performance assessment).

A licensee may extend the compliance period calculations without modification provided that the calculations are complete with respect to including key FEPs relevant to the performance period. It will be necessary for the licensee to communicate the additional uncertainties associated with events and processes that may occur in the long-term performance period if they are not represented in the compliance analyses. The analyses that are developed for the compliance period may not be sufficient for the performance period analyses if (1) disruptive processes are expected to occur during the performance period that have not been included in the compliance period analyses or (2) the cumulative impact from repetitive events over the longer timeframes is not included and the repetition of those processes and events could lead to significant impacts. It is appropriate for a licensee to consider potentially beneficial natural processes, such as dispersion and dilution, in addition to detrimental processes. In general, the greater the geological and geomorphological stability of a potential disposal site, the greater the likelihood that FEPs that may occur in the performance period will already be represented in the licensee's compliance period analyses. However, the representation of a particular FEP in the compliance period analyses may be different in the performance period analyses. Even if the same set of FEPs may be appropriate for both analyses, they may be represented differently in each analysis.

The disposal site suitability requirements for the land disposal of LLW are provided in 10 CFR 61.50. The process to determine whether these criteria will be met during the compliance period is complementary to the FEP process (discussed in Section 6.2.2). If the disposal site suitability requirements are met for the 10,000-year compliance period, many of these same criteria will likely also be met for the performance period. However, very long-term processes (e.g., long-term erosion) and those events associated with long time periods (e.g., movement of tectonic plates) are FEPs that may be relevant for the performance period. For example, 10 CFR 61.50(a)(5)(ii) is a requirement associated with plate tectonics involving very long-term processes and events that may need to be represented in performance period analyses but are not required to be represented in the compliance analyses.

Many potential disruptive site scenarios involving certain forms of flooding, landslides, earthquakes, and volcanoes will not be evaluated in the performance period. Potential sites containing FEPs that occur with such frequency and extent to significantly affect the ability of the disposal site to meet the performance objectives of 10 CFR Part 61, Subpart C, for the compliance period would not be considered suitable for LLW disposal. These site scenarios may also preclude defensible modeling and prediction of long-term impacts.

The number of possible site scenarios that can be developed will be reduced after screening based on the site suitability FEPs. However, FEPs that have been screened from further consideration in the compliance period may not be able to be screened from further consideration for the performance period analyses. For example, the rate of erosion may be estimated to be sufficiently low over the compliance period and, as a result, the FEP of erosion is not within the scope of the compliance period analyses. However, over the analysis timeframe for the performance period, the rate of erosion may be significant enough that the FEP should be included within the scope of the performance period analyses. Therefore, the compliance period FEP processes will reduce the FEPs applicable to the performance period, but some of the FEPs eliminated in the compliance period may apply to the performance period.

6.2.2 Screening of Features, Events, and Processes Based on the Requirements in 10 CFR 61.50

A licensee should complete the FEP screening and scenario development process for the performance period in a risk-informed manner. Some of the site characteristics for the compliance period are either required to be present or to be absent (i.e., hydrological site characteristics for 500 years) because they are precursors of poor long-term performance. The site suitability characteristics for the performance period may be evaluated considering radiological risk (i.e., they are not exclusionary simply by their presence or absence).

Appendix B presents hazard maps that the NRC staff created related to the features and phenomena of the criteria in 10 CFR 61.50. The hazard maps illustrate the FEPs associated with the site suitability requirements. The maps cannot be displayed in this document at sufficient size to be used to determine if one of these phenomena would impact any specific location. The figures provide only an illustration of potentially impacted areas.

Regulators should not use the hazard maps in Appendix B to prohibit disposal because the resolution of the maps and the precision and accuracy of the techniques used to generate them may not be sufficient for site-specific evaluations. However, regulators should use the maps to determine when greater review effort and more technical basis should be expected for a licensee's site-specific evaluation. In addition, the data used to produce these maps could be used, with geographic information system software, to perform screening-level FEP analyses. However, the data used to produce these figures in Appendix B were not collected at a scale of resolution sufficient to allow detailed site-specific evaluations by either regulators or applicants.

For the performance period analyses, FEP screening based on the requirements in 10 CFR 61.50 does not need to be resource intensive. Background information and knowledge of the geologic history of the site can be used to evaluate the likelihood of an FEP being present at a potential disposal site during the performance period. This evaluation is a qualitative exercise using information on the history of a disposal system. The phenomena and features that have occurred at the disposal site can be used to make judgments about the future and to help make screening determinations. For example, a disposal system may be near a floodplain but outside of the influence of its associated processes. A licensee may examine the location, nearby topography, and the history and events and provide information on the potential for future floodplain formation at the disposal site. With diverse information from various sources, a licensee should establish a sufficient technical basis to support the exclusion of a floodplain or a near-surface water table forming at or near the proposed disposal site. However, for the performance period, information from the disposal system may point to renewed flooding of the area or it may simply be insufficient to form a basis to screen the process out. In such a case, the licensee would include floodplain formation in the assessment for the performance period.

6.2.3 Future Human Behavior

FEPs that describe future human behavior in the performance period should be consistent with present knowledge of the conditions in the region surrounding the disposal system. It is not necessary for a licensee to project changes in society, changes in human biology, or increases or decreases in human knowledge or technology. The selection of receptors and exposure pathways in the performance period analyses should be limited to the consideration of the natural variability in conditions, processes, and events.

Over long timeframes, the future environmental conditions may be significantly different from those in the present. If shorter term analyses limited the behavior, characteristics, and pathways of receptors based on present conditions, the licensee should assess the impact from *significant* changes in climatic or environmental conditions (see Section 2.5.4.2). A licensee should modify the behavior of potential receptors over the performance period using current climatic analog locations. For example, groundwater may not be potable in the present, but the long-term analyses could examine whether that site characteristic is expected to be present (or absent) under future climate states. If licensees were to provide a side-by-side comparison of the assumed characteristics of receptors (e.g., pathways, consumption rates, and exposure times) with those of generic "screening" receptors used in the dose analyses, the reviewer could determine the importance of the assumed receptor characteristics. This sort of comparison is strongly recommended. Generic receptor characteristics are found in a variety of documents (NRC, 1992; NRC, 1981a; NRC, 1982b). This type of comparison is also good practice for compliance period assessments. Estimates of future disposal site performance that are primarily based on the engineered design, waste characteristics, and site characteristics are likely to be less speculative than estimates that rely on assumptions about future human behavior.

6.3 Analyses for Long-Lived Waste

A licensee should develop performance period analyses to demonstrate how the disposal facility has been sited and designed to reduce long-term impacts and to indicate its potential long-term performance. The term "analyses" is used here to describe different types of evaluations that may be performed, some of which may be quantitative and others that may be qualitative. A description of barrier capabilities and design considerations, in addition to a conservative screening evaluation, may be sufficient for lower-risk systems (e.g., limited quantities and concentrations of long-lived waste). A detailed quantitative assessment of long-term performance may be necessary for higher-risk systems (e.g., those containing significant quantities of concentrated, long-lived waste). Figure 6-1 is a diagram of the recommended approach to developing the analyses for the performance period.

The analyses for the performance period are not intended to predict future states of the LLW disposal system. The analyses associated with the performance period should represent a credible technical effort using available data and current scientific understanding to assess the long-term performance of the waste disposal facility, including consideration of uncertainties. Limited data may be available to licensees to support development of long-term analyses of the performance period. Licensees should use expected values (e.g., central tendency) and bounding values in their assessments of long-term impacts. Use of expected values, such as the median, can convey the most likely outcome. When expected value calculations are complemented with bounding value calculations, the potential impact of uncertainty on the expected outcome can be conveyed. For the expected value calculations to be useful, the expected values must be representative of the site-specific conditions and features. Expected value calculations must have adequate supporting information to be of utility to regulators and other stakeholders. Bounding values may be necessary when data are not available or are very sparse, or formal expert judgment may be necessary.

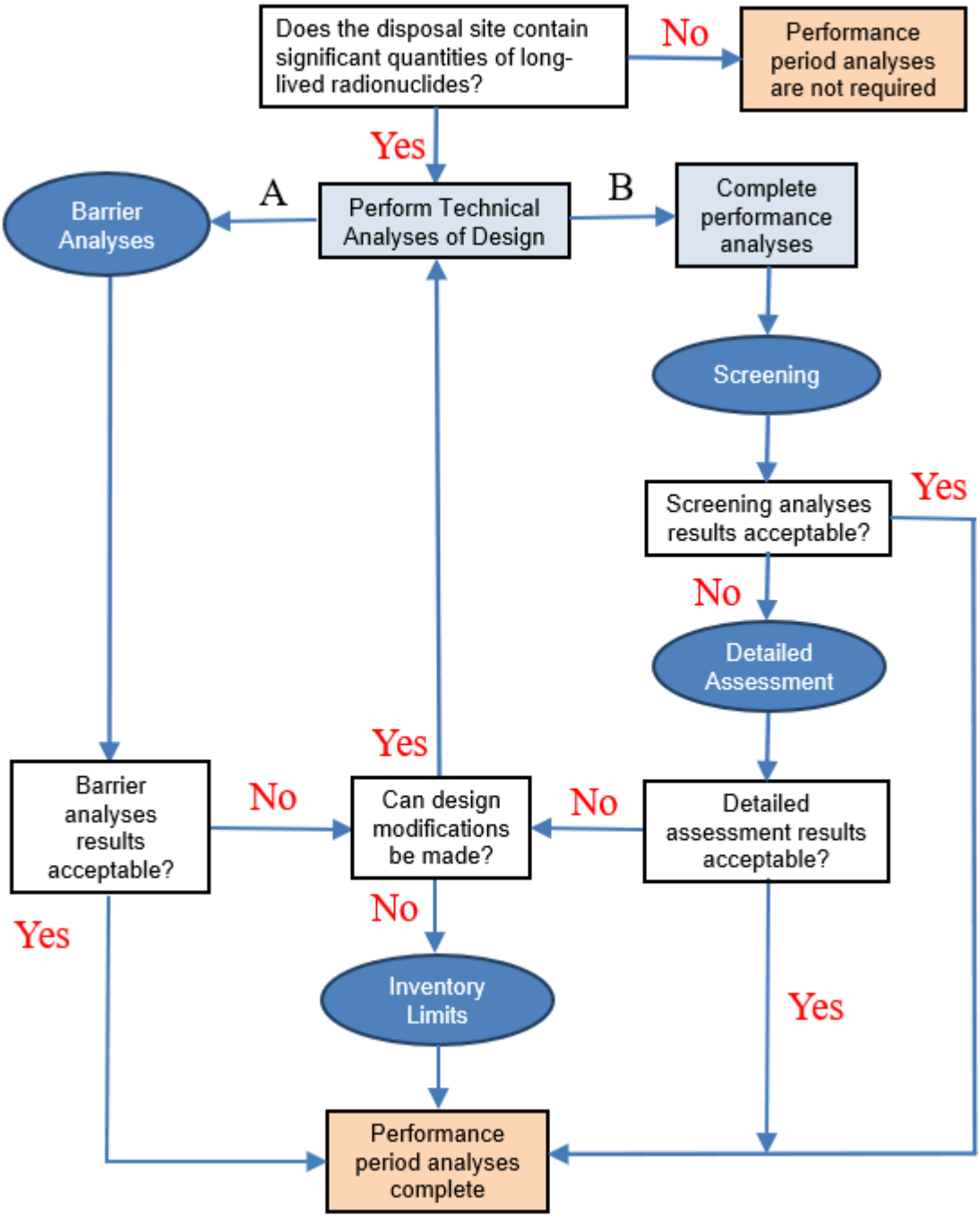


Figure 6-1 Recommended Approach to Developing Performance Period Analyses

Although the use of bounding values may be necessary, the NRC staff recommends that a licensee be cautious in using bounding values for performance assessment or other analyses used to assess long-term performance. A bounding assessment and assigning a bounding value for every parameter are two different things. Many environmental systems can have complex and nonlinear responses, making selection of a bounding value for a specific

parameter challenging, if not intractable. An appropriate bounding value may not be intuitive to the licensee, and it may only be bounding locally and not globally. In addition, the compound effect of selecting numerous bounding values in the analyses can result in nonphysical results. For example, it would be unreasonable for a licensee to evaluate a high-energy disruptive process that destroys the engineered barriers of the waste disposal facility without also considering the resultant dispersion and dilution of the waste. Even in hypothetical bounding calculations, it is useful to provide some context for the reasonableness of the calculations.

The analyses for the performance period may be quantitative, semi-quantitative, or qualitative, depending on the waste characteristics or other factors. As discussed previously, the performance period analyses should be risk-informed. Different approaches are acceptable for providing information on long-term performance. The NRC staff recommends that licensees perform two sets of analyses for the performance period:

- performance analyses to demonstrate that releases from disposal of long-lived waste will be reduced to the extent reasonably achievable for the performance period (shown as B on Figure 6-1)

- barrier analyses to understand the performance of engineered and natural barriers (shown as A on Figure 6-1)

Both types of analyses should be performed but a licensee may provide justification that one type is sufficient. The analyses are complementary. The performance analyses provide estimates of how releases will be reduced by the proposed disposal concept, engineered barriers, and disposal site. In other words, performance analyses estimate the expected central tendency of future long-term performance. The barrier analyses identify which barriers are being relied on, practical changes that could be made (if necessary), and the impact of those changes on estimated performance. If the performance analyses find that the performance of the system during the performance period is expected to be comparable to the estimated performance during the compliance period, then barrier analyses may not be necessary. Barrier analyses can still be useful to provide understanding of the important barriers being relied on to achieve acceptable long-term performance.

6.3.1 Performance Analyses

A licensee should first use simple, conservative screening analyses. The screening analyses may identify that the radiological risks are acceptable or that allowable limits (see Chapter 8.0) or other controls may be necessary. If the results of the screening analyses are not acceptable, a licensee may limit disposal of certain types of waste, or they may make design modifications. In addition, quantitative performance period analyses may be performed to determine whether the expected radiological risks[2] to the public from the disposal action are acceptable. The results of refined performance period analyses may demonstrate that the radiological risks are acceptable even though the results of the conservative, screening analyses show that they are not.

[2] Other metrics such as fluxes of radionuclides in the environment or concentrations of radionuclides in the environment may also be used. However, radiological doses are used in the compliance period and provide an apples-to-apples comparison for regulatory analysis.

6.3.1.1 Types of Performance Analyses

The benefit of screening analyses is that they are relatively easy to perform and document; therefore, they are easier for stakeholders to review and understand, often facilitating decision- making. Providing transparency is an important part of developing the confidence of stakeholders. Screening analyses will be significantly less resource intensive for a licensee to complete compared to probabilistic multi-physics simulations. Screening performed with conservative parameters and calculations are not radiological risk calculations and should not be interpreted as such. They should be clearly described as hypothetical and pessimistic with the objective of identifying whether the potential exists for unacceptable radiological risk to a member of the public and to an inadvertent intruder in the performance period. Many beneficial features, processes, and characteristics may be purposely ignored in the calculations. If the estimated doses to the public and intruder from the screening analyses are below the limits in the 10 CFR 61.41(a) and 10 CFR 61.42(a) performance objectives for the compliance period, additional performance period analyses are not necessary.

The NRC staff cannot determine the appropriate conservative screening analyses that should be applied for all potential designs, waste streams, and disposal sites because there are too many combinations. Conservatism may be defined differently for different decisions. However, for all sites, licensees should provide, and reviewers should evaluate, the following:

- the potential radiation exposure pathways to the public and intruder from disposed waste
- the release pathways expected to be most significant
- the technical basis for the conservatism of the screening analyses
- how the parameterization and representation of the screening analyses has accounted for uncertainty and variability
- the barriers and processes that reduce or mitigate releases

For many disposal sites, a potable groundwater pathway will be a primary pathway for releases to the environment. A conservative screening analysis for the groundwater pathway would be one for which all of the waste inventory is available for release, solubility limits are not applied or are set at conservative values, and delays due to sorption during transport are eliminated by setting distribution coefficients to zero or very small values. However, the physical limitations on mass transport processes would still be included in waste release modeling. For example, if a waste container had a pore volume of X and the volumetric flow rate into the container was a small fraction of X per unit time, the exchange process should still be included in the simulation. In addition, the screening assessment should include the dilution during transport arising from the geometry of the waste, hydrogeological system, and infiltration processes.

If screening analyses have been performed and the projected results are not acceptable, then a licensee may modify the facility design, develop limits on the types of waste that are acceptable for disposal, or perform additional analyses. A licensee must demonstrate that 10 CFR 61.41(b) and 10 CFR 61.42(b) requirements will be met.

The analyses that a licensee may perform for the performance period will be similar to the performance assessment and inadvertent intruder assessment completed for the compliance

period. The guidance in Chapters 2.0, 3.0, and 4.0 for the compliance period applies to the performance period analyses. One primary difference is that licensees will need to consider the additional uncertainties introduced by the long timeframes associated with the performance period.

Uncertainties associated with the performance period may be larger than those associated with the compliance period. As the timeframe for the analyses is extended, temporal processes that have a rate that is insufficient to result in a significant change to performance before may be significant when evaluated for the longer performance period. For example, carbonation of a cementitious barrier may be sufficiently slow that the passivation of rebar is not affected over the 1,000- or 10,000-year compliance period. However, the rate of carbonation could be sufficient to reduce the protective passivation of the rebar after the compliance period, leading to deterioration or failure of the cementitious barrier during the performance period.

Figure 6-2 is an example of the scope of the technical analyses for the compliance period and performance period for a hypothetical site. Figure 6-2 has shaded areas to show that some processes and events will apply only to the compliance period, some will apply only to the performance period, and others will need additional analyses to determine their applicability. The scope of the analyses for the compliance period may include both short- and long-term processes and events depending on the type of waste that is disposed. A licensee will assess some processes in the performance assessment for the compliance period, whereas others may clearly be applicable only to the performance period. In a quantitative analysis of the performance period, a licensee should assess those FEPs that were determined to be insignificant for the compliance period based on rates that were too slow or likelihoods that were too low. The licensee should determine whether those FEPs are relevant to the performance period. It may be useful for a licensee to perform an iterative assessment to determine the appropriate scope of the performance period analyses.

6.3.1.2 Effectively Manage Radioactive Releases

There are no dose limits established for the performance period in 10 CFR 61.41(b) and 10 CFR 61.42(b). The performance metric a licensee must apply to the assessment of long-term releases during the performance period is that releases of radioactivity from the disposal site must be effectively managed during the performance period (10 CFR 61.41(b)) and that exposures to an inadvertent intruder must be effectively managed during the performance period (10 CFR 61.42(b)). This metric affords flexibility to a licensee to consider socioeconomic factors when assessing long-term protection of public health and safety.

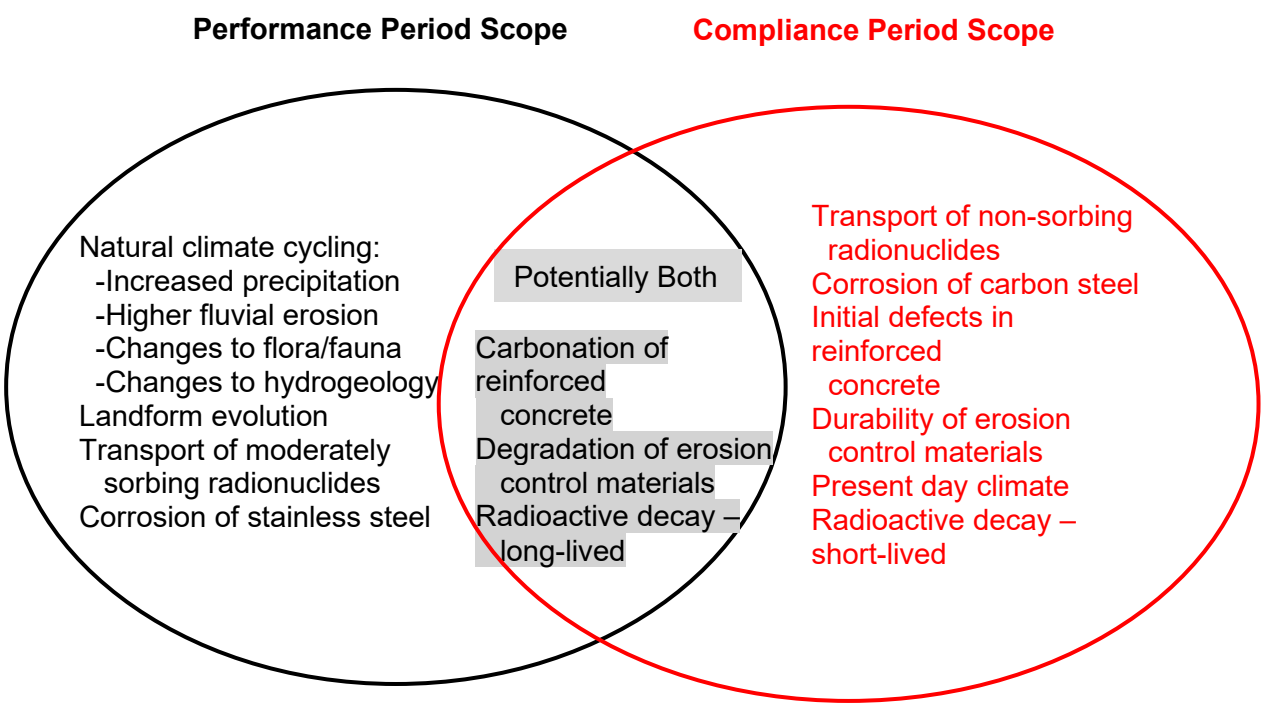


Figure 6-2 Scope of the Technical Analyses for the Compliance Period and Performance Period for a Hypothetical Site

Because of the problems associated with discounting over long timeframes, the NRC staff recommends that licensees provide the proportion of resources that are proposed to be used to protect public health and the environment for different timeframes. These resource estimates should be provided in present-day values. Timeframes to consider may include the following:

- institutional control period (100 years)
- Class C and GTCC waste intruder barrier period (500 years)
- compliance period (1,000 years)
- compliance period for significant quantities of long-lived radionuclides (10,000 years)
- performance period (site-specific values > 10,000 years)

The regulations specify individual dose limits (for the compliance period) rather than collective dose to a population. A licensee should estimate the projected individual doses for the performance period and the resources used to reduce those impacts to certain dose values or the projected cost to achieve different dose values. Dose values that a licensee may consider for comparison purposes may include 10 CFR 61.41(a) public annual dose limit for the compliance period (25 mrem/yr), 10 CFR 61.42(a) intruder annual dose limit for the compliance period (500 mrem/yr), 10 CFR Part 20 public annual dose limit (100 mrem/yr), and background radiation values for the site, or other limits for which a licensee provides technical bases. Radiological doses at very long timeframes are based on many unstated assumptions. However, radiological doses provide a metric for stakeholders to consider. Other metrics may be appropriate for a licensee to consider, such as concentrations in the environment and flux rates. The analyses should demonstrate that a reasonable attempt has been made through site selection, facility design, and waste acceptance to reduce releases to the public to the extent reasonably achievable for the performance period (see Example 6.1).

In summary, a licensee should provide the following in the performance period analyses:

- a summary of present-day resources, with no discounting, used to reduce releases for different regulatory timeframes
- a description of the additional resources needed to achieve a greater reduction in releases during the performance period, or why further reductions are not possible
- a discussion of why additional resource usage is not warranted

If the performance of the disposal system during the performance period is comparable to the expected performance during the compliance period, a licensee does not need to estimate the potential for additional reductions through modifications to the disposal system.

6.3.2 Barrier Analyses

Licensees should use barrier analyses to identify and describe the capabilities of barriers, the challenges and stresses expected to be imposed on barriers, and the contribution of barriers to reducing or delaying releases of long-lived radionuclides into the environment. Licensees can use different types of analyses, ranging from qualitative to quantitative, to demonstrate how the barriers of the disposal facility reduce long-term impacts. Barrier and component analyses can be used to satisfy the requirements in 10 CFR 61.13(e) to illustrate the long-term performance of barriers and components of an LLW disposal system. Barrier analyses involve decomposing the performance of the disposal system into the performance of individual barriers or combinations of barriers under assumed scenarios or configurations.

Licensees should discuss the capabilities of engineered and natural barriers to reduce releases or exposures. The discussion can help stakeholders understand the performance of the disposal system by addressing the events and processes that may impact the barriers' capabilities, the expected persistence and durability of the barriers, and the basis for these expectations. At long timeframes, the performance of an engineered barrier is likely to diminish. A challenge with taking a qualitative approach to describing barrier performance is determining the actual barrier performance rather than the potential barrier performance. The potential barrier performance may not be realized because the performance is deteriorated or eliminated by processes and events, or because the performance is masked by the performance of other barriers (see Chapter 7.0). As-built performance can differ from as-designed expectations. Quantification of the performance of the engineered and natural barriers is useful because the estimated performance of the barrier is represented in the calculation regardless of whether the performance is close to potential or has significantly deteriorated. Quantitative barrier analyses can reduce some of the challenges and provide estimated performance based on available information.

Example 6.1: A disposal site is located in a semi-arid environment in the southern United States in an area of low relief and long-term accretion. The waste streams proposed for disposal have a Sum of Fractions (SOF) of 4.3 when evaluated against the concentrations in Table H-1 in Appendix H. The licensee determined that the site would dispose of significant quantities of long-lived radionuclides, and therefore, used a 10,000-year compliance period. Performance period analyses are required. Because the compliance period analyses demonstrated that the performance objectives would be met by a significant margin, the licensee elects to perform a screening analysis for the performance period by extending the compliance period calculations to the performance period with conservative parameters.

Conclusion: The licensee provides the NRC with a list of the potential radiation exposure pathways to the public. The dominant release pathway in the compliance period analysis was from contaminated groundwater. It is anticipated that this is the dominant exposure pathway for the performance period analyses. The licensee performs an assessment of FEPs that were screened out of the compliance period assessment to determine whether any of those phenomena are potentially significant to the performance period analyses. The only significant process identified is natural cycling of the climate. Because the site is located in the southern United States, the impact of natural cycling of climate is represented in the screening analyses by assuming a wetter and cooler climate (e.g., greater infiltration). Conservatism introduced in the screening analyses for the performance period includes elimination of sorption during transport, assuming the engineered cover provides no reduction in natural recharge rates, and assuming the carbon steel waste packages provide no barrier to release or transport. Because solubility limits were not applied in the compliance period analyses, they are not adjusted for the performance period analyses. The receptor characteristics are adjusted to be consistent with the climate state.

The screening analyses for the performance period results in an estimated all pathways peak dose of 40 mrem/yr at 30,000 years. A sensitivity analysis of the long-term infiltration rates is included to address uncertainty in future climate state. The licensee includes a comparison of the flux (g/yr) of naturally-occurring radionuclides from the disposal facility with those originating from natural sources (a nearby river). The fluxes from the facility are less than those from natural sources.

The licensee also develops a cost comparison of some engineered alternatives to the disposal facility and how they could impact performance period doses. Only technologies that would significantly increase cost or are unproven result in a significant decrease in projected impacts. Because the projected doses from the conservative screening analysis do not significantly exceed the compliance period dose limit, only a simple assessment of alternatives and their impacts is warranted.

The licensee performs barrier analyses to determine the most significant barriers of the system that are reducing releases. For the performance period, the licensee determines that dilution and dispersion during transport are very significant. In addition, solubility limits and sorption during transport could be very important, though they are not credited in the conservative screening analyses.

Because the compliance period scope was supplemented and conservatism was used in the analyses, the performance period analyses are sufficient to demonstrate that releases for the performance period have been reduced to the extent practical even if the estimated doses exceed the compliance period dose limit.

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Barrier analyses may involve estimating the performance of individual components or materials in the disposal system and providing the basis for the performance of the component or material. For example, if a robust engineered cover using durable rock is provided for erosion protection, estimation of the durability of the rock over the long-term may provide confidence in future performance without detailed landform evolution modeling. Estimation of the ages and stability of surrounding analogous landforms may be useful in providing inferential information about the long-term stability of the waste disposal site.

Quantitative analyses, such as extension of technical analyses calculations from the compliance period to the performance period, can provide estimates of the ability of the disposal system to reduce long-term impacts, if the scope of the compliance period analysis is sufficient for the performance period analyses.

6.3.2.1 Methods

Different methods are available to perform barrier and component analyses, including, but not limited to, one-off analyses, one-on analyses, and factorial designs (Esh et al., 2001; NRC, 2004a; Eisenberg and Sagar, 2000). Typically, the term “one-off” analysis refers to varying one parameter — in this case, one barrier — at a time. Barrier and component analyses are usually performed by isolating the performance of the barrier or component. The biggest challenge in performing these analyses is usually communicating what the results mean and how they should and should not be interpreted, because the calculations provide a hypothetical situation that may never arise (e.g., elimination of a geologic unit, or failure of all waste packages at a single instant).

Barrier and component analyses can be performed at different levels of resolution. Different levels of resolution can provide important detail to help focus the regulatory review. For example, representation of a disposal system as its components — an engineered cover, disposal vaults, wasteform, unsaturated zone, and saturated zone — can convey broadly the areas contributing to performance in mitigating risks. Refinement of that analysis, such as looking at individual layers in a multilayer engineered cover, may identify specific areas of performance.

Barriers and components in the disposal system (e.g., engineered barriers and disposal site) may reduce the magnitude of doses or change the timing of when doses could occur. The barrier and component analyses examine changes to both the magnitude of projected doses and the timing of those doses. Some barriers may impact both metrics, while others may impact only a single metric.

Environmental system models will include coupling or integration of components. The licensee will need to clearly identify how processes that may affect multiple components have been considered in the barrier and component analyses. In addition, disruptive processes and events may impact multiple barriers or components. Disruptive processes and events may be more important to consider during the performance period than during the compliance period because of the longer time during which they could occur. A licensee can perform barrier and component analyses after a disruptive event is assumed to occur to understand how the components of the disturbed system may be contributing to limit the impacts from the disruptive event.

In one-off analyses, the performance of a single barrier or component is neglected to understand the contribution of the barrier to performance when the system is operating under the anticipated range of conditions. Each barrier or component is analyzed in this manner, and the relative performance, such as the change in peak dose, is compared. For example, the contribution to performance of an engineered cover could be evaluated by setting the infiltration into the disposal system to a value that represents a natural recharge rate in the region of the disposal facility. If the engineered cover had other contributions to performance (e.g., reducing radon fluxes), those should also be eliminated in a one-off analysis. The results are best expressed on a relative basis, such as percent change, as the analyses may be unphysical. These types of barrier analysis calculations can have value because they clearly identify which elements of the system are contributing the most to performance and, therefore, which elements should have the most technical basis and most rigorous review effort applied to them.

Individual components or barriers may have a performance function redundant to that of other components or barriers. A one-off analysis result that indicates the performance did not change when a component was "turned off" could show that the barrier or component truly does not contribute significantly to overall performance. However, the resulting lack of estimated performance could indicate that a different barrier is providing a redundant functionality. A licensee can use different analyses, such as one-on and factorial designs, to complement the one-off type of barrier and component analyses to reveal this type of redundant functionality.

"One-on" analyses are those in which only a single barrier or component is represented to determine the potential performance of the barrier. A benefit of the one-on analysis is that it identifies the hypothetical maximum consequence that the waste could produce. However, the usefulness of this type of analysis is reduced when the likelihood for the hypothetical consequences becomes overly unlikely. One-on analyses can be useful in identifying when different barriers may be providing redundant performance functions. A disadvantage of one-on analyses is that they do not address the likelihood of the hypothetical result ever being achieved. If it is recognized that the purpose of the analyses is to understand how barriers could contribute, misinterpretation of the results can be avoided. The base case analysis, with all barriers and components present, represents the best estimate of disposal system performance, if the level of performance assigned has an adequate technical basis.

A licensee can use factorial designs to provide a more complete picture of the contribution of various barriers and components to the overall system performance. In the factorial design, all combinations of barriers being "on" or "off" are generated. This type of analysis can require significant resources, depending on the number of barriers and components in the system being evaluated and the computational expense of the models being evaluated. When a full factorial assessment is not practical, a licensee can consider a partial factorial assessment. The compilation and interpretation of results from a factorial barrier assessment are not straightforward. The relative change in performance will be much different depending on the number of barriers or components that may be active in the calculation. One way to overcome this problem is simply to rank the barrier contributions for each similar calculation. Example 6.2 provides additional detail on how this may be accomplished for a partial factorial assessment.

Barrier addition analysis is a process in which the hypothetical maximum consequence of the waste is generated. After the licensee estimates the maximum consequence, barriers or components can be added one by one until the full system is represented. A challenge in barrier addition analyses is that the sequence in which the barriers are added may influence the

results ascribed to a barrier. Barriers added early in the sequence are more likely to show large performance benefits than barriers added later in the sequence.

For barrier and component analyses to be most useful, they should be carefully performed by analysts who understand the components and barriers of the system. The level of underperformance assigned to a barrier or component, which is subjective, can influence the results and interpretation of the importance of that barrier or component. The underperformance ascribed may represent both the amount of degradation expected as well as pessimism in the estimate of performance. Licensees should clearly explain the underperformance and, if possible, should assign this underperformance based on the confidence in the understanding of the performance of the barrier or component. A barrier with a strong technical basis for its performance should be much less likely to not perform than one with a weak or limited technical basis. Although barrier and component analyses can be useful in understanding how a system may perform, the analyses may not correctly represent how the system is expected to perform.

6.3.2.2 Design, Site, and Overall Performance

Differentiation between barriers that are engineered (i.e., design) and those that are natural (i.e., site) can be useful in understanding performance and explaining to stakeholders why the system is expected to protect public health and safety. Generally, engineered components provide greater benefit at earlier times, and natural system components provide greater benefit at later times. The engineered design should be integrated into the natural site, which can make a clear separation less obvious. In addition, it is expected that natural system conditions will influence the performance of the engineered design. In some cases, the engineered design could influence the performance of the natural system (e.g., leaching of cement that impacts sorption in the unsaturated zone or erosion of surface soils at the toe of a slope). Engineering judgment can be used to classify barrier and component types if the analysis is transparent and traceable. Classification is the separation of the different barriers into classes based on the type of barrier (i.e., engineered or natural). Some phenomena may be difficult to classify, for example the chemical environment inside a waste container. The goal for the licensee is to apply the classification that provides the clearest understanding of overall system performance.

Example 6.2: A disposal system has five primary barriers, #1 through #5. A factorial barrier assessment is completed. The simulations are as follows:

#1	#2	#3	#4	#5
off	on	on	on	on
on	off	on	on	on
on	on	off	on	on
on	on	on	off	on
on	on	on	on	off
off	off	on	on	on
off	on	off	on	on

[full matrix not shown]

off	off	off	on	off
off	off	off	off	on

An overall performance measure is generated by calculating the rank of the analysis result relative to the rest of the results of that class (i.e., analyses with the same number of components turned "on" or "off"), then generating the average rank over all classes.

For example, the results for the one-off class are as follows:

#1	#2	#3	#4	#5
+5%	+30%	+1%	+200%	+17%

This results in the following ranks for the one-off class:

#1	#2	#3	#4	#5
4	2	5	1	3

The average change for each two-off combination is the following (any two-off that includes a #1 in the combination goes into the average change for #1):

#1	#2	#3	#4	#5
+57%	+89%	+132%	+310%	+37%

This results in the following ranks for the two-off classes:

#1	#2	#3	#4	#5
4	3	2	1	5

The average rank of each barrier averaged over each class provides a barrier importance measure:

#1	#2	#3	#4	#5	
4	2.5	3.5	1	4	(average rank for each barrier)

1

7.0 DEFENSE-IN-DEPTH

Defense-in-Depth:
The use of multiple, independent, and, where possible, redundant layers of defense so that no single layer, no matter how robust, is exclusively relied upon.

The NRC's safety philosophy has long included the concept of defense-in-depth. In 10 CFR 61.2, defense-in-depth is defined as "the use of multiple, independent, and, where possible, redundant layers of defense so that no single layer, no matter how robust, is exclusively relied upon." The purpose of defense-in-depth is to compensate for uncertainties. Consistent with the NRC's regulatory philosophy, the regulations at 10 CFR 61.10(c) require that the licensee describe the defense-in-depth protections that enhance the resiliency of the facility in complying with the performance objectives specified at 10 CFR 61.41 and 10 CFR 61.43.

Defense-in-depth protections are used to prevent, contain, or mitigate exposure to radioactive material according to the hazard present, the relevant scenarios, and most importantly the associated uncertainties. Defense-in-depth protections ensure that the risks resulting from the failure or degradation of some or all the barriers and controls, including failure or degradation caused by human error, are acceptable. These protections help to provide reasonable assurance that 10 CFR 61.41 and 10 CFR 61.43 performance objectives can be met, considering the uncertainties in projecting the behavior of the land disposal facility over both the operational and postclosure periods.

Licensees should describe the defense-in-depth protections that enhance the resiliency of the facility in complying with the performance objectives. To show that resiliency, the description should identify the use of multiple protections and describe their capabilities, including how the various barriers maintain independence and provide redundancy where possible. The description of the defense-in-depth protections can be principally drawn from risk insights derived from the results of 10 CFR 61.13 technical analyses (e.g., from parameters of the performance assessment such as barriers to limit contact with the waste). The 10 CFR Part 61 regulations are themselves structured to provide defense-in-depth such that demonstrating that the regulations will be met provides some level of defense-in-depth protection.

This chapter describes the information that a licensee should provide, and a reviewer should evaluate with respect to identifying defense-in-depth protections. Sections 7.1 and 7.2 discuss the NRC's defense-in-depth philosophy and elaborate on key concepts of the defense-in-depth regulatory philosophy as they apply to LLW disposal facilities, respectively. Section 7.3 provides guidance on identifying and describing defense-in-depth protections during the operational and postclosure phases of the land disposal facility life cycle.

7.1 Background on Defense-in-Depth

Defense-in-depth is a concept that has been used since at least the 1960s in the context of ensuring nuclear reactor safety. The philosophy is intended to deliver a design that compensates for uncertainties in knowledge of facility behavior, component reliability, or operator performance that might compromise safety. Application of defense-in-depth is usually for active systems.

The **Defense-in-Depth philosophy** is intended to deliver a design that compensates for uncertainties in knowledge of facility behavior, component reliability, or operator performance that might compromise safety.

In the context of nuclear reactor safety, defense-in-depth has traditionally focused on layers of protection to prevent accident initiators, contain radioactivity, and mitigate exposures through safety systems. The defense-in-depth concept has evolved from its early narrow application in the context of nuclear reactor safety to a more expansive application as an overall safety strategy for radioactive materials that includes the multiple barrier approach.

In the 1995 Policy Statement on the Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities (NRC, 1995a), the NRC recognized that complete reliance cannot be placed on any *single* element of the design, maintenance, or operation of a facility. The policy statement highlighted the need for redundancy in active safety systems and a multiple barrier approach to protect against releases. An essential property of defense-in-depth is the concept of successive barriers or layers. These barriers or layers are commonly represented within the NRC's regulatory framework in two different ways (Sorensen, et al., 1999). First, the NRC's framework requires the use of layers of protection, such as the prevention of accident initiators, the quick termination of accident sequences, and the mitigation of accidents that are not successfully terminated. Second, the NRC's framework requires the use of multiple physical barriers.

In 1999, SRM-SECY-98-144, "Staff Requirements—SECY-98-144—White Paper on Risk-Informed and Performance-Based Regulation," dated March 1, 1999, the NRC approved descriptions of many terms, including defense-in-depth (NRC, 1999d). The SRM describes defense-in-depth as an element of the NRC's safety philosophy that employs successive compensatory measures to prevent accidents or mitigate damage if a malfunction, accident, or naturally caused event occurs at a nuclear facility. The philosophy ensures that safety will not be wholly dependent on any single element of the design, construction, maintenance, and operation of a nuclear facility. The net effect of incorporating defense-in-depth into design, construction, maintenance, and operation is that the facility or system in question tends to be more tolerant of failures, external challenges, and uncertainty in the behavior of the facility.

More recently, in NUREG-2150, "A Proposed Risk Management Regulatory Framework," issued April 2012 (NRC, 2012c), the NRC has characterized defense-in-depth protections as part of the development of a holistic vision for facilities regulated by the NRC, including land disposal facilities. The NRC's risk-informed, performance-based characterization indicates that defense-in-depth protections ensure (1) appropriate barriers, controls, and personnel are present to prevent, contain, and mitigate exposure to radioactive material according to the hazard present, the relevant receptor scenarios, and the associated uncertainties and (2) that the risks resulting from the failure of some or all of the established barriers and controls, including human errors, are maintained acceptably low.

The regulation in 10 CFR 61.10(c) requires licensees to describe defense-in-depth protections that enhance the resiliency of the facility in complying with the performance objectives. The regulations in 10 CFR Part 61 also implicitly incorporate the concept of defense-in-depth into the regulatory framework. Implicit defense-in-depth provisions of the regulations include multiple performance objectives, as well as requirements for site suitability, site design, facility operation, site closure, environmental monitoring, waste acceptance, land ownership and institutional control, and financial assurance.

7.2 Defense-in-Depth Concepts for a Land Disposal Facility

The NRC's use of the defense-in-depth philosophy is intended to deliver a design whose performance is resilient to uncertainties in knowledge of facility behavior, component reliability, or operator performance that might compromise safety. While justifiably important for nuclear facilities with large potential risks, it is similarly important for land disposal facilities for radioactive wastes, which can contain relatively large radiological hazards. Disposal facilities, being the endpoint of the nuclear fuel cycle, are responsible for containing and limiting the release of the radioactivity for time periods far into the future. The use of defense-in-depth protections is a prudent approach to managing the uncertainty associated with estimating performance far into the future to ensure the safe disposal of radioactive wastes.

As defined at 10 CFR 61.2, defense-in-depth relies upon multiple, independent, and, where possible, redundant layers of defense so that no single layer, no matter how robust, is exclusively relied upon. Multiple layers provide confidence that if an individual layer fails or underperforms, other layers of defense would be available to protect health and safety and the environment. Redundant layers provide confidence that should an individual layer fail or underperform, another layer will be available to provide similar capabilities to protect health and safety and the environment. Licensees should interpret redundancy from an overall system performance viewpoint and not from the functionality of individual barriers or systems. Independent layers enhance confidence that the layers of defense are less likely to fail or underperform because of common-cause failure modes in which a single event or process can diminish the capabilities of each layer simultaneously. Multiple, independent, and redundant layers are intended to provide a margin of safety to account for uncertainty in the evolution of the land disposal facility over time. The margin of safety needed depends on the hazard presented by the waste emplaced in the land disposal facility and by the potential risk of harm to health and safety and the environment.

The performance of most land disposal facilities is the product of engineered systems and the site geology. A licensee should emphasize those features as well as requirements for site suitability, site design, facility operation, site closure, environmental monitoring, waste acceptance, land ownership and institutional control, and financial assurance when describing defense-in-depth protections used to mitigate the impact of uncertainties.

7.2.1 Multiple Layers

Reliance on multiple layers of defense ensures that safety will not be wholly dependent upon any single element of the design, construction, maintenance, or operation of the land disposal facility. An essential part of the technical analyses performed for a land disposal facility is the identification and evaluation of the impacts of uncertainties. Uncertainties can be difficult to identify, and some future uncertainties may not be identified. Defense-in-depth protections may be used to mitigate those uncertainties. Multiple layers of protection use numerous, diverse protection mechanisms or actions to ensure safety. Layers of defense can consist of attributes including physical or chemical barriers to radionuclide release, appropriate controls that help ensure that the physical barriers (e.g., fences) perform as intended, and trained and qualified personnel who are focused on safety (see Figure 7-1).

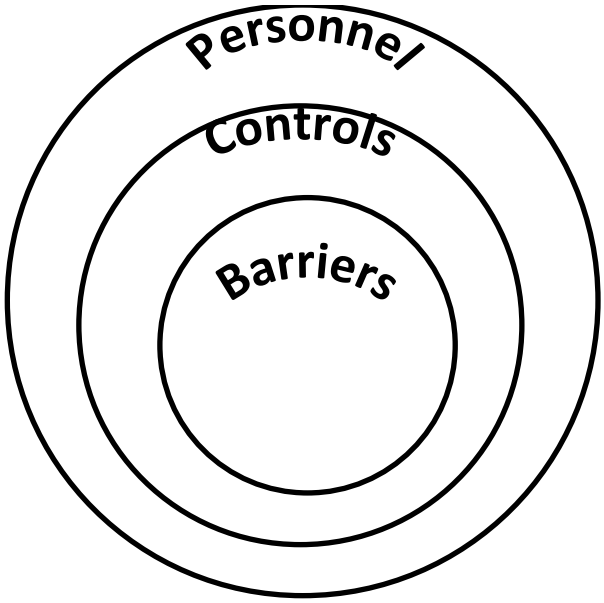


Figure 7-1 Multiple Layers of Defense

Some of these layers apply to operational safety whereas others apply to postclosure safety. Some types of layers such as controls and personnel can act as safety functions only before and during the institutional control period. A licensee may use multiple different types of the same layer of defense.

Each of the layers has a capability or safety function that is intended to reduce releases and exposures to workers and the public. In the context of waste disposal, especially for postclosure safety, a layer of defense does not need to be 100 percent effective to be considered as providing defense. The function performed by an individual layer may be active, passive, or in some cases, a layer may provide both active and passive safety functions. Active safety functions are those that require action by the licensee to monitor and maintain protection. Active safety functions do not apply for postclosure safety after the institutional control period. Active safety functions would apply during operations, such as to mitigate the impacts associated with accidents when disposing of GTCC waste. For example, an air filtration system requires continual maintenance, as well as an available energy source to ensure that its functionality to remove radioactive particulates is available when needed. An air filtration system may be used in a temporary storage area to mitigate potential impacts from a fire involving GTCC waste. Passive safety functions do not require ongoing activity or external energy inputs from the licensee to provide protection. For instance, a wasteform, once created, typically would not require ongoing maintenance to perform its safety function (i.e., limit the release of radionuclides), though its safety function would likely degrade over time. The role of barriers, controls, personnel, and their associated safety functions in demonstrating defense-in-depth for a land disposal facility is discussed in more detail below. For land disposal of LLW, passive layers of defense are more appropriate for the postclosure period because they do not require ongoing maintenance and monitoring. Active layers of protection are applicable only before the institutional control period.

When identifying barriers, controls, or personnel relied upon to mitigate uncertainties, licensees should clearly describe the functionality or capability provided by the barrier, control, or personnel to achieve the performance objectives and provide defense-in-depth protections. The description of the defense-in-depth protection should include a technical basis for the protection and associated uncertainty in the function of the barrier, control, or personnel. In some cases, the defense-in-depth protection may only be necessary for a specific timeframe. For example, a licensee identifies a waste container as a barrier to waste release for short-lived radionuclides. The licensee should specify the time period over which the protection provided by the waste container is necessary to demonstrate that the performance objectives are met. Figure 7-2 depicts approximate time periods over which different types of defense-in-depth protections may be appropriate.

7.2.1.1 Barriers

For the land disposal of LLW, barriers can take different forms, such as a container, a wasteform, a wall, or a restricted area of land. Barriers can be either engineered or natural. Barriers are intended to improve the licensee's ability to demonstrate that the land disposal facility will meet 10 CFR Part 61 performance objectives and add to defense-in-depth. In the context of disposal, barrier safety functions, which are also described as barrier capabilities, are typically intended to (1) prevent or limit the contact of water with the waste, (2) limit the release or rate of release of radioactivity from the waste, or (3) prevent or limit biotic contact with the waste (for inadvertent intrusion). For normal operations and abnormal events (e.g., accidents), barrier safety functions are typically designed to prevent, contain, and mitigate exposure to radioactive material.

Engineered barriers are man-made features that are designed to mitigate the effect of anthropogenic events and natural events and processes on the performance of the disposal facility and to limit human activities that may expose individuals to radiation or initiate or accelerate release of radioactivity from the waste through environmental pathways. The intent of engineered barriers is to improve the performance of the land disposal facility and add to the defense-in-depth provided by the facility design and construction. Examples of engineered barriers include (1) engineered surface covers, which are designed to limit infiltration into and erosion of disposal units and (2) waste containers and wasteforms, which may provide shielding to site workers during emplacement and preclude or limit the release of radionuclides from the disposal units.

Natural barriers are those inherent to the disposal site that limit exposures to the waste. Examples of natural barriers include the climatic conditions or the hydrogeologic units of the disposal site. Limited infiltration associated with an arid climate may result in long travel times of radionuclides from the waste to underlying groundwater. Likewise, hydrogeological units may slow the movement of radionuclides from the disposal site because of sorption processes.

7.2.1.2 Controls

A control can be an apparatus or mechanism that, through its manipulation or administration, protects public health and safety and the environment around the land disposal facility. During operations, controls are intended to prevent, contain, and mitigate exposure to radioactive material at or in the vicinity of the land disposal facility. Operators employ controls to ensure the performance of barriers, to prevent activities, or to ensure that certain activities occur during the operational period. Institutional controls are used after closure of the land disposal facility.

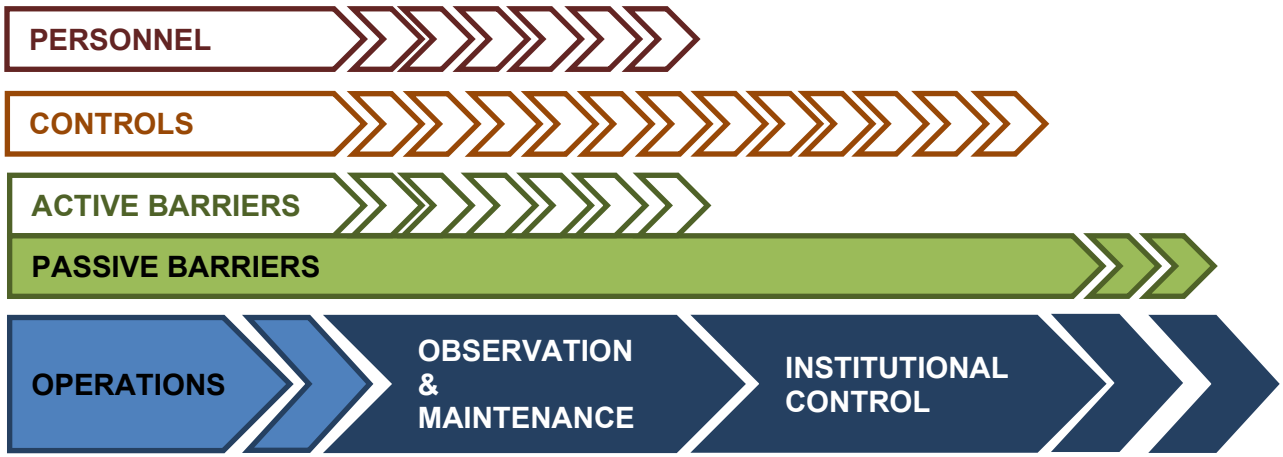


Figure 7-2 Land Disposal Facility Life Cycle and Timeframes for Defense-in-Depth Layers (Duration of life cycle timeframes are not to scale. Dark blue timeframes are the postclosure period.)

Controls can largely be classified as either engineering or administrative controls. Engineering controls include any man-made apparatus that is designed to ensure the performance of a barrier or provide protection independent of a barrier. For example, an engineering control may include something as simple as a visible or audible alarm as part of access control to restricted areas or more complex systems such as a fire alarm and suppression system to limit the damage from a potential fire. Administrative controls encompass a wide array of managerial mechanisms. Administrative controls include provisions related to organization and management, procedures, record keeping, material control and accounting, waste acceptance, and management review. Administrative controls can also include legal mechanisms such as land ownership, as required by 10 CFR 61.59(a).

7.2.1.3 Personnel

Properly trained and qualified personnel are necessary to ensure proper design, installation, operation, or administration of barriers and controls that prevent, contain, or mitigate exposure to radioactive materials. Properly trained and qualified personnel also ensure that human errors that can lead to failure of established barriers and controls are kept acceptably low and safety is maintained. Personnel may be able to mitigate exposures in the event of an accident. For example, properly trained emergency responders can mitigate potential releases of radioactivity from the land disposal facility in an accident or emergency, such as suppressing a fire. Because reliance on personnel inherently involves activity, personnel always provide active safety functions. Designated personnel may be an appropriate layer of protection during operations; however, the NRC intended in the regulations at 10 CFR Part 61 that active maintenance and monitoring of the site after closure is not relied upon for safety. Therefore, reliance on personnel for defense-in-depth protections after closure is not appropriate.

7.2.2 Independent Layers

In addition to multiple layers of protection, which are discussed in Section 7.2.1, the definition of defense-in-depth in 10 CFR Part 61 requires that the land disposal facility incorporate independent layers of protection. An independent layer should provide a safety function that is not dependent upon other layers of defense to perform the safety function. For instance, a licensee may use an engineered surface cover as a barrier to limit the flow rate of water contacting the waste and a wasteform as another barrier to limit the solubility of certain radionuclides in water contacting the waste. If the engineered cover were to fail, resulting in a higher flow rate of water into the disposal unit, the wasteform would still be expected to maintain low concentrations of solubility-limited radionuclides.

In some cases, multiple layers of defense may appear to be independent, but the safety functions performed by each layer may be dependent. For example, a land disposal facility may dispose of waste in a metallic container with safety functions to limit the contact of water with the wasteform and limit the release of radionuclides. In addition, a licensee may dispose of waste in a metallic wasteform that is also relied upon to limit the release of radionuclides. The waste container and wasteform may appear to provide independent functionality; however, certain chemical environments could cause degradation to both the waste container and wasteform simultaneously. The degradation of the container may directly result in more rapid degradation of the wasteform, resulting in a dependency between the container and wasteform to limit releases of radionuclides from a disposal unit.

7.2.3 Redundant Layers

The definition of defense-in-depth states that layers of protection are redundant, where possible. The use of redundant layers increases confidence that safety is not reliant upon any single layer. Redundancy is the duplication of capabilities of layers of defense from a system perspective, to prevent the failure of the entire disposal system if a single item or component relied upon for safety fails or provides a safety function that is less than expected. Licensees should demonstrate, where possible, that the land disposal facility has redundancy for key functions relied upon for safety. Redundancy from a system perspective means that if a layer of defense were to fail, there are other layers of defense that would compensate for the failure or degradation of the layer of defense.

Redundancy is desirable but is only necessary to mitigate uncertainties for high-risk scenarios (e.g., a fire during operations for some types of GTCC waste). Redundancy may go beyond implementing multiple layers. Redundancy may necessitate that the safety function performed by a layer be duplicated or even triplicated or more, where possible, for safety-critical components in high-risk accident scenarios. The duplication of a safety function may occur within the same layer (e.g., as a redundant component) or in other layers incorporated into the siting, design, construction, maintenance, or operation of the disposal facility. For instance, different stratigraphic units beneath a disposal facility may each sufficiently retard the migration of key radionuclides, thereby providing redundancy of the safety function should the licensee's understanding of the primary unit's sorptive capabilities prove incorrect. In this example, licensees should provide reasonable assurance that groundwater flow occurs through the stratigraphic units and that preferential flow pathways that could minimize the water contact with the stratigraphic units would not be limited.

7.2.4 Safety Margin

Safety margin is the capability remaining after demands are placed on the functionality of the layer or system. In some high-risk scenarios such as certain types of accidents (e.g., fires), it may be necessary to ensure that an adequate safety margin has been provided. For instance, licensees may impose a design requirement on an engineered barrier that provides confidence that the barrier's capability exceeds the demand imposed (e.g., by using a pre-determined safety factor). Safety factors are commonly used in engineering design. This excess capability would provide confidence that safe conditions are maintained during normal operations considering uncertainties or in the event of abnormal occurrences, accidents, or disruptive events. The concept of the safety margin is to ensure that safety functions are sufficient to account for uncertainty in the characterization of the demands, as well as the robustness of the functionality. Safety margins can be determined for an individual layer or the entire disposal facility, but the margin for the disposal facility should be the primary interest for licensees and regulators.

Meeting the performance objectives, as demonstrated by 10 CFR 61.13 technical analyses, demonstrates a level of safety margin for an LLW disposal facility because the dose limits established in the performance objectives are purposely set below the public dose limit specified in 10 CFR Part 20 (to account for the possibility of exposure from multiple facilities). Understanding the safety margin for abnormal occurrences, accidents, or disruptive events is important to provide confidence that the performance objectives can be met, even if less likely, but plausible, disruptive or accident scenarios occur. Because abnormal occurrences, accidents, and disruptive events are plausible, but generally less likely, licensees may account for the likelihood of these events or processes occurring to determine the expected safety margin. For the postclosure periods, the licensee should demonstrate that for plausible abnormal occurrences, accidents, or disruptive events the potential doses would remain below those for which intervention would be necessary if they were to occur today. Guidance on these levels is discussed in more detail in Section 7.3.3.2 for each postclosure period. Section 2.5.4 discusses the terms used to describe scenarios to be included in 10 CFR 61.13 technical analyses (i.e., reasonably foreseeable, less likely, but plausible, and implausible). These terms and types of scenarios can also be used to describe the capabilities of defense-in-depth protections and the supporting bases.

7.2.5 Risk-Informed Approach

Defense-in-depth should be applied in a risk-informed manner. As the hazard from the waste increases, more robust defense-in-depth protections may be needed to account for uncertainty in the performance of the barriers, controls, and personnel used to maintain safety. Alternatively, less robust layers of defense may make additional redundancy necessary, depending on the uncertainty in the performance provided by the layer(s) and the risk posed by the waste. The timeframe for which the defense-in-depth protections are provided should be appropriate for the time period over which significant risks are possible. As uncertainty in the functionality and reliability of barriers, controls, or personnel increases, additional layers of defense may be needed to provide confidence that safety can be maintained. For instance, for waste containing significant quantities of long-lived radionuclides, additional layers of defense may be needed to account for the uncertainty associated with projecting performance over very long time periods. For shorter-lived hazards, fewer layers of protection may be sufficient.

7.3 Defense-in-Depth Protections

The regulation at 10 CFR 61.10(c) requires the application to describe defense-in-depth protections that enhance the resiliency of the disposal facility in complying with the 10 CFR 61.41 and 10 CFR 61.42 performance objectives. The applicant should identify defense-in-depth protections, describe the capabilities of each protection relied upon to maintain safety, and provide a basis for the capabilities of each defense-in-depth protection. Licensees should identify the defense-in-depth protections and describe their capabilities in the context of the performance objectives in 10 CFR Part 61, Subpart C, for protection of the general population (10 CFR 61.41) and protection of individuals during operations (10 CFR 61.43). This section describes acceptable approaches licensees may take to identify defense-in-depth protections, describe the capabilities of each defense-in-depth protection relied upon for safety, and justify the capabilities of defense-in-depth protections.

Because the NRC's use of the defense-in-depth philosophy is intended to deliver a design that accounts for uncertainties in knowledge of facility behavior, component reliability, or operator performance that might compromise safety, licensees should also describe, either qualitatively or quantitatively, the margin available to maintain safety. Defense-in-depth protections will typically be discussed in the system description (see Section 2.4) and considered in technical analyses, such as the performance assessment, in order to demonstrate that the performance objectives will be met. Licensees should be able to draw, principally, upon the results and risk insights gained from 10 CFR 61.13 analyses to identify and describe defense-in-depth protections and the safety margins that the protections provide.

7.3.1 Identification of Defense-in-Depth Protections

Licensees should identify the defense-in-depth protections that provide resiliency of the disposal system and mitigate the impacts of uncertainties. The identification of these protections, or layers of defense, should demonstrate that multiple barriers, controls, or personnel are used at the land disposal facility. Section 7.2.1 describes barriers, controls, and personnel in the context of defense-in-depth protections for a land disposal facility. Licensees should indicate when layers are included for redundancy and when it is not possible to provide a redundant layer for significant defense-in-depth protections.

The defense-in-depth protections for the operational versus the postclosure time periods are likely to be different because active maintenance of the site after closure is not anticipated. Following closure of the land disposal facility, defense-in-depth protections shift from a collection of active and passive barriers, controls, and personnel that are used during operations to reliance on passive barriers and institutional control. This shift is necessary because the regulations at 10 CFR Part 61 specify that active maintenance of the disposal site beyond monitoring, surveillance, and minor custodial activities cannot be relied upon after the period of postclosure observation and maintenance, and without ongoing maintenance, active barriers and active institutional controls cannot be relied upon indefinitely.

Defense-in-depth protections for the postclosure period may include, but are not limited to the following:

- engineered features (e.g., surface covers, wasteforms, and containers) and design considerations such as limiting the radionuclide inventory and the depth of waste disposal
- natural characteristics (e.g., hydrogeology) of the disposal site that are intended to contain the waste
- controls such as institutional controls, which are designed to limit access to the disposal site for a limited period of time, or government land ownership

The identification of defense-in-depth protections should be consistent with the barriers, controls, or personnel relied upon in 10 CFR 61.13 technical analyses. In some cases, the layers identified to describe defense-in-depth may not be represented in 10 CFR 61.13 analyses. This may often be the case for personnel identified as a defense-in-depth protection. In these cases, licensees should provide a basis for identifying the layer as a defense-in-depth protection. In other cases, there may be less significant barriers, controls, or personnel that provide demonstrable safety functions in 10 CFR 61.13 technical analyses. A licensee does not need to identify a comprehensive list of layers of defense. Rather, a licensee only needs to identify that there are multiple, independent, and, where possible, redundant layers of defense, so that no single layer is exclusively relied upon.

As part of the identification of defense-in-depth protections, licensees should also identify which layers of defense are included for redundancy. Licensees should identify the primary layer and redundant layers that are expected to be relied upon in the event the primary layer degrades or fails. Licensees may associate a secondary layer as a redundancy for multiple primary layers and do not need to limit the association of a redundant layer with a single primary layer. In some cases, multiple layers can provide redundancy without subordinating one of the layers in the identification. The classification of primary versus secondary in this case may be in name only. Licensees should identify the layers for which a redundant capability or safety function is not possible or necessary and should provide a basis demonstrating that a redundancy is not possible. The next section discusses acceptable approaches for demonstrating that redundancy is not possible.

Reviewers should confirm that the licensee has identified multiple layers of defense. While licensees do not need to identify a comprehensive list, reviewers should confirm that the layers of defense identified by the licensee are consistent with the layers represented in 10 CFR 61.13 technical analyses. Reviewers should evaluate whether the most significant barriers, controls, or personnel that are relied upon for meeting the performance objectives are identified as defense-in-depth protections.

Reviewers should confirm that the licensee has identified at least one redundant and one independent layer of defense, though there may be different layers that provide the redundancy and independence. Reviewers should confirm that the identification clearly associates a redundant layer with each primary layer of defense. If a redundant layer is not identified, reviewers should confirm that the licensee has provided an adequate basis to support that a redundant layer is not possible for significant defense-in-depth protections. Additional detail on an adequate basis to demonstrate that a redundant layer is not possible is included in the next

section. Reviewers should confirm that the licensee has identified which layers ensure independence. Independence should be ensured for the disposal system. In other words, sufficient independence is demonstrated when common-cause failures are implausible or result in acceptable consequences. The next section provides guidance on ensuring that the safety functions for independent layers are not subject to common-cause failures.

7.3.2 Description of Capabilities (Safety Functions)

Licensees should describe the capabilities, or safety function(s), provided by each layer that is identified as a defense-in-depth protection. Licensees should describe the capability of the individual layers identified to provide resiliency and mitigate uncertainties. In addition, the description should include a technical basis supporting the capability of the defense-in-depth protection. The technical basis supporting the capability can be drawn from the technical basis used to support the layer's representation in the 10 CFR 61.13 analyses. Section 2.2 provides guidance on developing support for the technical analyses, and specific guidance for each of the 10 CFR 61.13 analyses is provided in their respective sections of this document (Chapters 3.0 through 6.0).

For a land disposal facility, the safety functions of defense-in-depth protections are often focused on (1) limiting the contact of water or individuals with the waste, (2) limiting the release of radionuclides from the waste, or (3) limiting the rate of radionuclide transport through the site to the general environment. In addition, safety functions cover other considerations such as maintaining structural stability. The safety function(s) of a layer of defense may vary depending on the performance objective. In addition, a layer of defense may provide more than one safety function. Licensees should identify the capabilities the layer provides.

If a layer of defense is included for redundancy, the licensee should indicate the other layers of defense the redundant layer is intended to support. The capabilities of the redundant layer need only be comparable, not identical, to the capabilities provided by the primary layer of defense. For example, a waste container may be identified as a primary layer of defense during the postclosure period to limit the release of radionuclides. A second waste container, while providing an identical redundancy to limit the release of radionuclides, would typically be neither practical nor necessary. Rather, other layers of defense, such as natural site characteristics, that also limit or slow the release of radionuclides from the disposal site, could provide redundancy for the waste container because the other layers offer comparable safety functions in terms of radionuclide release to the waste container.

If a licensee determines that redundancy for a significant defense-in-depth protection is not possible, the licensee should provide a basis supporting that determination. In deciding whether a redundant layer is possible, licensees may consider physical, technological, or economic limitations. However, the additional cost of limiting inventory is typically not expected to be overly burdensome from either a physical, technical, or economic consideration and is expected to provide the greatest certainty that significant uncertainties can be mitigated. Therefore, licensees and reviewers should consider inventory limits as added controls to ensure that significant uncertainties are mitigated when other redundant layers of defense may not be possible for technological or economic reasons. Regardless of the basis for demonstrating that redundancy is not possible for a primary defense-in-depth protection, the licensee should provide justification that reasonably foreseeable and plausible scenarios would not result in significant uncertainties, should they occur, when it is not possible to provide redundancy for a significant defense-in-depth protection. Licensees can draw risk insights from the results of the

10 CFR 61.13 analyses to support a justification that significant uncertainties would not result for reasonably foreseeable and plausible scenarios.

Licensees should describe whether a safety function is independent or dependent upon the safety functions of other layers. If a layer's safety function depends on another layer, the licensee should identify the other layer(s) in each layer's safety function description. If a layer's safety function is independent of other layers' safety functions, the licensee should provide justification that reasonably foreseeable and plausible common-cause failure scenarios would not result in significant consequences, should they occur. Alternatively, licensees could identify additional layers of defense, which would render the likelihood of the common-cause failure scenario as implausible.

Licensees should ensure that the description of the safety function is consistent with the representation of the layer of defense in the 10 CFR 61.13 analyses. For instance, if a waste container is identified as a layer of defense used to provide confidence that the performance objective for protection of the general population can be met, the description of its safety function should be consistent with its representation in the performance assessment. If the waste container is relied on to limit water contact with the waste and it is modeled as degrading over time in the performance assessment, the licensee should identify the degradation of its safety function as part of the description for defense-in-depth.

In some cases, layers of defense may not be amenable to representation in one of the 10 CFR 61.13 analyses (e.g., the response of personnel to an emergency such as a fire). In these cases, licensees should document the safety functions performed by the layers for demonstrating defense-in-depth. For the emergency personnel example, this may include written procedures for training and emergency response, staffing plans, or agreements with local emergency responders.

Each layer of defense is likely to have a time over which it will be designed to perform its intended safety function. For instance, the safety functions of engineered barriers would typically be expected to degrade or fail at some point in the future. However, some site characteristics may continue to provide safety functions indefinitely. The licensee should describe and justify this time in its technical basis supporting the safety function of the layer. If the time over which the layer is intended to perform its safety function significantly exceeds relevant experience, the licensee should provide additional support that the layer will likely achieve its goal. Additional support could include the use of additional redundant layers of defense.

Licensees should describe the uncertainty in a layer's ability to perform its intended safety function in either qualitative or quantitative terms. The description of the uncertainty should be generally consistent with the layer's representation in 10 CFR 61.13 analyses. Licensees should identify heterogeneity and variability in the behavior or reliability of the barrier, control, or personnel, as well as challenges to the functionality that result from plausible scenarios. Challenges should include both reasonably foreseeable scenarios expected during normal operations, as well as less likely, but plausible, scenarios such as abnormal conditions, accidents, or disruptive events.

Reviewers should evaluate the description of the capability, or safety function, of each layer to ensure that it clearly describes how the layer provides defense-in-depth protections and which performance objective the safety function supports. Reviewers should confirm that the

description addresses whether the safety function is included for redundancy and whether the safety function is independent of or dependent on another layer's safety function.

Reviewers should confirm that the safety function for a layer identified as redundant is linked to the layer it is intended to support. Reviewers should confirm that the redundant layer's safety function provides a comparable, though not necessarily identical, capability to the primary layer's safety function. If a licensee is not able to identify another barrier that could provide a comparable safety function for redundancy, reviewers should confirm that a comparable safety function is not possible and that the lack of redundancy would not result in significant uncertainties that cannot be mitigated.

In addition to evaluating the licensee's basis for a redundant safety function, reviewers should evaluate whether the layers identified as independent are subject to common-cause failures based on reasonably foreseeable scenarios, as well as less likely, but plausible, scenarios. Plausible common-cause failures may result from a common initiating event such as a seismic event that fails two barriers simultaneously. Plausible common-cause failures may also result from a cascade of processes such as when a geochemical environment leads to the degradation of a barrier whose degradation produces an environment that leads to the degradation of a different barrier, which may not have been affected by the initial aggressive environment.

Reviewers should confirm that the licensee's safety function descriptions discuss the time over which the layer's safety function is intended to perform and the uncertainty in whether the layer can perform the safety function. For layers of defense that are expected to perform for time periods that significantly exceed relevant experience, reviewers should confirm that the licensee has provided justification. Reviewers should coordinate the review of the safety function descriptions with reviews of 10 CFR 61.13 analyses to ensure that the layers and their safety functions have been represented in the analyses consistent with the licensee's description of the safety function for defense-in-depth. If redundant layers are expected to perform over different periods of time than the primary layers, reviewers should confirm that the licensee has identified other layers to ensure that redundancy, where possible, is provided for the entire time period of interest or that the consequences would remain acceptable should redundancy not be possible for the entire time period.

Reviewers should confirm that the licensee has described the uncertainty in each layer's ability to perform its intended safety function. Reviewers should coordinate their review of the description of uncertainty with the reviews of the 10 CFR 61.13 analyses to ensure that the uncertainty descriptions are generally consistent, and that the licensee described the heterogeneity and variability in the behavior or reliability of the barrier, control, or personnel. The description should also address challenges to the functionality provided by a barrier, control, or personnel that result from plausible scenarios.

7.3.3 Basis for Capabilities

After describing the safety function provided by each layer identified as a defense-in-depth protection, licensees should describe how the layers will mitigate uncertainties and how no single layer will be relied upon exclusively. The detail provided by licensees in describing the technical basis should be risk-informed. For layers of defense that provide one or more significant safety functions, licensees should describe the basis for the safety function provided by the layer of defense.

Licensees can use a variety of methods to demonstrate that safety functions will be achieved and that no single layer will be relied upon. The methods may be quantitative, qualitative, or a combination of the two approaches. Quantitative approaches attempt to assign numerical values to the performance of the layer and the resulting safety margin provided by the layers of defense relied upon for safety. Qualitative approaches describe the safety functions and the resulting safety margin of the various layers of defense.

Licensees may use different approaches to develop the basis for the performance of layers for different phases or even within a phase of the land disposal facility's life cycle. For instance, while some layers of defense used during operations may be amenable to quantification, other layers of defense may not be, such as the safety margin provided by an emergency response plan to mitigate the consequences of an accident. Typically, licensees should be able to use the results of 10 CFR 61.13 analyses to demonstrate that the layers will mitigate significant uncertainties. For instance, licensees should draw risk insights from the performance assessment and operational safety assessment (see Appendix G, Section G.6.1) to demonstrate that no single layer is relied upon exclusively over the various time periods.

Licensees may use the results of analyses, such as barrier analyses including one-off or what-if types of analyses, to demonstrate that adequate independence and, where possible, redundancy are provided and that no single layer is relied upon to mitigate uncertainties. The results of these analyses can be used to demonstrate that if any single barrier fails to perform, another barrier is available to provide a similar and adequate function. The results of 10 CFR 61.13 analyses can also be used to demonstrate that a lack of redundancy combined with common-cause failures would not result from reasonably foreseeable or less likely, but plausible scenarios. If a lack of redundancy or common-cause failure were to occur for plausible scenarios, the licensee should demonstrate that the consequences would not be so large as to require intervention today. The following sections discuss consequences that may require intervention today.

The life cycle of a land disposal facility can be divided into two broad phases: the operational phase and the postclosure phase. The operational phase is the time during which the licensee is constructing the facility, receiving waste for disposal, or preparing the facility for closure. The postclosure phase extends from the cessation of operations far into the future depending upon the type of waste accepted for disposal. At a minimum the postclosure phase includes the compliance period. For facilities disposing of waste containing significant quantities of long-lived radionuclides, the postclosure phase would also include the performance period. Because of the differences in length of the time periods for the operational and postclosure phases, the identification and description of defense-in-depth protections may be different for each phase.

The following sections describe considerations for licensees and reviewers for the two major life cycle phases of a land disposal facility and provide guidance on describing a land disposal facility's defense-in-depth protections.

7.3.3.1 Operational Period

During the operational phase of the land disposal facility, defense-in-depth protections may be similar to those used at other nuclear facilities that present similar hazards. Defense-in-depth protections during operations may include, but are not limited to (1) the selection and use of facility capabilities, such as structures, systems, and components of the facility design,

(2) programmatic processes, such as the processes of constructing, operating, maintaining, testing, and inspecting the facility, as well as processes that ensure facility safety through its operational lifetime, and (3) risk-informed strategies that manage the risks of accidents, including the strategies of accident prevention and mitigation.

During the operational period, licensees can draw upon risk insights gained from the analyses used to demonstrate compliance with the performance objectives for protection of individuals during operations (i.e., 10 CFR 61.43) to support the identification and description of the defense-in-depth protections that are included. For instance, the land disposal facility would use procedures and engineering controls as part of a radiation control program, required by 10 CFR 20.1101, "Radiation protection programs," to minimize occupational doses and doses to members of the public. In this example, the licensee should identify the procedures and controls that are relied upon for safety if plausible scenarios that challenge safety functions occur, describe the safety functions associated with the procedures and controls and when those safety functions are expected to be necessary, and describe the uncertainty in the safety functions to perform adequately.

Licensees should examine whether the layers of defense, their associated safety functions, and their uncertainty ensure that uncertainties in performance during operations will be mitigated by limiting the likelihood or consequences associated with reasonably foreseeable and less likely, but plausible scenarios. For facilities or activities with more significant hazards (e.g., disposal of high-activity waste) that could result in significant potential exposures to workers or the public during operations, additional redundancy, where possible, and a stronger basis for independence among the layers may be necessary. For the purposes of defense-in-depth, reviewers should consider potential exposures during operations as significant when they are expected to exceed the occupational dose limits or the dose limits for members of the public specified in 10 CFR Part 20, Subpart C, "Occupational Dose Limits," and Subpart D, "Radiation Dose Limits for Individual Members of the Public," respectively. For plausible scenarios with significant exposures, licensees should consider additional layers of defense, further redundancy, and where possible, improved independence or reduced uncertainty in the safety function of the layers.

7.3.3.2 Postclosure Period

Licensees applying defense-in-depth protections for land disposal facilities should recognize differences between operating facilities and closed land disposal facilities. While waste is being disposed and before a land disposal facility is closed, defense-in-depth protections can be provided by active or passive safety systems commensurate with the hazard and complexity of the activities, as is the case in other operating nuclear facilities. Licensees applying defense-in-depth measures to ensure land disposal facility performance for long timeframes following site closure, however, must account for the difference between an operating land disposal facility that has active safety systems and the potential for active control and intervention and a closed land disposal facility, which relies upon passive barriers. A closed disposal site is a passive system, and assessment of its safety over long timeframes is best evaluated through consideration of the relative likelihood of threats to its integrity and performance. With respect to the long-term performance of the disposal site, and in particular for the disposal of long-lived radionuclides, defense-in-depth is provided through the diversity and capabilities of the components and attributes of the disposal site (e.g., wasteform, container, engineered features, depth of the disposal unit below the land surface, hydrologic and geochemical characteristics).

Diversity in the capabilities of the components and attributes of the disposal site and its design increases the resilience of the disposal site to contend with unanticipated degradation or external challenges. This diversity also compensates, in part, for uncertainties in the long-term estimation of disposal site performance . The NRC staff continues to hold that each layer of defense must make a definite contribution to the isolation of the waste, so that the staff can find with reasonable assurance that no single layer of defense will be relied upon exclusively to achieve the overall safety objectives over the compliance period. Disposal of LLW is predicated on the expectation that attributes of the disposal site, in combination with engineered features, will minimize the migration of radionuclides away from the disposal site. However, the capabilities of site characteristics and engineered features are subject to many uncertainties. Engineered features generally are considered more durable over short time periods as compared to periods longer than a few hundred years when uncertainties in degradation rates and natural events may be more significant. The NRC staff expects that licensees will rely on both the natural site characteristics and the engineered features to provide defense-in-depth protections and reasonable assurance that the overall performance of the disposal site will be adequate over long time periods.

As a result of uncertainty in the behavior of the layers of defense during the postclosure period, licensees may also need to consider additional controls to ensure that the barriers relied upon to mitigate uncertainties will perform adequately. These controls may include QA controls during the design, construction, operation, and maintenance of engineered barriers (e.g., engineered surface covers, wasteforms, or containers) or additional inventory controls to limit the amount of waste disposed at the disposal site (see Chapter 8.0). These controls are applied during operations to provide confidence that the layer of defense will help mitigate uncertainties during the postclosure period. QA applied to the implementation of engineered barriers should contribute to reduced uncertainty in barrier performance in the postclosure period. Diversity in the capabilities of the passive safety functions provided by the disposal site (e.g., wasteform, container, engineered features, depth of disposal unit below the land surface, hydrologic and geochemical characteristics of the disposal site) increases the resilience of the disposal site to unanticipated failures or external challenges and compensates for uncertainties in the long-term estimation of performance of the disposal site.

For the postclosure period, licensees can draw on risk insights gained from the results of the 10 CFR 61.13 technical analyses (e.g., performance assessment, intruder assessment) to identify that defense-in-depth protections are included and to describe their capabilities. The postclosure period is divided into up to two timeframes: the compliance period and, if required, the performance period. Depending on the waste received, the performance period may not apply, and defense-in-depth protections may not be needed for the performance period (i.e., analyses that extend beyond 10,000 years). Because 10 CFR Part 61 does not allow a licensee to credit ongoing active maintenance and monitoring at the disposal site after the institutional control period, the layers of defense for these two time periods are expected to be barriers and inventory limits after 100 years following closure.

Licensees can use results from sensitivity analyses to support the identification of the defense-in-depth protections. For instance, Section 6.3.2 describes how to conduct barrier analyses for demonstrating compliance with the 10 CFR 61.41 performance objective. Barrier analyses are commonly performed as part of a performance assessment, and though the guidance in Section 6.3.2 is focused on the performance period, it applies to barrier analyses for the compliance period as well. By examining the consequences of early degradation or failure

of one or more of the layers of defense, licensees can demonstrate that safety can be maintained or determine when additional layers of defense may be needed.

To support the description of the capabilities of the layers of defense, licensees should examine the reasonably foreseeable scenarios considered for demonstrating that the performance objectives will be met, and the less likely, but plausible, scenarios (e.g., accidents, disruptive events, abnormal occurrences). If the performance objectives are met, significant consequences are not expected for reasonably foreseeable scenarios. However, significant consequences may be possible for less likely, but plausible, scenarios.

Licensees should demonstrate that the defense-in-depth protections will mitigate uncertainties such that there is confidence that the limit specified in 10 CFR 61.41(a) will be met. Licensees could show that probabilistic dose curves derived from 10 CFR 61.13 analyses are below designated limits. Licensees must demonstrate that the specified limit is not exceeded for the mean doses during the compliance period to demonstrate that the performance objectives are met. If the peak of the mean dose curve exceeds the specified limit, then additional layers of defense or, where possible, added redundancy are necessary. Chapter 3.0 contains guidance on demonstrating that the performance objectives are met for protection of the general population. Chapter 5.0 presents guidance on demonstrating the long-term stability of the disposal site.

Once licensees demonstrate that the performance objectives are met, they should show that significant uncertainties have been mitigated by defense-in-depth protections. Licensees should demonstrate that the 95th percentile of annual doses from probabilistic analyses of the central scenario is less than the dose limits for members of the public (i.e., 1 mSv (100 mrem)), as described in Section 3.2.4.3 of NUREG-1573, and less than 20 mSv (2 rem) for protection of inadvertent intruders, an exposure level for human intruders above which alternative options for waste disposal are to be considered (IAEA, 2011).

Additionally, if the peak annual dose from any single or subset of realizations of the central or alternative site scenarios for 10 CFR 61.13 analyses exceeds 50 mSv (5 rem) for either protection of the general population or inadvertent intruders, licensees should examine those realizations (or alternative site scenarios) to determine whether additional layers of defense, independence, or, where possible, redundancy would be beneficial to mitigate uncertainties. The 50 mSv (5 rem) guidance threshold provides confidence that any plausible projected central or alternative scenario would not be expected to exceed a level that would almost always justify intervention in an emergency. The ICRP advises that intervention would almost always be justified for existing exposures that would exceed 100 mSv (10 rem) (ICRP, 2007). Additional layers of defense, independence, or, where possible, redundancy should ensure that doses for plausible scenarios are kept well below levels where future intervention to mitigate significant exposures would be necessary or that the likelihood of the realization would be reduced by the additional layers of defense to render it implausible.

If a licensee performs deterministic rather than probabilistic 10 CFR 61.13 analyses, it should demonstrate that annual doses from reasonably foreseeable or less likely, but plausible, scenarios are less than 1 mSv (100 mrem) for protection of the public and less than 20 mSv (2 rem) for protection of inadvertent intruders to ensure that doses would be maintained well below levels where future intervention to mitigate significant exposures would be necessary.

Disposal facilities containing waste that could result in potential doses from reasonably foreseeable or less likely, but plausible, scenarios that exceed those mentioned in the previous paragraphs during the compliance period are considered higher risk facilities. Licensees should use additional layers of protection or confirm that sufficient independence and, where possible, redundancy will be provided to ensure that common-cause failures are minimized and adequate redundancy in the safety function is provided so that a significant exposure is highly unlikely to occur.

In deciding whether additional layers of defense would be needed, licensees may consider technological or economic limitations to employing additional barriers. However, the additional cost of limiting inventory is typically not expected to be overly burdensome from either a technical or economic consideration and is expected to provide the greatest certainty that the performance objectives can be met. Therefore, licensees and reviewers should strongly consider additional inventory limits as added controls or alternative siting to ensure that significant uncertainties can be mitigated by the disposal system.

8.0 WASTE ACCEPTANCE AND WASTE CLASSIFICATION

In 10 CFR 61.23, "Standards for issuance of a license," the regulations specify standards that must be met to receive a license to operate a land disposal facility for LLW. The standards require licensees to demonstrate that the waste acceptance criteria and other components of their proposals provide reasonable assurance that the performance objectives specified in 10 CFR Part 61, Subpart C, will be met. The regulations at 10 CFR 61.58, which specify the requirements for waste acceptance, require licensees to identify (i) criteria for the acceptance of waste for disposal, (ii) acceptable methods for characterizing the waste, and (iii) a program to certify that waste meets the acceptance criteria prior to receipt at a land disposal facility. Figure 8-1 illustrates the main components of waste acceptance under 10 CFR Part 61. The regulations require licensees to review the content and implementation of the waste acceptance criteria, characterization methods, and certification program at least annually.

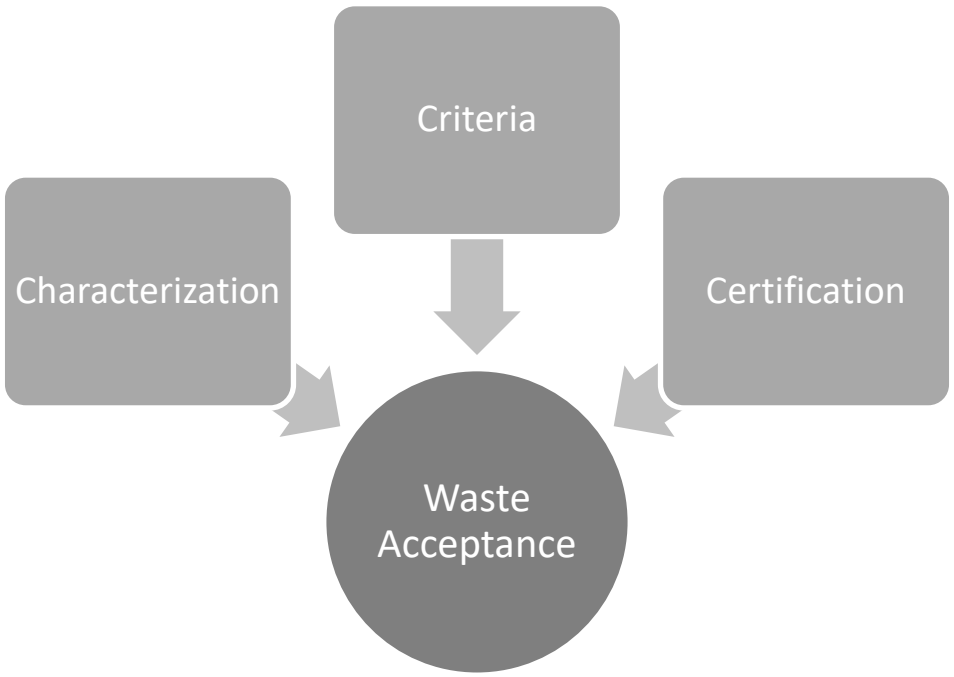


Figure 8-1 Waste Acceptance Components

This section describes the information that a licensee should provide, and a reviewer should evaluate, with respect to the waste acceptance requirements. First, a licensee should develop waste acceptance criteria to identify the following for waste generators: (1) the allowable limits on radionuclides, (2) acceptable wasteforms and container specifications, and (3) restrictions or prohibitions on waste to be accepted for disposal. Section 8.1 describes the process of waste classification by generators so that the waste may be shipped to the disposal facility. Second, Section 8.2 describes the information that licensees should provide to demonstrate that the waste acceptance criteria will be adequate. Third, waste must be adequately characterized to demonstrate that it meets the acceptance criteria. Section 8.3 provides guidance to licensees on acceptable methods for waste characterization. Finally, waste must be certified to ensure

that waste meets the acceptance criteria and is, therefore, suitable for disposal. Section 8.4 describes information that licensees should include in a certification program to ensure that waste received at the facility is acceptable for disposal.

The regulations in 10 CFR 61.58 specify the following:

> (a) *Waste acceptance criteria*. Each licensee must implement waste acceptance criteria approved by the Director, that provide reasonable assurance of compliance with the performance objectives of subpart C of this part. Waste acceptance criteria may be either generic or site-specific.
>
> > (1) *Generic waste acceptance criteria*. Licensees may implement the generic waste acceptance criteria set forth in § 61.55 and § 61.56.
> >
> > (2) *Site-specific waste acceptance criteria*. Licensees may implement site-specific waste acceptance criteria that have been approved by the Director. Site-specific waste acceptance criteria must provide safety equivalent to the requirements in § 61.56. Waste with radionuclide concentrations in excess of the Class C limits codified at 42 U.S.C. § 2021c(a)(1)(A) on a sum-of-fractions basis is Greater-Than-Class C waste and site-specific limits established under this part cannot change that designation. Site-specific waste acceptance criteria must specify the following:
> >
> > > (i) Allowable activities and concentrations of specific radionuclides. The allowable activities and concentrations for each specific radionuclide that the licensee intends to accept for disposal, developed from the technical analyses.
> > >
> > > (ii) Acceptable wasteform characteristics and waste container specifications. The wasteform characteristics of the waste to be accepted for disposal and the specifications for all waste containers that will be deployed during operations at the facility.
> > >
> > > (iii) Restrictions or prohibitions on waste, materials, or containers that might affect the facility's ability to meet the performance objectives in subpart C of this part.

Under the waste acceptance requirements of 10 CFR Part 61, licensees may develop either generic waste acceptance criteria based on the waste classification limits in 10 CFR 61.55 (e.g., in Tables 1 through 3) and waste characteristics in 10 CFR 61.56, or site-specific waste acceptance criteria developed from the technical analyses required in 10 CFR 61.13. Site-specific waste acceptance criteria could also be developed from a combination of technical analyses results and the waste classification limits.

8.1 Waste Classification

A licensee's waste classification considers both the short- and the long-lived radionuclides that may be present in the waste. Waste classes and the corresponding technical requirements jointly provide reasonable assurance of safety from waste generation, through waste

management, transportation, and disposal. The NRC developed different classes of waste (A, B, C, and GTCC) with corresponding requirements for each class.

The classes were derived from a generic regulatory analysis when 10 CFR Part 61 was developed (NRC, 1982a; NRC,1982b) that made various assumptions about the waste and how it was likely to be disposed. The waste classification limits given in 10 CFR 61.55 were based on a defined set of potential intruder scenarios and exposure pathways whose relevance to a specific disposal facility depends on the design of the facility. Class A waste is the least hazardous, and GTCC waste is the most hazardous. Wastes may be stable or unstable, but only certain types of Class A waste may be disposed of without stabilization. Some types of unstable Class A waste (e.g., biodegradable waste), depending on the radiological composition of the constituent radionuclides, require stabilization before they can be safely disposed of in the near surface. Although some unstable Class A wastes may safely be disposed of without stabilization, licensees must ensure (e.g., through segregation of such wastes) that interactions between unstable Class A wastes and other wastes will not result in a failure to meet performance objectives. Stable Class A wastes may be disposed of with other classes of waste. Class B, Class C, and GTCC waste must be stable for proper disposal.

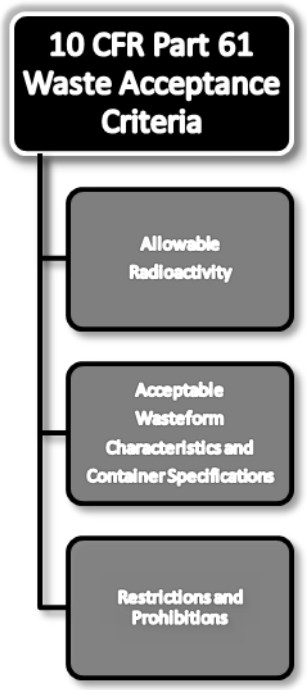


Figure 8-2 Components of 10 CFR Part 61 Waste Acceptance Criteria

Figure 8-2 shows the three components of the waste acceptance criteria. “Allowable radioactivity” refers to the allowable activities and concentrations of radionuclides. Under the new waste acceptance requirements of 10 CFR Part 61, licensees may develop either generic allowable limits for radioactivity based on the waste classification limits in 10 CFR 61.55 (e.g., in Tables 1 through 3) or site-specific allowable limits for radioactivity for disposal based on the technical analyses required in 10 CFR 61.13 (or a combination of technical analyses results and the waste classification limits). For instance, licensees disposing of waste that is similar to the waste streams considered in the development of the waste classification limits may wish to rely on those limits, and would, therefore, adopt generic waste acceptance criteria. Licensees disposing of waste streams that differ substantially, in quantity or in the radionuclides present, from those considered in the development of the waste classification limits in 10 CFR 61.55 may develop limits from the results of their technical analyses. Similarly, licensees with facility designs, operational practices, or site characteristics that differ significantly from those considered to develop the waste classification limits may want to develop site-specific waste acceptance criteria. Site-specific waste acceptance criteria may be advantageous or necessary to alleviate the potential burden on licensees from technical requirements that are not necessary for certain wastes to be disposed in a facility or to provide additional restrictions for wastes with differing characteristics.

According to 10 CFR 61.57, each package of waste must be clearly labeled to indicate the waste class in accordance with 10 CFR 61.55 when being shipped for disposal in a land disposal facility. Each package of waste must also be clearly labeled to identify any additional information required by the facility's waste acceptance criteria, as developed in accordance with 10 CFR 61.58.

Appendix G to 10 CFR Part 20 provides requirements for transfers of LLW intended for disposal at licensed land disposal facilities and for waste manifests. Instructions for completing the NRC's UWM appear in NUREG/BR-0204, Revision 3 (NRC, 2020a). A waste generator, collector, or processor who transports, or offers for transportation, LLW intended for ultimate disposal at a licensed LLW land disposal facility must prepare a manifest reflecting information requested on NRC Forms 540, 541, and542, as applicable. The licensee does not need to use these NRC Forms themselves as long as the manifest reflects the information requested on the applicable NRC Forms.. Box 17 of Form 541 identifies four designations to describe the waste class: "AS – Class A Stable," "AU – Class A Unstable," "B – Class B," and "C – Class C." As discussed in NUREG/BR-0204, Revision 3, waste generators shipping GTCC waste should use "GTCC" in Box 17.

Waste being prepared for shipment must be classified according to 10 CFR 61.55, even if the disposal facility operator intends to accept the waste for disposal based on their site-specific waste acceptance criteria derived from their 10 CFR Part 61.13 technical analyses. Requiring the waste manifest form to continue to reflect the waste classification according to 10 CFR 61.55 will enable continuity with existing operations and ensure that prohibitions associated with waste class continue to be identified and implemented. Additional information on the site-specific waste acceptance criteria and waste characteristics should be included in pages attached to the Form 541. The operator of a disposal facility may request that generators include additional description of this waste, as needed for waste acceptance and disposal.

The SOF approach to determining the classification of mixtures of radionuclides applies to either the classification of waste under 10 CFR 61.55 or based on the results of the site-specific technical analyses. If using site-specific technical analyses to establish allowable concentrations and quantities of radionuclides, the licensee should use the SOF approach to determine acceptable mixtures of radionuclides, based on derived concentrations that demonstrate compliance with each performance objective. A licensee may either combine the most limiting allowable limits derived from analyses of different performance objectives, or use allowable limits derived for each performance objective (see Example 8-1).

Example 8-1: A disposal facility operator develops a performance assessment and intruder assessment. The operator calculates limiting concentrations of radionuclides. There are no credible operational accident scenarios that result in limiting concentrations. For the performance assessment and intruder assessment, the allowable limits of Cs-137, Sr-90, Pu-238, I-129, and Tc-99 are shown below. A generator wishes to dispose of waste containing the concentrations shown below. Should the operator accept the waste?

	Performance Assessment Limiting Concentrations (Ci/m^3)	Intruder Assessment Limiting Concentrations (Ci/m^3)	Proposed Waste Concentrations (Ci/m^3)
Cs-137	2000	100	50
Sr-90	40	10	2
Pu-238	1500	150	15
I-129	2	40	0.1
Tc-99	5	50	3

Result: The waste contains more than 1 times the value provided in Table 1 of 10 CFR 61.55 for I-129, so the waste is GTCC. The State does not have a prohibition on the disposal of GTCC waste. The licensee examines the most limiting allowable limit for each radionuclide:

$$SOF = \frac{50}{100} + \frac{2}{10} + \frac{15}{150} + \frac{0.1}{2} + \frac{3}{5} = 1.45$$

Because the SOF is greater than 1 the licensee calculates the SOF for each performance objective:

$$SOF_{61.42} = \frac{50}{100} + \frac{2}{10} + \frac{15}{150} + \frac{0.1}{40} + \frac{3}{50} = 0.8625$$

$$SOF_{61.41} = \frac{50}{2{,}000} + \frac{2}{40} + \frac{15}{1{,}500} + \frac{0.1}{2} + \frac{3}{5} = 0.735$$

Because each SOF is less than 1, the waste can be accepted for disposal if all other site-specific waste characteristics and stability requirements are met.

8.2 Waste Acceptance Criteria

The regulations in 10 CFR 61.52, “Land disposal facility operation and disposal site closure,” require that waste to be disposed of must meet approved waste acceptance criteria. Under 10 CFR 61.58, licensees must submit proposed waste acceptance criteria for approval by the regulator. Once the regulator has approved the waste acceptance criteria, under 10 CFR 61.58, licensees wishing to modify the criteria must request an amendment, because the waste acceptance criteria will be incorporated in the license. This section provides guidance on developing and modifying waste acceptance criteria, and it discusses information that licensees should include to allow the regulator to evaluate whether the proposed waste acceptance criteria provide reasonable assurance that the performance objectives will be met.

The regulations in 10 CFR 61.58 specify the requirements for waste acceptance at a disposal facility for LLW. Waste acceptance criteria are intended to ensure that the waste received for disposal is consistent with the technical analyses used to demonstrate reasonable assurance that the performance objectives of 10 CFR Part 61 will be met. The regulations require that the waste acceptance criteria indicate allowable activities and concentrations of specific radionuclides, acceptable wasteform characteristics and container specifications, and restrictions or prohibitions on waste, materials, or containers. Waste acceptance criteria may be generic, based on the limits in 10 CFR 61.55 and the waste characteristics in 10 CFR 61.56, or they may be site-specific, based on the licensee's technical analyses or on a combination of the technical analyses and 10 CFR 61.55 and 10 CFR 61.56. Figure 8-2 depicts the minimum components of the waste acceptance criteria.

Licensees may need to specify other criteria beyond those required by 10 CFR Part 61, to satisfy other regulatory requirements. For instance, land disposal facilities may also be subject to other Federal or State regulations that limit certain non-radiological materials because of their potential impact on public health and safety and the environment. This guidance document addresses only the waste acceptance criteria required by 10 CFR Part 61.

8.2.1 Allowable Activities and Concentrations

Under the waste acceptance requirements of 10 CFR Part 61, licensees may develop either generic allowable limits for radioactivity based on the waste classification limits in 10 CFR 61.55 (i.e., those of Tables 1 through 3), or site-specific limits based on the technical analyses required in 10 CFR 61.13. Site-specific allowable limits may also be based on a combination of results from the technical analyses and the 10 CFR 61.55 waste classification limits. For instance, licensees disposing of waste that is similar to the waste streams considered in the development of the waste classification limits may wish to rely on those limits and adopt generic waste acceptance criteria. Licensees disposing of waste streams that differ substantially, in quantity or in the radionuclides present, from those considered in the development of the waste classification limits may wish to develop site-specific limits from the results of their technical analyses. Similarly, licensees with facility designs, operational practices, or site characteristics that differ significantly from those considered to develop the waste classification limits may want to develop site-specific waste acceptance criteria.

The characteristics of the waste, the disposal site, and the design are the primary determinants of risk from the land disposal of radioactive waste. Not all radioactive waste streams are suitable for near-surface disposal. Regardless of the method selected to develop allowable activities and concentrations, licensees must demonstrate through technical analyses (i.e., under 10 CFR 61.13) that the performance objectives of 10 CFR Part 61, Subpart C, will be met.

This section describes the information that a licensee should provide and that a reviewer should evaluate with respect to the development of allowable activities and concentrations of radionuclides for acceptance for disposal. Reviewers may want to consult Section 6.1.1 of NUREG-1200 to verify that the allowable limits specified by the licensee's waste acceptance criteria are reasonable, given the types and quantities of radionuclides projected for disposal.

8.2.1.1 Site-Specific Allowable Limits Derived from Technical Analyses

Radioactivity from waste disposed in a land disposal facility may need to be limited to ensure (1) protection of the general public from releases, both during operations and after operations have ceased, (2) protection of individuals who may inadvertently intrude into the disposal site after active institutional controls have been removed, (3) protection of individuals during operations, and (4) stability of the disposal site after closure. Limits on radioactivity can also provide defense-in-depth to enhance the resiliency of the facility in complying with the performance objectives. The appropriate limits may vary from site to site depending on the waste streams proposed for disposal, facility design, and site characteristics.

Licensees that develop site-specific limits on radionuclide activities or concentrations from the results of the analyses required in 10 CFR 61.13 should document how they developed the proposed limits. Figure 8-3 depicts a general process for developing allowable limits from the results of the analyses. The licensee should focus on radionuclides that may affect the facility's ability to meet the 10 CFR Part 61 performance objectives. The limits may be for a specific waste stream, or they may be total limits for the disposal site that are translated to per-package or per-shipment limits. Limits for specific waste streams may be more appropriate when unique factors may affect the facility's ability to meet one or more performance objectives. These factors may include, but are not limited to, release from the wasteform, the concentrations of radionuclides in the waste, the potential for criticality due to the presence of special nuclear material in the waste, the radiation emitted from the waste, or the heat generated by the decaying waste. Many of these factors are associated with GTCC waste (see Appendix G).

Licensees should develop allowable limits, for either total activity or concentration, from the peak dose resulting from a unit activity of each radionuclide for the performance objective that is the most limiting for that radionuclide. To establish the limits, the licensee should select the scenario(s) to be used to demonstrate that the performance objectives are met, as described in Chapters 2.0, 3.0, and 4.0. Licensees should compare the peak dose for each radionuclide with the limits specified in the performance objective. Licensees may calculate a limit for each radionuclide using the following formula:

$$Limit_{i,s} = \frac{Dose\ Limit_s \times Activity(0)_i}{Peak\ Dose_{i,s}} \qquad (8.1)$$

Where:

$Limit_{i,s}$ total activity (becquerels (Bq) or activity concentration (Bq/m^3 or Bq/g) limit of radionuclide *i* in waste for scenario[1] *s*

$Dose\ Limit_s$ dose limit (mSv/yr) for scenario *s*

$Activity(0)_i$ initial total activity (Bq) or activity concentration (Bq/m^3 or Bq/g) of radionuclide *i* in the waste

$Peak\ Dose_{i,s}$ annual peak dose (mSv/yr) resulting from the activity $Activity(0)_i$ of radionuclide *i* for scenario *s*

[1] "Scenario" here refers to the central scenario for the performance objective under consideration.

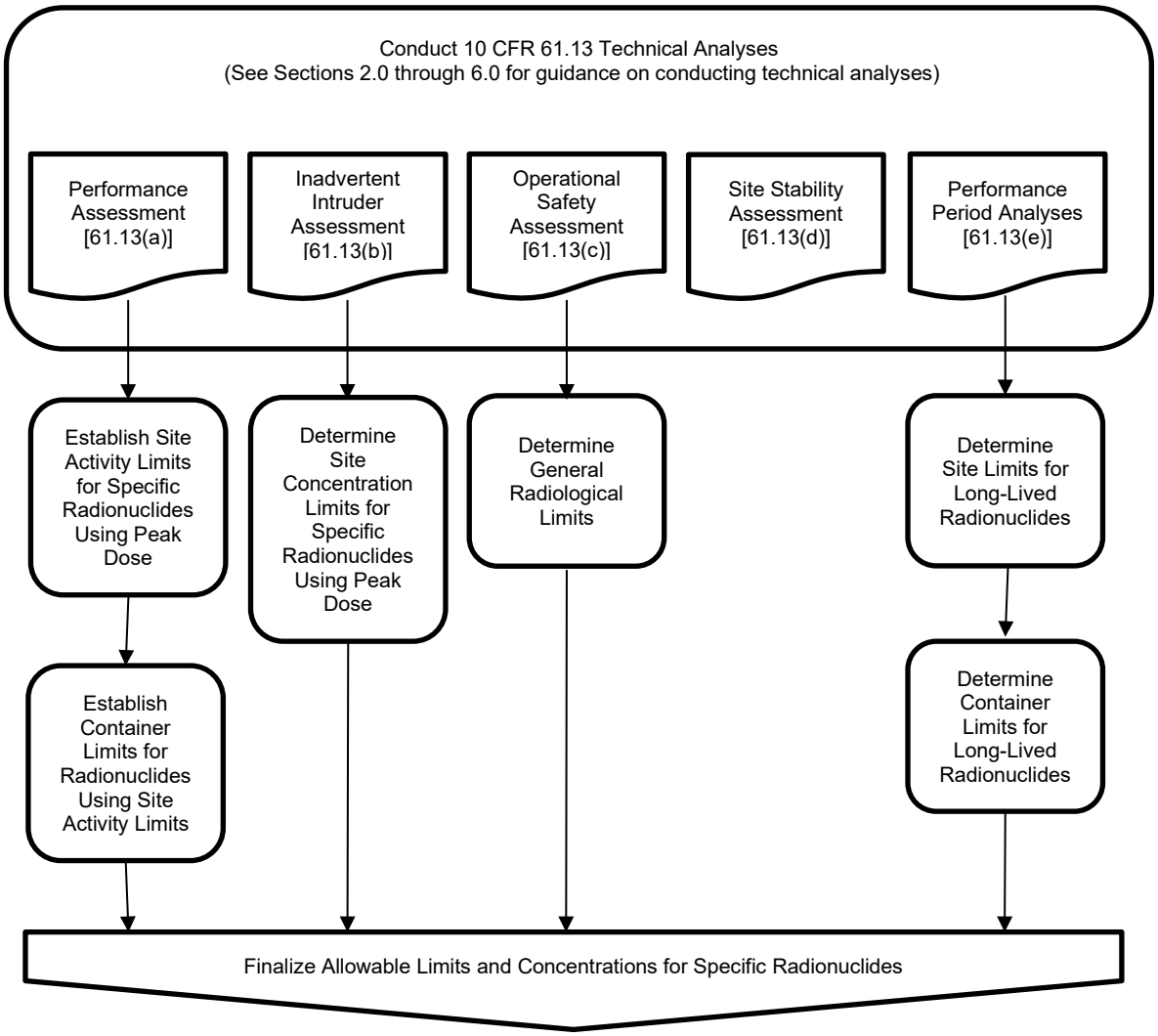


Figure 8-3 Example Process for Developing Site-Specific Allowable Limits Derived from Technical Analyses

For radionuclides that have disposal site activity and concentration limits, licensees should determine limits for individual containers, so that a waste shipment to the disposal facility can be certified that the allowable limits are met, and the waste is acceptable for disposal. In developing limits for individual containers, licensees should ensure that the assumptions and methods used are consistent with the conceptual models in the corresponding technical analyses. For example, licensees may be able to develop higher container concentration limits than the average site concentration limit determined by the inadvertent intruder assessment, because of mixing with non-contaminated materials such as backfill. However, licensees should demonstrate that facility design, procedures, and controls would provide confidence that the concentration would not be likely to exceed the average site concentration over a volume equivalent to the volume evaluated in the intruder assessment.

Once licensees have developed radionuclide-specific limits based on the results of the various technical analyses, they should finalize the allowable limits by selecting the one that is most limiting for each radionuclide. For example, if the licensee develops a container limit that demonstrates that the general population will be protected, but also develops a container limit to demonstrate protection of the intruder that is more limiting (i.e., lower), the licensee should select the lower value.

Licensees should identify the technical basis for allowable limits specified in the site-specific waste acceptance criteria. The basis for each limit is typically derived from the results of the technical analyses. However, there may be other bases for some limits, such as other regulatory requirements (e.g., limits on fissile materials). The basis for proposed limits should identify the performance objective(s) or other regulatory requirements, if applicable, from which the limits are derived. For instance, proposed limits for some radionuclides may be more relevant to protection of facility workers during operations than to protection of the public from releases. The licensee should describe why each proposed limit is sufficient to demonstrate that the performance objectives, or other applicable regulatory requirements, will be met. Licensees will need to demonstrate that the individual limits, when taken together, will ensure that the performance objectives continue to be met.

Licensees should describe how they accounted for variability and uncertainty in anticipated inventories when developing the allowable limits. One method to mitigate uncertainty is to manage it through controls or restrictions. A key control to manage variability and uncertainty is to impose limits on the radionuclide inventory of the waste accepted for disposal. The inventory of radionuclides in the waste can be readily controlled, whereas the uncertainty associated with a natural system or engineered barriers, for example, may be difficult to define or, if understood, difficult to reduce. The following general guidelines are useful for the development of site-specific allowable limits under 10 CFR 61.58:

- Allowable limits should be established conservatively to reduce the need for future mitigation.

- The analysis used to develop allowable limits should include uncertainties.

The allowable limits a licensee derives will be affected by the complexity of the disposal site and the environment, the conservatism in the analyses, and the amount of information available to support the assessment. Figure 8-4 illustrates how these factors affect the relative confidence that the allowable limits derived from the technical analyses will be protective.

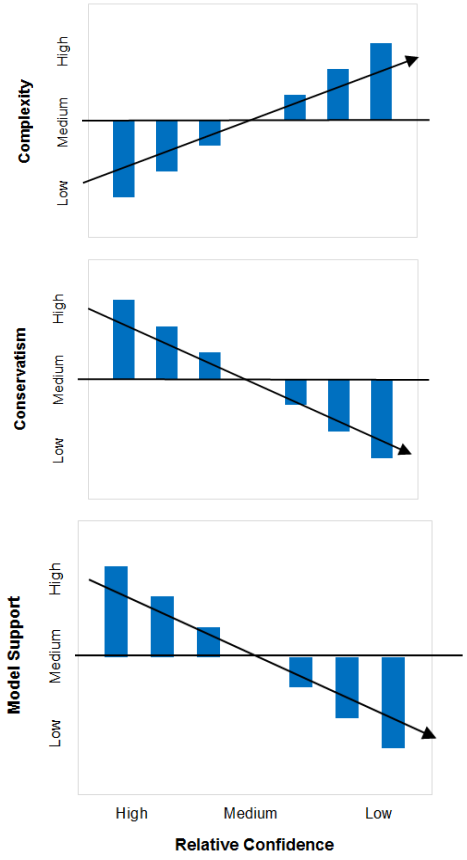


Figure 8-4 Influence of Key Variables on the Relative Confidence

When detailed, comprehensive information is available to support the analyses, licensees may apply a small margin of safety to the results of the analyses to establish the derived allowable limits, because they will have high confidence. On the other hand, if the disposal site environment is complex, licensees may need to apply a large margin of safety to provide reasonable assurance the performance objectives will be met. The uncertainty in the calculated risk will also affect the derived allowable limits. There is no absolute value (e.g., an inventory that corresponds to 10 percent of the 0.25 mSv/yr or 25 mrem/yr dose limit) that is appropriate to use to establish limits. Allowable limits can be viewed as one of several safety factors or defense-in-depth protections used to mitigate uncertainty in other (i.e., non-inventory) areas of the technical analyses.

Reviewers should coordinate their reviews of the site-specific allowable limits with their reviews of the technical analyses required to meet 10 CFR 61.13 (see Chapters 2.0 through 6.0). Reviewers should verify that the licensee's proposed limits on activities and concentrations are adequate, considering the source term inventory used in the performance and intruder assessments as well as the operational safety assessments of expected and accidental occupational exposures during handling, storage, and disposal of waste (e.g., GTCC waste).

The proposed limits should be consistent with the source term evaluated in the technical analyses. However, licensees may use various approaches to develop site-specific allowable limits. For example, a licensee may derive a limit from the results of one or more of the

analyses for a source term with a unit concentration (e.g., 0.037 kBq/g (1.0 nCi/g)), and then scale it to the limit specified by the relevant performance objective. In this example, the proposed limits would not be expected to be consistent with the source term employed in the analyses but should be consistent with the outcome of the analyses that meets the performance objective. The reviewer should evaluate whether the approach is appropriate. For example, it may not be appropriate for a licensee to assume a source term with unit concentrations when the radiological exposures are a non-linear function of the source term concentrations.

Reviewers should assess the spatial extent and distribution of the radionuclide inventory evaluated in the analyses to determine whether the allowable proposed limits are appropriate (e.g., when considering the disposal facility's operating procedures for emplacement). The spatial extent or distribution of radionuclide inventory at the disposal site may need to be controlled as a license condition. For example, if a licensee develops a total average concentration limit for a radionuclide from the intruder assessment based on a certain expected volume to which the intruder may be inadvertently exposed, then reviewers should confirm that the facility's operating procedures would reasonably ensure that the proposed limit would not be exceeded for any volume of the disposal unit equivalent to the intrusion volume assumed by the licensee in the assessment.

A licensee will not typically derive allowable limits from the site stability analyses; however, reviewers should evaluate the licensee's information if stability is considered when establishing the allowable limits. Stable wasteforms or containers may enable the licensee to dispose of waste with higher radioactivity than would be allowable for unstable waste. Reviewers should evaluate the estimated performance of the wasteforms, containers, or other design features that provide stability. Reviewers should coordinate this review with their review of acceptable wasteforms and containers, which is described Section 8.2.2. Reviewers should confirm that licensees are establishing site-specific waste acceptance criteria for wasteforms, containers, or design features that provide assurance the expected performance can be achieved. In some cases, the radionuclide inventory could significantly affect the stability of the disposal site after closure. For example, reviewers may need to evaluate whether thermal output or radiation emitted from radioactive decay for certain higher activity wastes, such as GTCC, would significantly degrade wasteforms or compromise the structural stability of waste containers. Reviewers should coordinate their review of the development of allowable limits with their review of acceptable wasteform characteristics and container specifications (see Section 8.2.2).

Reviewers should focus the review of site-specific allowable limits on those radionuclides expected to contribute most significantly to risk to the public, workers, and the environment. Typically, radionuclides with relatively high solubility, low sorption, high dose conversion factors, or significant ingrowth are of greater significance. However, the importance of certain radionuclides may vary based upon the specific performance objective under consideration. For instance, non-mobile radionuclides may be more significant for protection of inadvertent intruders.

Reviewers should evaluate whether the licensee's technical bases for the proposed limits provide assurance that the performance objectives will be met. For example, if an allowable limit for a particular radionuclide is based on the results of the performance assessment, reviewers should confirm that the limit does not preclude licensees from demonstrating that the other performance objectives (e.g., protection of inadvertent intruders) will be met. Reviewers should verify that the site-specific allowable limits are comprehensive and that all necessary limits are included. Reviewers may elect to conduct independent analyses to inform the review.

8.2.1.2 Generic Allowable Limits Derived from Waste Classification

Licensees that develop generic allowable limits for radionuclides from the waste classification requirements of 10 CFR Part 61 may report the concentration limits provided in 10 CFR 61.55 and demonstrate that the results of their 10 CFR 61.13 technical analyses meet the respective performance objectives. The limits specified in 10 CFR 61.55 should be applied on a per- package basis. Licensees may need to develop additional limits for radionuclides not listed in the tables in 10 CFR 61.55, particularly for waste that is significantly different from the waste considered in the analyses used to develop the tables (e.g., waste that contains significant quantities of radionuclides not listed in the Tables, such as uranium) (see NRC, 1981a). Tables 4.1 and 4.2 of NUREG-0945, "Final Environmental Impact Statement on 10 CFR Part 61 'Licensing Requirements for Land Disposal of Radioactive Waste,'" Volume 1, issued November 1982 (NRC, 1982b), list the waste streams and radionuclides considered in the analyses used to develop the tables. This section provides guidance on developing limits for radionuclides not listed in the waste classification tables.

The concentration limits specified in 10 CFR 61.55, together with the requirements for waste characteristics (see 10 CFR 61.56) and segregation (see 10 CFR 61.52(a)(1)–(2)), provide reasonable assurance that an intruder would be protected should they inadvertently be exposed to the waste in a near-surface disposal facility. Additional measures are required to prevent inadvertent intrusion into a specialized disposal facility because of the more concentrated waste allowed in those facilities, as described further in Appendix G. Classification involves consideration of both (1) long-lived radionuclides, whose potential hazard will persist long after precautions such as institutional controls, improved wasteforms, and deeper disposal have ceased to be effective, and (2) shorter-lived radionuclides, for which such precautions can be effective.

For licensees using generic waste acceptance criteria, waste classification is used to determine which waste characteristics requirements in 10 CFR 61.56 are necessary. Additional waste characteristic requirements (e.g., related to heat generation and radiolysis) apply to GTCC waste compared to other wastes. Waste classification is used to determine the waste segregation requirements from 10 CFR 61.52 that the disposal facility must meet during operations. Waste segregation provides assurance that the disposal of unstable waste will not adversely impact the performance of stable waste. Figure 8-5 illustrates the waste classification and segregation requirements for LLW. Licensees relying on the waste classification system to develop generic allowable limits must ensure that the criteria for both acceptable wasteform characteristics and facility operating practices are consistent with the related requirements for waste characteristics and segregation. Section 8.2.2 provides further guidance on developing criteria for acceptable wasteform characteristics.

Licensees can demonstrate compliance with the classification requirements for selected radionuclides by comparing the radionuclide concentrations in LLW to the values listed in the tables of long- and short-lived radionuclides in 10 CFR 61.55 (i.e., Tables 1 through 3). The radionuclide concentrations in these tables were developed, in part, from two considerations: (1) direct contact with the disposed waste (i.e., intrusion) and (2) potential consumption or use of contaminated groundwater (i.e., migration). Direct contact was the primary consideration because it was amenable to evaluation on a generic basis.

As established in NUREG-0782, the approach first considered protection of the inadvertent intruder. The approach then identified a limited set of radionuclides that are significant from the

standpoint of migration. Radionuclides significant from the standpoint of migration are generally more important for protection of the general population; the NRC staff expected these radionuclides to be evaluated as part of the demonstration that the performance objective for protection of the general population (i.e., 10 CFR 61.41) is met. Appendix F summarizes the approaches and assumptions used to develop the waste classification limits in 10 CFR 61.55.

For near-surface disposal, 10 CFR 61.55 specifies the three classes of waste, A, B, and C, that resulted from the analysis. Upper concentration limits are also defined for Class C waste. Wastes containing radionuclide concentrations higher than the upper limits are defined as GTCC wastes. Some GTCC wastes are suitable for near-surface disposal. Technical analyses are used to determine which GTCC wastes are suitable for near-surface disposal and which may not be suitable. An upper limit on GTCC wastes suitable for near-surface disposal is established in 10 CFR 61.52(a)(4) of 370 kBq/g (10,000 nCi/g) for long-lived alpha-emitting transuranic radionuclides. However, there may be instances where additional wastes would be acceptable for near-surface disposal with special processing or design. These instances would be approved by the Commission on a case-by-case basis.

In addition, 10 CFR 61.52(a)(4) specifies that GTCC wastes exceeding 370 kBq/g (10,000 nCi/g) long-lived alpha-emitting transuranic radionuclides but not exceeding the concentrations in 10 CFR 61.55(b) Table 3 are generally suitable for disposal in a specialized land disposal facility licensed by the NRC. For instance, it may be suitable to dispose of disused sealed sources in narrow-diameter boreholes at depths of a few hundred meters. Appendix G provides additional guidance on GTCC waste disposal.

Class A wastes may be categorized as either stable or unstable. Unstable Class A wastes are wastes for which there are no stability requirements, but which must be disposed of in a segregated manner from other wastes. They are defined in terms of maximum allowable concentrations of certain isotopes and certain minimum requirements on wasteform that are necessary for disposal. The minimum requirements are specified at 10 CFR 61.56(a). Unstable Class A waste must be segregated from other waste as required in 10 CFR 61.52(a)(1), unless a licensee can demonstrate that the waste also meets the stability requirements in 10 CFR 61.56(b) (in which case it is categorized as stable Class A waste). The technical analyses required in 10 CFR 61.13 could demonstrate, based on site-specific conditions, that additional disposal practices or waste characteristics—such as those required for Class B or C wastes—are necessary to demonstrate that the performance objectives will be met.

Class B wastes are wastes that must meet more rigorous requirements on wasteform to ensure stability after disposal. These stable wastes are defined in terms of allowable concentrations of radionuclides and requirements for a stable wasteform. The requirements for waste characteristics and for stability are specified at 10 CFR 61.56(a) and 10 CFR 61.56(b), respectively. In addition, the licensee's technical analyses may determine that additional disposal practices or waste characteristics—such as those required for Class C wastes—are necessary to ensure that the performance objectives will be met.

Class C wastes are wastes that need to be placed in a stable form, disposed of in a segregated manner from unstable wasteforms, and disposed so that a barrier is provided against potential inadvertent intrusion after institutional controls are no longer assumed to be effective. These wastes are defined in terms of allowable concentrations of radionuclides and requirements for disposal by deeper burial or some other intruder barrier, as well as stability requirements.

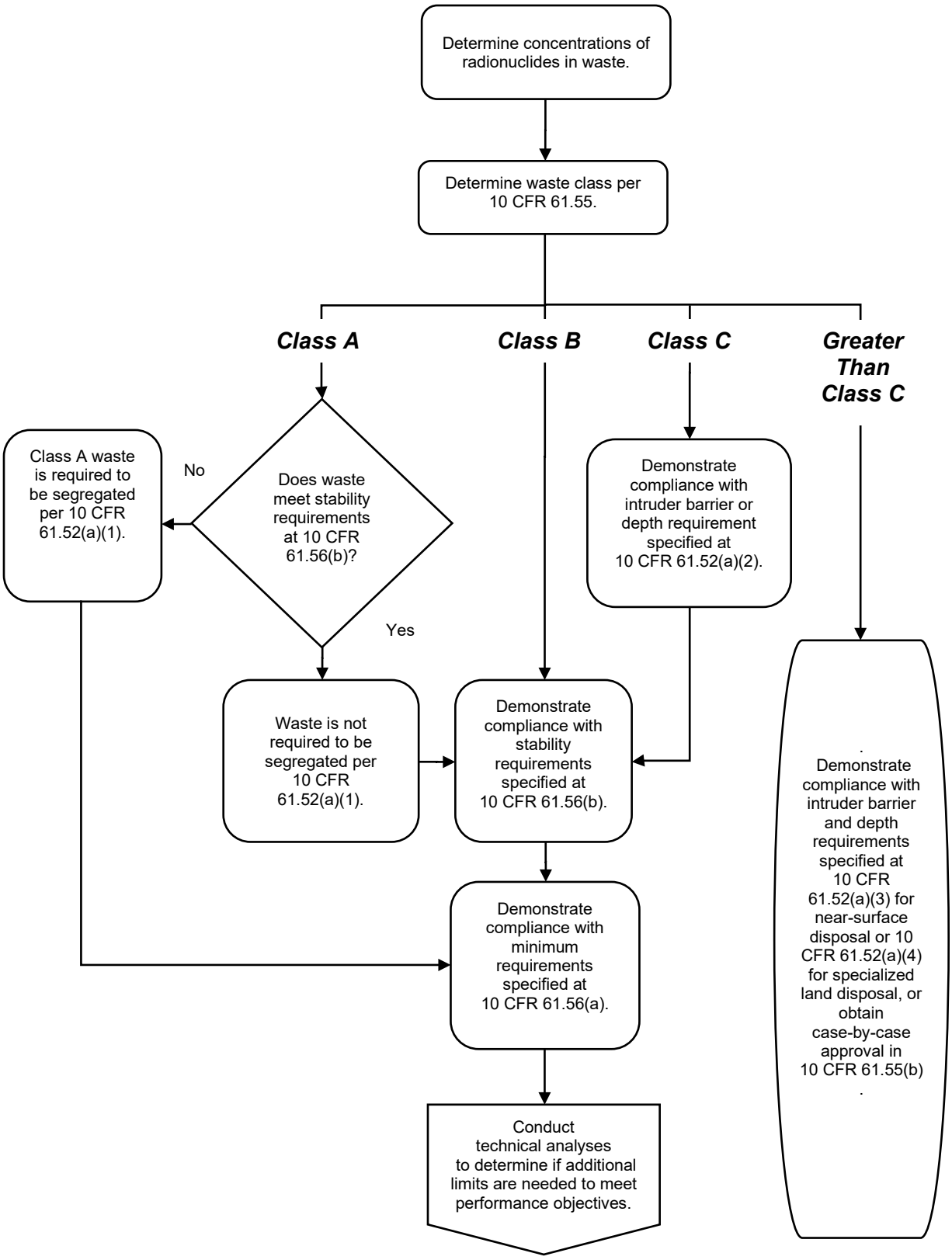


Figure 8-5 Waste Classification and Segregation for Waste Classes

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The requirements for waste characteristics and for stability are specified at 10 CFR 61.56(a) and 10 CFR 61.56(b), respectively. Requirements for deeper burial or protection with an intruder barrier are specified at 10 CFR 61.52(a)(2). The licensee's technical analyses may determine that additional disposal requirements are necessary to provide reasonable assurance that the performance objectives will be met.

GTCC wastes are wastes that need to be placed into a stable form, disposed of in a segregated manner from unstable wasteforms, and disposed of with an intruder barrier and a minimum burial depth of at least 5 m (16 ft) below the land surface. These wastes are defined by radionuclide concentrations that exceed the values developed for Class C wastes and do not exceed the values in 10 CFR 61.55(b), Table 3. The requirements for waste characteristics and for stability are specified at 10 CFR 61.56(a) and 10 CFR 61.56(b), respectively. Additional waste characteristics requirements for GTCC waste are specified at 10 CFR 61.56(c). Requirements for deeper burial and protection with an intruder barrier are specified at 10 CFR 61.52(a)(3), and an upper limit on near-surface disposal of GTCC wastes with greater than 370 kBq/g (10,000 nCi/g) of long-lived alpha-emitting transuranic radionuclides is specified in 10 CFR 61.52(a)(4). The licensee's technical analyses may show that additional disposal requirements are necessary to provide reasonable assurance that the performance objectives will be met. The near-surface disposal of GTCC wastes above the 370 kBq/g (10,000 nCi/g) limit is subject to case-by-case evaluation in accordance with 10 CFR 61.55(b)(2)(iii) and may be approved by the NRC depending on the specifics of the proposal. In addition, such waste may be acceptable for disposal in a specialized land disposal facility in accordance with 10 CFR 61.55(b)(2)(iii) if it does not exceed any of the values in 10 CFR 61.55(b) Table 3.

Because the classification limits were developed from an analysis of inadvertent intrusion,[2] they are not intended to provide reasonable assurance that all the performance objectives of 10 CFR Part 61 will be met. Licensees using the waste classification requirements for their generic waste acceptance criteria may need to develop additional limits based on the results of the technical analyses used to demonstrate that the remaining performance objectives will be met. For example, radionuclides prone to migration may require additional limits (e.g., limits on total inventory) to ensure that the 10 CFR 61.41 performance objective will be met.

As specified in 10 CFR 61.55(a)(6), waste that does not contain any of the radionuclides listed in Tables 1 and 2 is considered Class A. This waste is subject to the same requirements as Class A waste that contains radionuclides listed in either Table 1 or Table 2. However, for such waste, the licensee's technical analyses may show that additional limitations or restrictions, such as those required for Class B or C waste, are necessary to provide reasonable assurance that public health and safety will be protected. Therefore, licensees may need to specify limits for radionuclides not specifically identified in the waste classification requirements in 10 CFR 61.55 if these radionuclides significantly affect the demonstration that the performance objectives will be met. In these cases, licensees should refer to Section 8.2.1.1 for guidance on developing allowable limits from technical analyses.

The BTP "Final Waste Classification and Waste Form Technical Position Papers" dated May 11, 1983 (NRC, 1983d), provides guidance on classifying waste according to the waste classification requirements. The BTP CA (NRC, 2015a) provides guidance on acceptable methods of concentration averaging and encapsulation. Section 4.1 of NUREG-1200 provides

[2] The analysis also considered different accident scenarios such as fires, but the results of those scenarios were not carried forward when establishing the concentration limits for radionuclides.

guidance to assist reviewers determine whether licensees have adequate procedures to ensure that waste disposal is conducted in compliance with 10 CFR 61.55 and 10 CFR 61.56.

8.2.1.3 Insignificant Radionuclides

Licensees may choose not to develop limits for radionuclides that contribute, in aggregate, no greater than 10 percent of the projected dose limits prescribed in the performance objectives. That is, the sum of the contributions from all radionuclides considered insignificant should be no more than 10 percent of the limit for a performance objective. Radionuclides that could be excluded based on comparison to the limit for one performance objective (e.g., protection of the public from releases) may need to have limits to comply with another performance objective (e.g., protection of inadvertent intruders). Once a licensee has demonstrated that certain radionuclides are insignificant, the dose from the insignificant radionuclides may be excluded from the demonstration that the performance objective is met. This guidance is assuming that a licensee is very unlikely to be within 10 percent of any performance objective, because a licensee must account for uncertainties.

When licensees determine that certain radionuclides are insignificant and eliminated from further consideration, they should justify their decision to consider them insignificant. Licensees should also be aware that they may need to revisit these decisions if new information warrants it. For instance, if a disposal facility proposes to accept new waste streams with significantly different radionuclide inventories than previously considered, the licensee may need to assess whether radionuclides that were previously considered insignificant continue to remain so. If a licensee obtains updated information suggesting that the behavior of a radionuclide in the site environment has significantly changed, the licensee should reassess the significance of the radionuclide to meeting the performance objectives.

8.2.1.4 Application of Allowable Limits

Licensees will need to develop the allowable limits on a radionuclide-by-radionuclide basis. Because the performance objectives are based on the total contribution of all radionuclides to the dose limits, licensees will need to perform an SOF calculation to demonstrate that the performance objectives will be met. SOF calculations are also performed for waste classification under 10 CFR 61.55(a)(7). The SOF is the sum, over all radionuclides present in the waste, of the ratio of each radionuclide's activity to its allowable limit. To meet the performance objectives, the SOF must be no greater than 1. Licensees can use the following formula to sum the ratio of the activity to its corresponding allowable limit:

$$\sum_{i=1}^{N} \frac{Activity_i}{Limit_i} \leq 1 \qquad (8.2)$$

Where:

$Activity_i$ the total activity (Bq) or activity concentration (Bq/m^3 (Ci/m^3) or Bq/g (Ci/g)) of radionuclide *i*

$Limit_i$ the allowable limit (Bq (Ci), Bq/m^3 (Ci/m^3), or Bq/g (Ci/g)) of radionuclide *i* from the most restrictive scenario used to demonstrate that the performance objectives will be met

Licensees should develop a mechanism to report the inventory that has been disposed of and to identify when action may be needed. The following are among the types of actions that could be needed:

- Notify the NRC staff that a certain percentage of the allowable limit has been received.
- Evaluate whether updates to the technical analyses are needed.
- Update the technical analyses (establish new allowable limits).
- Modify the disposal facility design, including acceptable wasteforms (see Section 8.2.2).
- Limit or prohibit further disposal of a certain type of waste (see Section 8.2.3).

When significant changes to the operations of the disposal facility occur, licensees should evaluate whether the allowable limits need to be updated. Some operational changes may trigger an update to allowable limits whereas others may not. The former include, but are not limited to, a proposal to receive new material, receipt of significant new information on the site characteristics or engineering design, changes to intruder barriers, changes to the understanding of the performance of key components of the disposal system and updating of the technical analyses. Licensees should establish criteria for updating and revising allowable limits. If a licensee uses a process to determine the significance of a potential change to the established allowable limits, it should maintain a comprehensive list of all items that were determined to not sufficiently impact the allowable limits. The cumulative impact from many small changes can be additive, in such a way that, in total, the changes would require revision of the allowable limits to avoid the need for future mitigation.

If a licensee revises allowable limits, it should provide to regulators a basis for the revisions that includes a comparison of the changes to parameters or models and the basis for those changes. The comparison will provide transparency and facilitate the regulator's review of the revised allowable limits. It may also increase stakeholder confidence in the revisions. If limits are relied upon for defense-in-depth, the licensee should provide a basis that defense-in-depth protections remain adequate.

8.2.2 Acceptable Wasteform Characteristics and Container Specifications

Under 10 CFR 61.58(a)(2)(ii), licensees must specify acceptable wasteform characteristics and container specifications as part of the site-specific waste acceptance criteria. This information, together with the other waste acceptance criteria (e.g., radionuclide limits), helps provide reasonable assurance that the performance objectives will be met. Acceptable wasteform characteristics and container specifications include properties that facilitate handling of the waste at the disposal facility, promote stability of the waste to minimize subsidence and water contact with the waste, minimize release and migration of radionuclides from the disposal site, deter or preclude inadvertent intrusion into the waste, or limit exposures during operations.

All waste must meet the minimum requirements specified in 10 CFR 61.56(a), which are designed to facilitate handling of waste and protect the health and safety of personnel at the disposal facility. Licensees may also identify additional criteria to facilitate handling and protect facility personnel, depending on the operational practices and environmental conditions at the disposal facility. Additional criteria might include acceptable limits for the external surface dose rate and heat generation of a waste package, labeling and marking to be applied to waste packages, container specifications, or specific requirements for acceptance of bulk waste.

Licensees should provide a basis for including the additional criteria, explaining why they were developed and what performance objectives they support (as applicable).

For certain waste streams, licensees will need to meet more rigorous requirements on wasteform than the requirements to ensure stability after disposal and demonstrate compliance with the performance objectives. The requirements to ensure stability are specified in 10 CFR 61.56(b). For licensees relying on the waste classification requirements to develop generic waste acceptance criteria, Class B, Class C, and GTCC waste must meet the requirements to ensure stability. Licensees developing site-specific waste acceptance criteria from the results of the technical analyses will need to identify the wastes that will need to meet stability requirements. The site-specific approach allows for flexibility in determining what stability requirements (e.g., segregation) are necessary for the facility to meet the performance objectives.

Licensees may demonstrate that the stability requirements are met by using one or more of the following approaches:

- a stable wasteform
- a container that provides stability
- facility design

The approach that a licensee uses to demonstrate stability will determine the waste acceptance criteria needed to ensure stability. For example, if a licensee constructs a reinforced concrete vault and demonstrates that it provides adequate stability for the required timeframe, the waste acceptance criteria may not need to address stable wasteforms or container specifications for stability. Stable wasteforms or container specifications for stability are necessary when waste degradation may significantly affect the licensee's ability to demonstrate that the performance objectives will be met. For example, they are necessary for wasteforms with sufficient radioactivity that its release from the disposal site would significantly affect public health and safety. Likewise, stable wasteforms may be necessary to minimize exposures to inadvertent intruders from higher activity waste, since stable wasteforms tend to provide a recognizable and non-dispersible waste. Licensees should err on the side of providing too much stability rather than too little. Stability makes it easier to demonstrate that the performance objectives will be met and decreases the likelihood that future mitigation will be needed.

Licensees should provide a technical basis for the criteria for allowable wasteform characteristics and container specifications. Licensees relying on the waste classification requirements to develop generic waste acceptance criteria may use the requirements of 10 CFR 61.56 as a basis. Licensees developing site-specific waste acceptance criteria from the results of the technical analyses should clearly identify how the acceptable wasteform characteristics and container specifications provide safety equivalent to that ensured by the requirements in 10 CFR 61.56, and they should explain how the criteria are consistent with the assumptions, modeling approaches, and results of the technical analyses. They should identify which performance objectives the criteria support and the time periods over which the criteria are intended to protect public health and safety. For example, some acceptable wasteform characteristics may only need to be relied upon to protect facility personnel during operations; others may be important to protect the general population from releases of radioactivity from longer-lived waste, or to protect intruders from inadvertent exposures. The criteria should provide reasonable assurance that the longevity required of the wasteform, and container

capabilities can be demonstrated. The basis for the criteria may include traditional engineering tests or specifications for waste containers. The criteria may also be supported by the results of laboratory testing, predictive modeling, or comparison to analogs.

The BTP on wasteform test methods (NRC, 1991b) provides guidance for waste generators on wasteform test methods that are considered acceptable for complying with the 10 CFR Part 61 stability requirements. It identifies conditions that stable wasteforms should meet to demonstrate that the stability requirements are met. The guidance applies to licensees that develop generic waste acceptance criteria from 10 CFR Part 61 waste classification requirements. It is also considered generally applicable to licensees that develop site-specific waste acceptance criteria from the results of the technical analyses required in 10 CFR 61.13. However, in some cases, the approaches in the BTP may need to be amended because of site-specific conditions. In these cases, licensees should provide a technical basis for their divergence from the guidance. Licensees may also consult IAEA-TECDOC-864, “Requirements and Methods for Low and Intermediate Level Waste Package Acceptability,” issued February 1996 (IAEA, 1996), which lists considerations for establishing container specifications. Section 3.2.4 of this document provides guidance associated with wasteforms.

Reviewers should coordinate their reviews of the acceptable wasteform characteristics and container specifications with their reviews of the technical analyses required by 10 CFR 61.13 (see Section 3.2.4, Chapter 4.0, and Chapter 5.0). Reviewers should verify that the licensee’s proposed criteria for wasteform characteristics and container specifications are consistent with the wasteforms and containers assessed in the performance assessment and intruder assessment, as well as assessments of expected and accidental occupational exposures during waste handling, storage, and disposal. Licensees should establish criteria to ensure that wasteforms and containers will provide the expected performance estimated in the technical analyses.

Reviewers should focus their review of wasteform characteristics and container specifications on the waste streams expected to contribute most significantly to risk to the public, workers, and the environment. Which wasteform characteristics and container specifications are significant will depend on the disposal site characteristics and engineering design. Reviewers should evaluate the licensee’s technical analyses to understand which wasteform characteristics and container specifications are important for demonstrating that the performance objectives will be met. These may include mechanical properties to ensure stability or limit the likelihood or consequences of potential accidents during operations. They may also include durability or leaching characteristics to minimize releases to the general environment. Furthermore, reviewers should evaluate whether the allowable wasteform characteristics and container specifications would significantly affect the stability of the disposal site after closure. For example, reviewers may need to evaluate whether waste containers could withstand anticipated mechanical loads after disposal. Reviewers should coordinate their review of the development of acceptable wasteform characteristics and container specifications with their review of site-specific allowable limits from the technical analyses (see Section 8.2.1.1).

Reviewers should evaluate whether the licensee’s technical basis will provide reasonable assurance that the performance objectives will be met. For example, if a particular wasteform characteristic or container specification is based on the results of the performance assessment, reviewers should confirm that the limit would not preclude licensees from demonstrating that the other performance objectives (e.g., protection of inadvertent intruders using the intruder assessment) are also met. Reviewers should evaluate whether the criteria ensure that the

wasteform characteristics and container specifications can be expected to persist for the duration needed to demonstrate compliance with the performance objectives. Reviewers may elect to conduct independent analyses to inform the review.

8.2.3 Restrictions or Prohibitions

The regulations in 10 CFR 61.58(a)(2)(iii) require licensees to specify restrictions or prohibitions on waste, materials, or containers that might affect the ability to meet the performance objectives. Licensees should identify specific radionuclides, chemical or hazardous materials, or containers or types of containers that are restricted or prohibited from acceptance at the facility. The restrictions and prohibitions should adequately reflect those identified in the waste characteristic requirements specified in 10 CFR 61.56(a).

Reviewers should assess the licensee's list of restrictions or prohibitions to verify it is comprehensive and adequately considers the waste characteristics requirements specified in 10 CFR 61.56(a). Reviewers should assess whether the licensee's list is consistent with the assumptions, modeling approaches, and results of the technical analyses used to demonstrate that the performance objectives will be met. Reviewers may perform independent modeling to support this review.

8.3 Waste Characterization Methods

Licensees are required, in accordance with 10 CFR 61.58(b), to provide methods for characterizing waste for acceptance. The methods shall identify the parameters to be characterized and the uncertainty in the characterization data that is considered acceptable. The regulations specify that, at a minimum, the following information is required to adequately characterize waste for acceptance:

- Physical and chemical characteristics—Information on the physical and chemical characteristics of the waste supports waste handling, the determination of compatibility with the container and other waste, and any potential treatment or conditioning processes needed. Physical characteristics may include a description of the material including its density, consistency, and appearance. Chemical characteristics may include pH, reactivity, chemical compounds present, and the presence of hazardous or toxic constituents.

- Waste volume, including any stabilization or absorbent media—Information on volume supports waste handling decisions. It is important for determining or verifying the concentration of radionuclides for comparison with the waste acceptance criteria. Volume information should include the container volume, the actual waste volume, and the container utilization factor. The container utilization factor represents the portion of the container volume that is filled with waste, including stabilization or absorbent media. The container volume should represent the volume of the disposal site that will be occupied by the container. The actual waste volume should include the volume of stabilization or absorbent media. If used, stabilization or absorbent media should be identified.

- Weight of the container and contents—Information on weight should include container weight (or mass) to be handled. Weight information may be important for meeting

stability and transportation requirements, as well as for determining or verifying the concentration of radionuclides for comparison to the waste acceptance criteria.

- Radionuclide identities, activities, and concentrations—This information may include the total activity in a container, the identities and activities of radionuclides whose activities or concentrations per unit volume or mass must meet specified allowable limits for acceptance, radiation dose levels at the surface of the container, and external contamination levels on the surface of the container as well as associated uncertainty.

- Characterization date—The characterization date helps determine the validity of the characterization documentation.

- Generating source—Identification of the generating source helps determine the validity of the characterization documentation. Information on the generating source may include packaging date, generator site, location of the process that generated the waste, and information on conditioning, if applicable.

- Any other information needed to characterize the waste to demonstrate that the waste acceptance criteria in 10 CFR 61.58(a) are met—This includes additional information about the waste that is important to the licensee's ability to demonstrate protection of public health and safety. This information should be identified from waste acceptance criteria that are drawn from the results of the technical analyses used to demonstrate that the performance objectives or the waste classification requirements will be met. For example, data on mechanical properties of wasteforms or containers may be needed to ensure that criteria for stability can be met.

The purpose of these requirements is to ensure that the licensee's knowledge of the waste's characteristics is commensurate with the assumptions and approaches employed in the technical analyses used to develop the proposed waste acceptance criteria. The knowledge of the waste characteristics should be sufficient to demonstrate that the waste acceptance criteria are met.

For generic waste acceptance criteria developed from the waste classification requirements in 10 CFR 61.55, waste characterization methods should be commensurate with the assumptions and approaches used to develop the waste classification requirements (which are summarized in Appendix F). For site-specific waste acceptance criteria developed from the technical analyses, waste characterization methods should be consistent with the assumptions and approaches employed in the analyses to derive the criteria. The waste characterization methods should provide assurance that the criteria can be met. For example, the limits for Class B and C waste in the 10 CFR 61.55 waste classification requirements were developed from an analysis that assumed Class B and C waste would be stable. As a result, 10 CFR Part 61 requires Class B and C waste to be disposed of in a stable form. Therefore, licensees may need to specify site-specific waste acceptance criteria to ensure that wastes that would be classified as Class B or C waste are stable. In this example, licensees also need to specify acceptable waste characterization methods to demonstrate that the site-specific stability criteria will be met.

Regardless of the method used to develop waste acceptance criteria, licensees should specify acceptable waste characterization methods, criteria for determining the uncertainty in the

characterization data, and the documentation required to ensure that sufficient detail is available to demonstrate that the waste acceptance criteria of the land disposal facility are met.

8.3.1 Acceptable Waste Characterization Methods

Licensees shall specify methods for adequately characterizing waste for the purposes of demonstrating that the waste acceptance criteria are met. These specifications should include methods for characterizing the radionuclide content of the waste, significant waste characteristics, and container properties. The methods should identify the characterization parameters and acceptable uncertainty in the characterization data. Licensees should use these waste characterization methods to ensure that generators provide reasonable estimates of the radionuclide content of their waste and the necessary waste characteristics for comparison with the waste acceptance criteria. Characterization methods are specific to individual waste streams and account for the radiological and other characteristics of the waste streams. The disposal facility operator should ensure that the generator's characterization data are close enough in time to when the demonstration that waste meets the acceptance criteria is performed, such that the waste stream characteristics have not changed. When proximate characterization is not possible and the time interval from characterization to disposal is large enough to significantly affect whether the waste meets the waste acceptance criteria, the disposal facility operator should make sure there is a basis supporting the assumption that the earlier characterization remain acceptable for demonstrating that the waste acceptance criteria are met. The IAEA has developed guidance on strategies and methodologies for radioactive waste characterization that licensees may find helpful for developing acceptable waste characterization methods (IAEA, 2007).

8.3.1.1 Acceptable Methods for Characterizing Activities and Concentrations

The first step in characterizing the waste is to determine the activities and concentrations of radionuclides in the waste that have allowable limits for acceptance. Acceptable methods for this characterization would likely include either direct measurement of individual radionuclides, or indirect methods that infer activities or concentrations of radionuclides from other measurements or knowledge of the waste. The BTP on waste classification and wasteform (NRC, 1983d) describes these methods in more detail.

Direct measurement of individual radionuclides provides the most confidence that the allowable activities and concentration limits identified in the waste acceptance criteria are met. However, the NRC staff recognizes that direct measurement may not always be feasible or warranted. For example, activities or concentrations of certain radionuclides may be overly difficult to measure with current technology (e.g., the may fall below minimum detection capabilities), or personnel safety considerations may limit direct measurement of certain waste streams. In these cases, licensees are permitted to accept other methods (e.g., indirect or material accountability methods) to demonstrate that the allowable limits are met.

Indirect methods infer radionuclide activities or concentrations from material accountability, characterization by source, or the use of scaling factors. Material accountability relies on inferences from the difference between the quantity of radioactive material entering a given process and the quantity exiting. Characterization by source is similar to material accountability and involves determining the radionuclide content through knowledge and control of the source of the waste. Indirect methods often rely on the use of scaling factors to relate the inferred

activity or concentration of one radionuclide to another radionuclide or gross radioactivity that is measured.

Indirect methods may be appropriate for determining activities or concentrations of difficult-to-measure radionuclides, provided there is reasonable assurance that the indirect methods can be correlated with actual measurements. Licensees should require that generators develop correlations between measured or known quantities and the inferred quantities on a generating-facility and waste-stream basis. NRC guidance on the use of indirect methods to determine the radionuclide inventory appears in "Regulatory Issue Summary 2015-02, "Reporting of H-3, C-14, Tc-99, and I-129 on the Uniform Waste Manifest," issued February 2015 (NRC, 2015b). The BTP on waste classification and wasteform (NRC, 1983d) provides additional guidance on acceptable uses of indirect methods in waste classification. This guidance may be useful for identifying methods to demonstrate that site-specific allowable limits developed from technical analyses are met. Licensees should consider the issues identified by the NRC staff in Information Notice 86-20, "Low-Level Radioactive Waste Scaling Factors, 10 CFR Part 61," dated March 28, 1986 (NRC, 1986b), when specifying criteria for the application of indirect methods to determine whether waste meets the allowable limits. NUREG/CR-6567, "Low-Level Radioactive Waste Classification, Characterization, and Assessment: Waste Streams and Neutron-Activated Metals," issued August 2000 (NRC, 2000b), and IAEA Nuclear Energy Series No. NW-T-1.18, "Determination and Use of Scaling Factors for Waste Characterization in Nuclear Power Plants," issued April 2009 (IAEA, 2009a), provide information on scaling factors that licensees may wish to consider when identifying criteria for the use of indirect methods.

Although the NRC guidance documents cited above are focused on determining concentrations for demonstrating compliance with generic allowable limits developed from the waste classification requirements of 10 CFR Part 61, they may also be helpful for demonstrating that site-specific allowable limits developed from the technical analyses are met. As part of certification, licensees should require generators to give details on how any scaling factors were derived and whether periodic re-analysis of the scaling factors resulted in a revision to the scaling factors. If a generator determines that scaling factors need to be revised, it should determine whether the revision affects previous shipments of waste to the facility. Licensees should assess how revisions to scaling factors may affect acceptable inventories, both previous disposals and future acceptable waste acceptance criteria.

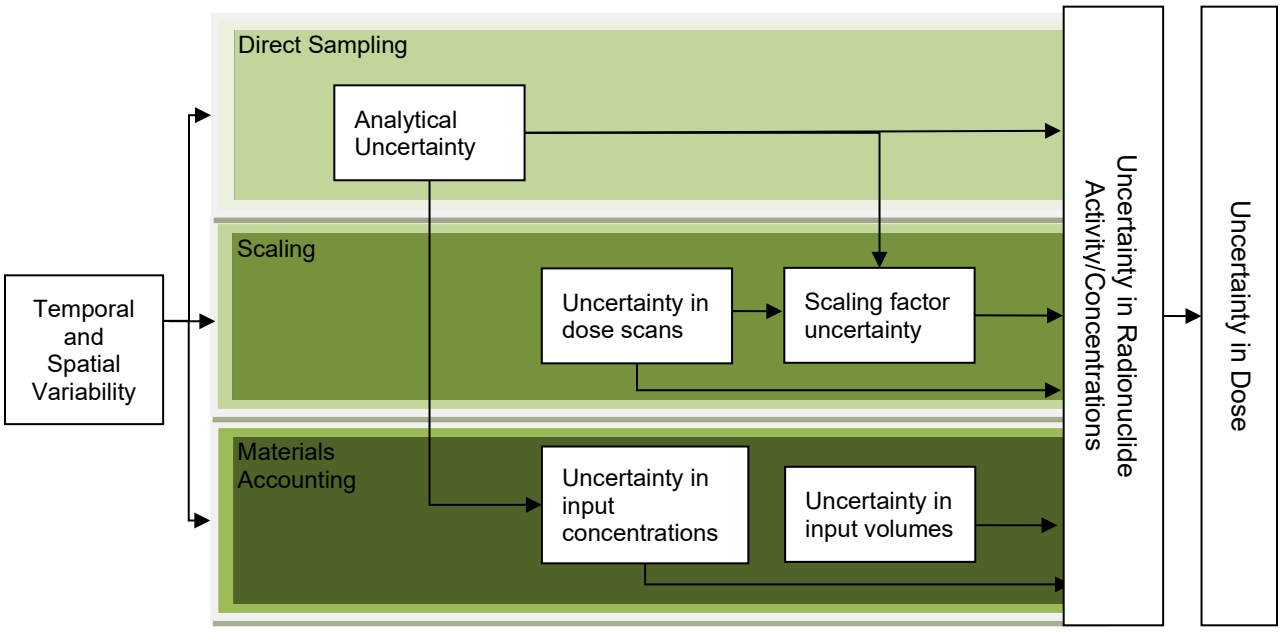


Figure 8-6 Sources of Uncertainty in Various Methods of Determining Radionuclide Activities or Concentrations in Waste Packaged for Disposal

Each of the methods described is subject to various sources of uncertainty (Figure 8-6), including, but not limited to, the following:

- sample representativeness, due to temporal and spatial variability in the following:
  - concentrations of directly sampled radionuclides
  - samples used to establish scaling factors
  - samples used to establish concentrations of process inputs
- analytical uncertainty in sampled radionuclides
- uncertainty in dose rate scans
- uncertainty in any scaling factors for unsampled containers
- uncertainty in radionuclide concentrations in inputs and input volumes, if the radionuclide concentrations in the product are based on the inputs and are not independently measured

8.3.1.1.1 Characterization Methods for Criteria Based on the Waste Classification Requirements

The BTP on waste classification and wasteform provides guidance on methods for determining concentrations to demonstrate that generic allowable limits developed from 10 CFR Part 61 waste classification requirements are met. It states that the NRC staff considers a reasonable target for determining measured or inferred radionuclide concentrations to be that concentrations are accurate to within a factor of 10. However, more precision may be required

in certain cases to demonstrate that the performance objectives will be met. Licensees should include uncertainty in the characterization of radionuclide activities and concentrations in the technical analyses.

The regulation at 10 CFR 61.55(a)(8) provides acceptable methods for determining radionuclide concentrations for comparison with allowable activities and concentrations. A licensee may average the concentration of a radionuclide over the volume of the waste, or over the weight of the waste, if the concentration is expressed in nanocuries per gram. The BTP CA contains the NRC staff's guidance on acceptable methods of concentration averaging. The BTP CA is based on many of the same methods for performing an intruder assessment that were used to develop this guidance.

In developing the BTP CA, the NRC staff considered scenarios identical to the intruder receptor scenarios described in Section 4.2 of this document, as well as receptor scenarios in which an individual may be exposed to discrete items. For instance, the staff considered a waste handling receptor scenario to develop the BTP CA for disposal of discrete sources, which is not discussed in this guidance for intruder assessments. The staff used these receptor scenarios to develop the guidance in the BTP CA on limiting the range of waste concentrations that can be mathematically averaged in a single container. The BTP CA provides specific guidance on appropriate waste volumes over which concentrations should be averaged for waste classification by waste generators or processors.

This guidance focuses on determining appropriate concentrations to use in a site-specific intruder assessment. Irrespective of concentration averaging used to determine waste classification, radionuclide concentrations should be representative of the actual waste distribution (see Section 4.3.2). Conservative assumptions are appropriate.

8.3.1.1.2 Characterization Methods for Criteria Based on the Results of the Technical Analyses

Licensees developing site-specific waste acceptance criteria from the results of the technical analyses must identify acceptable methods for characterizing the waste to demonstrate that the waste acceptance criteria are met. The appropriate method for characterizing the waste will depend on the specific parameter being measured, the hazards associated with acquiring the information, and the amount and quality of data needed to adequately characterize the waste.

The specific parameters and the quantity and quality of the data should be consistent with the intended use of the information. Characterization data should focus on those parameters of the technical analyses that are significant for a licensee's demonstration that the performance objectives will be met. Licensees should specify waste acceptance criteria that include significant waste-related parameters of the analyses. The quantity and quality of data collected for each parameter should be commensurate with the parameter's importance to meeting the performance objectives. Licensees may use a graded approach in specifying the quality of the data. Characterization data for parameters that are more significant for demonstrating that the performance objectives will be met should generally require more robust pedigree than data for parameters of lesser significance.

As discussed in Section 8.3.1.1, adequate waste characterization may include a combination of direct and indirect methods. Direct methods may include sampling and laboratory analysis, as well as certain non-destructive evaluation techniques. Indirect methods may include both

non-destructive evaluation techniques and acceptable knowledge to supplement or provide data that might otherwise be collected by direct methods. The BTP on waste classification and wasteform (NRC, 1983d) provides guidance on the use of various methods to determine concentrations to demonstrate that allowable limits developed from the waste classification requirements of 10 CFR Part 61 are met. Licensees may also use the BTP CA. The methods discussed in these documents may be appropriate for use in characterizing waste for meeting waste acceptance criteria developed from the results of the technical analyses. Licensees should provide a basis for inclusion or exclusion of the methods discussed in the BTP on waste classification and wasteform for characterizing data to meet site-specific waste acceptance criteria developed from the results of the technical analyses. The basis should include a description of why the method from the BTP on waste classification and wasteform is appropriate or inappropriate depending upon whether it is included or excluded.

8.3.1.2 Acceptable Methods for Characterizing Wasteform and Containers

The regulations in 10 CFR 61.56 specify requirements for waste characteristics, which apply to all waste classes, as well as stability requirements, which apply only to Class B, Class C, and GTCC wastes because of their higher radioactivity. Waste stability helps to limit inadvertent intrusion exposures and minimize water infiltration into the disposal units. Wastes that are stable, and thus recognizable after the active institutional control period has ended, ensure that the impacts of inadvertent intrusion remain limited to discovery-type intruder scenarios. To the extent practical, Class B, Class C, and GTCC wastes should maintain their gross physical properties and identity over a 300-year period to be consistent with the concepts in 10 CFR 61.55(a)(2). However, certain wastes may need to meet the stability requirements for longer periods of time for a licensee to demonstrate that the performance objectives will be met.

In the BTP on wasteform test methods (NRC, 1991b), the NRC staff developed guidance on wasteforms to comply with waste characteristic requirements. The guidance applies to Class A waste that is not segregated from Class B or Class C wastes. Additional requirements, specified as license conditions, may be necessary for some types of waste, including waste categorized as Class A under 10 CFR 61.55(a)(6). Regulators may specify additional requirements in license conditions based on the need to mitigate potential exposures as demonstrated in the technical analyses.

Licensees should specify methods for characterizing the wasteform and container. There are generally two stages to the process of characterizing the waste to demonstrate that acceptable wasteform characteristics and container specifications are met. First, the licensee should require generators to define the wasteform characteristics and container attributes, including performance data (e.g., compressive strength, load-bearing capability, resistance to impact, corrosion, fire resistance). The disposal facility operator should identify which quality-related parameters need to be controlled, and how they will be controlled, to provide confidence that the waste acceptance criteria are met. Second, the licensee should require generators to confirm that the wasteform or container conforms to the applicable specifications. The disposal facility operator should verify that the generator's confirmation of the applicable specifications meets the quality requirements for characterization. This may include the timeliness of the generator's confirmation. For instance, if a waste container's structural stability is important and the waste is planned to be stored for extended periods before shipment for disposal, the environment in which it is stored may need to be controlled.

8.3.2 Data Quality Objectives Process

Waste characterization data support the demonstration that the waste acceptance criteria are met. For most waste, data are derived from statistical tests based on the results of testing one or more direct samples. A licensee should use the initial assumption, or null hypothesis, that each parameter to be characterized exceeds the allowable limits specified in the waste acceptance criteria. The characterization should be designed to provide information to reject this initial assumption. The NRC staff recommends that licensees use the Data Life Cycle (as shown in Figure 8-7) as a framework for planning, implementing, and evaluating characterization results. Disposal site operators and waste generators should coordinate with each other to apply the framework or a similar methodology to the waste characterization activities. Figure 8-7 summarizes the major activities associated with each phase of the Data Life Cycle for waste characterization (NRC, 2000d).

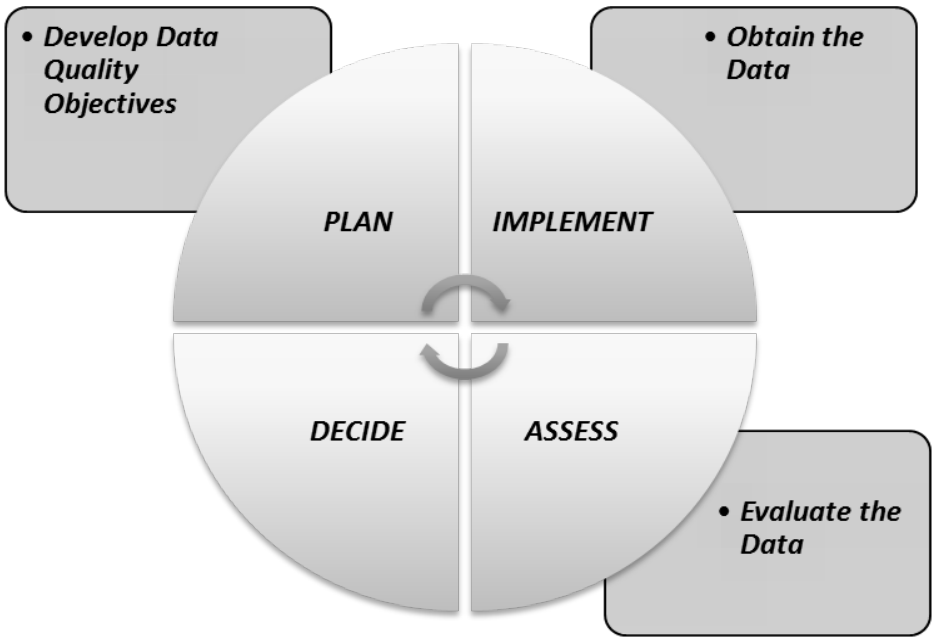


Figure 8-7 Data Life Cycle Framework for Waste Characterization

One aspect of the planning phase of the Data Life Cycle is the data quality objectives (DQO) process, a seven-step process for establishing data quality criteria and developing characterization plans:

(1) State the problem.
(2) Identify the goals of the study.
(3) Identify inputs to the decision.
(4) Define the study boundaries.
(5) Develop the analytic approach.
(6) Specify performance or acceptance criteria.
(7) Develop the plan for obtaining data.

The DQO process should use a graded approach to develop data quality requirements; that is, the level of effort expended to develop the DQOs should be commensurate with the importance of the data for demonstrating that the performance objectives will be met. A graded approach facilitates more effective characterization planning, accounting for how the data will be used.

DQOs should be qualitative or quantitative statements that do the following:

- Clarify the characterization objective.
- Define the appropriate type of data to collect.
- Determine the appropriate conditions for collecting the data.
- Specify limits on decision errors, which will serve as the basis for establishing the quantity and quality of data needed to demonstrate the waste acceptance criteria are met.

The DQO process can help ensure that the type, quantity, and quality of data will be appropriate for determining that the waste acceptance criteria are met. The EPA provides additional guidance on the Data Life Cycle and DQO process in EPA/240/B-06/001, "Guidance on Systematic Planning Using the Data Quality Objectives Process," issued February 2006 (EPA, 2006). Supplement 1 to NUREG-1575, "Multi-Agency Radiation Survey and Assessment of Materials and Equipment Manual (MARSAME)," issued January 2009 (NRC, 2009), provides guidance on applying the Data Life Cycle to disposition surveys of materials and equipment, to ensure that the surveys are adequate to meet the disposition requirements. Likewise, NUREG-1576, "Multi-Agency Radiation Laboratory Analytical Protocols Manual (MARLAP), issued July 2004" (NRC, 2004b), provides guidance on applying the Data Life Cycle to projects that require laboratory analysis of radionuclides, to ensure that the laboratory data will meet the data requirements. Licensees may find these guidance documents useful in developing DQOs for characterization data that they will use to demonstrate that the waste acceptance criteria are met.

8.3.3 Documentation

Licensees should require waste generators to provide sufficient documentation to confirm that the waste is adequately characterized to demonstrate that the waste acceptance criteria are met. The documentation that licensees require may vary across the waste streams accepted for disposal, depending on the complexity and the potential radiological impact of each waste stream. The documentation should include the following:

- Organization and responsibilities—Organizations and personnel responsible for waste characterization should be identified. Personnel responsible for collecting and managing characterization data should be properly trained to recognize the significance of the data. Qualifications of personnel should be included to provide assurance that the data are properly collected and managed. For instance, acceptable methods for waste characterization may specify the use of a certified laboratory. The documentation should include the laboratory's accreditation.

- Quality assurance—Waste characterization data should be collected according to an acceptable QA program. The documentation should identify the QA program. Section 9.1 of NUREG-1200 provides guidance on acceptable QA programs.

- Procedures—The procedures should describe the steps followed to characterize the waste, as well as the administrative processes for ensuring the type, quantity, and quality of data is appropriate to characterize the waste. Procedures should include processes for sampling, packaging, transportation, laboratory analysis, and data control. Section 8.6 of NUREG-1200 provides additional guidance on administrative and operating procedures.

- Records—Waste characterization records should include those that are necessary to meet the disposal facility's waste acceptance criteria as specified by the waste certification program (see Section 8.4). Records that contain characterization procedures, data, and specifications, including records related to the QA program, should be controlled documents that are subject to review, approval, and distribution procedures, as well as a process for making revisions. Existing record control programs may be adequate to provide the necessary controls.

8.4 Waste Certification Program

Licensees, in accordance with 10 CFR 61.58(c), must implement a program approved by the Commission to certify that waste meets the waste acceptance criteria prior to shipment to a land disposal facility. Waste certification also provides assurance that a disposal facility operates within the limits established to demonstrate that 10 CFR Part 61 performance objectives will be met. Once certified to meet a disposal facility's waste acceptance criteria, waste must then be managed to maintain its certification until its emplacement in a disposal unit.

The regulations in 10 CFR 61.58(c) specify that the waste certification program must do the following:

- Provide procedures for designating authority to certify and receive waste for disposal at the facility.

- Provide procedures for certifying that the waste to be received meets the waste acceptance criteria.

- Specify documentation required for waste acceptance including waste characterization, shipping information (including the information described in 10 CFR Part 20, Appendix G), and certification.

- Identify records, reports, tests, and inspections that are necessary to comply with the requirements in 10 CFR 61.80.

- Provide approaches for managing waste that has been certified as meeting the waste acceptance criteria in a manner that maintains its certification status.

8.4.1 Certification Program

Licensees should develop a certification program that defines administrative procedures to provide assurance that waste and its packaging meets the waste acceptance criteria of the disposal facility prior to shipment to the land disposal facility. The program should provide a traceable and verifiable record of and basis for certification. The certification program should address the following questions:

- Who is responsible for certifying that waste is acceptable for disposal and what are their qualifications?
- How and when shall waste be certified as acceptable for disposal?
- What documentation is required to provide a traceable and verifiable record for certification and how will certification be audited?
- How shall waste that has been certified be managed to maintain its certification?

The following sections provide guidance on adequately addressing these questions to meet the requirements of 10 CFR 61.58.

The principal documents that constitute the certification program should be subject to controls. Therefore, the certification program should identify which documents are to be controlled such as the waste certification program description, certification procedures, and QA program documentation. Document control includes review and approval, distribution to designated recipients, and a controlled process for document revision. Existing document control programs may provide the necessary controls for the documents that are part of the waste certification program.

8.4.1.1 Organizations and Responsibilities for Certification

The disposal facility operator is responsible for developing the requirements and managing the waste certification program. Waste certification is typically performed by the waste generator because they are the most knowledgeable about the waste and can most effectively characterize it as it is generated. If waste is stored for an extended time or sent to a waste processor, it may become more challenging and expensive to characterize it in accordance with the waste acceptance criteria. However, in some cases, such as for infrequent generators of small quantities of waste, a waste collector, a waste processor, or the disposal facility operator may be more qualified to perform the certification on behalf of the generator. The organization responsible for certification must certify that the waste is acceptable for disposal at the disposal facility according to the disposal facility's waste certification requirements, and it must obtain authorization from the disposal facility operator to transfer the waste for disposal.

A certification program should identify the individuals or organizations responsible for the certification process. These designees include representatives of the land disposal facility who are responsible for managing the certification program, and representatives of organizations responsible for complying with the land disposal facility's certification program to certify that the waste meets the acceptance criteria. These individuals or organizations may include waste generators, waste collectors, waste processors, the disposal facility operator, or other organizations or individuals qualified to certify that the waste meets the acceptance criteria. The

program should require that the disposal facility operator's designated authority identify, qualify, and approve all personnel who are designated to certify waste.

The certification program should identify the training requirements needed for the individuals involved in the program. It should require training of the official who certifies that the waste meets the acceptance criteria of the disposal facility. In addition, individuals should be trained in the procedures that control the parts of the certification process with which they are involved.

8.4.1.2 Certification Procedures

Licensees should implement the certification program using documented processes and procedures for certifying waste and for maintaining certification until the waste is emplaced in a disposal unit. The procedures should describe the administrative process that designated certification officials should follow to ensure that waste is certified prior to shipment to the land disposal facility. The procedures should require a signed statement certifying that the waste meets the disposal facility's waste acceptance criteria and, therefore, is acceptable for disposal. The signature on the certification statement confirms that the waste has been adequately characterized and necessary shipping requirements have been met.

The requirement that waste must be certified prior to shipment to the disposal facility prevents the transfer of waste that does not meet the disposal facility's waste acceptance criteria. It also prevents potential hazards associated with managing any waste rejected by the disposal facility to which it is transferred. Requiring certification before waste is shipped reduces the likelihood of having to recall a waste shipment because a certification official discovers, after the waste is in transit, that the waste does not comply with the waste acceptance criteria. Certification that the waste is ready for transfer and meets the waste acceptance criteria and the applicable transportation regulations is a control point in the transfer process. NRC Form 540 (part of the UWM) includes a certification statement that could be used as a component of the certification procedures. The procedures controlling waste transfer should not allow transfer to occur unless the certification statement has been signed. Once signed, the certification statement becomes part of the record for the transfer of the waste. Once the waste is certified as acceptable for disposal, the disposal facility can authorize transfer of the waste for receipt and disposal. The certification program should describe the administrative process for obtaining authorization from the disposal facility to transfer the waste for disposal once the waste has been certified.

The procedures should require characterization of the waste as well as inspection of the characterization process, to demonstrate that it meets the acceptance criteria. Section 8.3 provides guidance on acceptable characterization. For waste that does not meet the acceptance criteria when inspected, the procedures should specify the administrative process that a waste generator would need to follow to gain acceptance and properly certify that the waste is acceptable for disposal.

The procedures should document the steps for complying with the applicable transportation requirements for the transfer of certified waste to the land disposal facility, including those specified by the U.S. Department of Transportation; those in 10 CFR Part 71, "Packaging and Transportation of Radioactive Material;" and those in Appendix G to 10 CFR Part 20.

The procedures should also describe the process for maintaining the waste certification until the waste has been placed in a disposal unit at the land disposal facility. Section 8.4.2 provides guidance on procedures for maintaining certification. As part of certification maintenance, the

licensee should identify adequate procedures for receipt and inspection of waste at the disposal facility to ensure that arriving waste shipments are in compliance with applicable Federal regulations and the waste acceptance criteria. Section 4.1 of NUREG-1200 provides guidance on developing adequate procedures for receipt and inspection of waste arriving at a disposal facility. Section 8.6 of NUREG-1200 discusses guidance on the administrative and operating procedures that may be applicable. The IAEA has published guidance on inspection and verification of waste packages for near-surface disposal that may be applicable to developing adequate procedures (IAEA, 2000).

Finally, the procedures should describe the process for restricting access to disposal for waste generators that are not meeting the requirements of the certification program. The procedures should identify conditions that would warrant restriction of generator access to disposal, including, for example, radiological contamination, wasteform or container integrity deficiencies, improper characterization, improper manifesting, transportation violations, inadequate nuclear safety limits, and improper certification maintenance. Restrictions may include suspension of access to disposal capacity or possible heightened oversight by the disposal facility operator and its regulatory authority. The procedures should describe corrective actions necessary to restore a generator's access to the disposal facility following suspension.

8.4.1.3 Certification Documentation

The key document in the waste certification process is the certification statement. The certification statement is the documentation signed by a designated official that certifies that the waste meets the waste acceptance criteria of the disposal facility. The certification statement should contain all the information required by the certification program, including radiological properties of the waste, wasteform characteristics, and container specifications. Licensees should use the waste acceptance criteria to identify key elements to include in the waste certification statement. In addition to the certification statement, documentation should include confirmation that an official from the disposal facility to which the waste is to be transferred has authorized the transfer of the waste to the disposal facility.

The documentation supporting the waste certification statement may include or reference the following information.

- Waste Stream Profile—The waste stream profile is a description of the source, physical and chemical description, and upper limits on radionuclides of the waste stream.

- Radionuclide Content—Radionuclide content includes the concentration and inventory of radionuclides determined from waste characterization. Section 8.3 contains guidance on waste characterization.

- Radiological Surveys—Survey results include the determination of the surface contamination of the waste container and the external dose rate.

- Waste Container Attributes—Container attributes include information about the physical attributes (e.g., dimensions) of the container, as well as any procurement information relevant to certification. Disposal facility operators may require generators to provide container specifications. Container specifications should include a description of each specification's purpose, procedures for complying with the requirements of the

specification, a description of the container and manufacturing specification, and the results of tests to assess the integrity of the container.

- Uniform Low-Level Radioactive Waste Manifest—The UWM, or forms that are equivalent in content, are required by Appendix G to 10 CFR Part 20 for transfers of waste that are consigned to a licensed land disposal facility or waste collector (NRC, 2020). NRC Form 540 includes a certification statement.

- QA Records—QA records may include documentation of testing or inspections required for waste certification. They may also include a statement ensuring access for designees of the land disposal facility to perform audits and inspections, including access to the providers of procured items or services that are significant to certify that the waste acceptance criteria are met.

- Certification Maintenance Procedures—Certification maintenance procedures include the processes and controls required to maintain waste certification. Section 8.4.2 provides guidance on certification maintenance.

A licensee may use a graded approach to determine what information generators need to provide before receiving authorization to transfer waste to the disposal facility. A graded approach focuses on information that is significant for demonstrating that the waste acceptance criteria have been met and that the waste can be certified as acceptable.

The waste certification program should identify which records need to be maintained and how they are to be maintained. The certification program may detail specific records management requirements or may simply invoke an existing acceptable records management program, such as one that complies with the requirements of 10 CFR 61.80.

8.4.1.4 Audits of Certifications

A certification program should formalize procedures for independent audits of individuals or organizations designated to certify waste as acceptable for disposal by the disposal facility operator. Section 8.5 of NUREG-1200 provides guidance on plans for conducting reviews and audits of operational activities that may apply to audits of the certification process. The periodic audits should provide independent verification of the implementation of the certification program. The audit procedures should describe the principal documents of the waste certification record that will be audited and the frequency of audits.

The principal documents subject to inspection may include the waste certification statement, procurement or purchasing documents (e.g., for approved containers), radiological survey data, and laboratory testing data for characterization required to demonstrate conformance with the waste acceptance criteria. Audits may include observation of testing and characterization that are significant to demonstrating that the waste acceptance criteria are met, as well as observation of how the certification process is implemented by the certifying organization. Licensees may use a graded approach to determine the documents subject to audit and the frequency of the audits for a given certifying organization. The certification program should identify the records required to adequately document the audits, as well as the management requirements for the audit records. Licensees should maintain records of the certification audits for inspection by the NRC.

8.4.2 Certification Maintenance

Waste that has been certified as meeting the waste acceptance criteria for a land disposal facility must be controlled so that the certification remains valid until disposal at the facility. Otherwise, the waste will need to be re-certified. The waste certification program should identify the requirements for protecting the certification status of the waste. These requirements may be specific to one waste stream or applicable to all waste streams. The certifying organization should develop procedures for maintaining the waste certification that comply with the certification program's requirements. Certification maintenance may be important for waste that will be stored for long periods of time or will be treated or conditioned before disposal.

Requirements for maintaining the certification status include protecting the waste container, preventing unauthorized introduction of material into the waste, protecting the data marked on the waste container, and protecting any other capabilities relied upon for defense-in-depth. Requirements for protection of the waste container may include protection from environmental conditions during storage, conditioning, or transport (e.g., precipitation, heat, ultraviolet radiation) or designated limits for damage, should it occur. Waste may need to be controlled so as to prevent modification of the contents. These controls may include requirements for tamper indication devices and secured storage, depending on the waste stream. It is important to be able to connect each container to information about its certification. Licensees may need to have requirements for container markings to protect them from defacement or removal. Licensees should safely store all certification records.

8.5 Periodic Review

The regulation at 10 CFR 61.58(d) states that the approved waste acceptance criteria will be incorporated into the facility license and 10 CFR 61.58(e) requires disposal facility licensees to review the content and implementation of their waste acceptance program annually. The purpose of this annual review is to ensure that the content of the waste acceptance program continues to be adequate and that the program is being implemented in a way that continues to protect public health and safety. As part of this review, licensees should evaluate and document whether the waste acceptance criteria continue to protect public health and safety. If the evaluation indicates that the existing waste acceptance criteria continue to provide reasonable assurance that the performance objectives will be met, the documentation should include the basis for this finding. If the evaluation concludes that the waste acceptance criteria no longer provide reasonable assurance that the performance objectives will be met, the criteria should be updated. The licensee should submit both the amended criteria and the supporting technical analyses that demonstrate that the performance objectives will be met to the regulator as part of a request for a license amendment.

Periodic reviews should incorporate the following features to assess the procedural compliance, technical performance, implementation, and effectiveness of the facility waste acceptance program:

- Waste acceptance supervisory reviews—Onsite waste acceptance supervisors should periodically perform and document reviews of the effectiveness of the waste acceptance personnel in such areas as development of waste acceptance criteria, characterization adequacy, and procedural compliance.

- Quality assurance audits—Quality assurance audits should be performed by the onsite auditing group. Personnel in the auditing group should have sufficient waste acceptance training or experience so they can determine whether waste acceptance functions (e.g., characterization or certification) are being performed as required. These audits should also be performed periodically at generator sites.

- Corporate or contract audits—Offsite (corporate or contract) audits and evaluations should be performed to determine whether the waste acceptance program complies with the regulations and other requirements and whether objectives are being met, as well as to identify program improvements needed.

Acceptable records of periodic reviews should contain the date of the review, name of person(s) who conducted the review, persons contacted by the reviewer(s), areas reviewed, review findings, corrective actions, and follow-up. The licensee is not required to submit documentation of its periodic reviews to the NRC but should maintain the records as required in 10 CFR 61.80. However, if a licensee identifies during its periodic review a significant implication for public health and safety or common defense and security, the licensee shall notify the NRC as required by 10 CFR 61.9a(b).

9.0 SITE CLOSURE

Site closure and stabilization means those actions that are taken upon completion of operations that prepare the disposal site for postclosure observation and maintenance, institutional control, and finally, license termination. Site closure occurs after the disposal site is no longer receiving waste for disposal.

9.1 Closure Process

A licensee is required to file an application for closure in accordance with 10 CFR 61.20 and 10 CFR 61.28. Prior to final closure of the disposal site, licensees are required by 10 CFR 61.28(a) to submit an application to amend the license for site closure. The closure application must include a final revision of the safety case and a final disposal site closure plan, which updates and revises, as appropriate, the disposal site closure plan that was submitted with the license application in accordance with 10 CFR 61.12(g). The closure plan must include additional geologic, hydrologic, and other disposal site data pertinent to the long-term containment of emplaced radioactive wastes obtained during the operational period. The closure plan must include the results of tests, experiments, or any other analyses relating to backfill of excavated areas, closure and sealing, waste migration and interaction with emplacement media, or any other tests, experiments, or analysis pertinent to the long-term containment of emplaced waste within the disposal site. The closure plan must include (1) any revisions to the technical analyses necessary to support site closure that consider the waste disposed during operations and reflect significant changes to the human activities occurring in and around the site, and (2) the total volume and mass of waste that was disposed as well as the total radioactivity in curies of each radionuclide that was disposed.

Under 10 CFR 61.28(c), the NRC can only authorize closure of an LLW land disposal facility if there is reasonable assurance that the long-term performance objectives of 10 CFR Part 61, Subpart C, will be met. Licensees are required to take additional action before site closure to ensure that the LLW that has already been disposed, including significant quantities of long-lived radionuclides and other LLW streams that were not analyzed in the original 10 CFR Part 61 regulatory basis, will meet the performance objectives.

The requirement to update the technical analyses at closure is designed to ensure the following:

- The final inventory that has been disposed is evaluated in the analyses.
- The final analyses are consistent with monitoring data collected.
- The final closure plan design is evaluated in the analyses.
- New scientific or technical information that was developed during operations is reflected in the analyses.

If no significant changes have taken place since the last revision of the technical analyses, then the update to the technical analyses at site closure may require only a minimal level of effort and associated documentation. If the initial analyses or the last update of the technical analyses consistent with 10 CFR 61.24(l) appropriately accounted for the changes observed

during operations, then the licensee may submit documentation justifying the conclusion that the last update to the technical analyses is sufficient. It is appropriate for a licensee to document the changes between previous and final technical analyses and to demonstrate the impact of those changes by revising the analyses. If there are numerous and substantial changes, such as significant revisions to the conceptual models, a new engineered closure design, or new waste streams in the inventory, then a full revision of the analyses (including their documentation) is recommended for closure. The final closure analyses documentation should describe the final inventory at closure and list the quantities of all radionuclides that have been disposed.

Upon review and consideration of the application for closure, the regulator will issue an amendment authorizing closure if there is reasonable assurance that the long-term performance objectives in 10 CFR Part 61 will be met. The licensee must then maintain a period of postclosure observation and maintenance for five years to observe, monitor, and carry out necessary maintenance and repairs until the license is transferred to the disposal site owner. The period of postclosure observation may be shorter or longer than 5 years. The exact length is the time necessary to confirm that closure activities have not introduced instability to the site; this time may differ from site to site. Figure 9-1 shows the timeline from operation of an LLW disposal facility until termination of its license.

After the postclosure period, the licensee may apply to transfer the license to the disposal site owner. Transfer of the license will occur if the regulator determines the following:

- The closure plan has been properly implemented.
- Reasonable assurance has been provided that the performance objectives have been met.
- Funds for care and necessary records have been transferred to the disposal site owner.
- A postclosure monitoring program is operational for implementation by the disposal site owner.
- The Federal or State government agency that will assume responsibility for institutional control is prepared to assume control and ensure that the institutional control requirements will be met.

9.1.1 Institutional Controls

Institutional controls include administrative mechanisms to control access to the site and may include physical controls (e.g., fences, boulders, signs) to minimize disturbance to engineered measures established by the licensee to control the disposed waste. Institutional control is also described in 10 CFR 61.59(b). The institutional control program must be developed to physically control access to the disposal site following transfer of control from the licensee to the disposal site owner. Active maintenance should no longer be necessary at this point; however, necessary maintenance and custodial activities can continue. The institutional control program must provide for environmental monitoring, periodic surveillance, minor custodial care, and any other requirements and conditions imposed by the regulator. The institutional control program must also provide for the administration of funds to cover the costs of the activities.

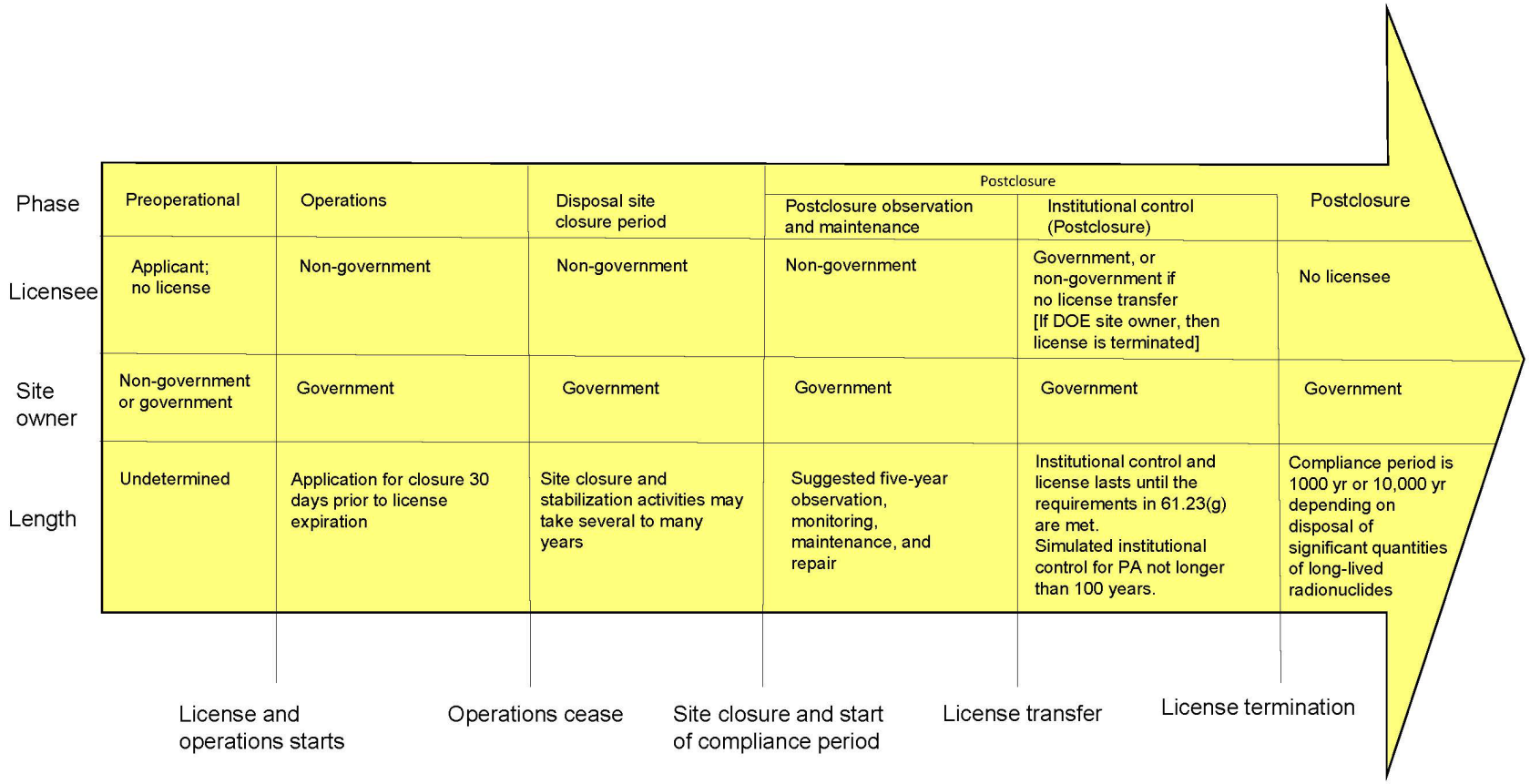


Figure 9-1 Timeline from Operations until Termination of License

In the technical analyses demonstrating that the performance objectives will be met, institutional controls cannot be relied upon for more than 100 years following transfer of control from the licensee to the site owner. A licensee may apply for an exemption to credit more than 100 years in the technical analyses. The regulator will determine the final period of institutional controls to be implemented. The licensee must consult with the Federal or State agency that has authority to administer the closed land disposal facility prior to the establishment of such institutional controls. Upon transfer of the license, the government landowner will be responsible for implementing and otherwise enforcing the institutional controls.

The postclosure institutional controls must be adequate to provide reasonable assurance of protection of public health and safety in accordance with the performance objectives in 10 CFR 61.41 and 10 CFR 61.42. In addition, they must provide reasonable assurance of the long-term stability of the disposed waste and disposal site so as to eliminate the need for maintenance.

After the institutional control period, the license may be terminated if the institutional control requirements have been met and if any additional requirements (e.g., resulting from new information developed during the institutional control period) have been met. In addition, permanent markers warning against intrusion must be installed.

9.1.2 Permanent Markers

The placement of permanent markers warning against intrusion is required in 10 CFR 61.31(c)(2). The inadvertent intruder assessment required to meet the performance objective in 10 CFR 61.42 and the development of waste acceptance criteria are the primary means of ensuring protection of public health and safety after closure of the disposal facility. As an extra layer of defense, markers can be used to deter future use of the site; however, the markers themselves should not be relied upon as engineered barriers.

Based on the siting guidelines, LLW disposal facilities will be in areas of low population density. Markers should warn people of the potential for danger should they disturb the site, but also should not invite disturbance due to curiosity or investigation. The following guidelines apply to the development of permanent markers for a commercial LLW disposal facility in the United States:

- There are no requirements for how long the markers must last. Markers should be designed to last as long as practicable, based on the information available when they are designed.

- The materials used should have little intrinsic value and should provide minimal incentive for disturbance or theft of the markers.

- Marker design should consider the present and projected future site environment. Markers should be durable and be visible for the timeframes desired.

- The marking system should be cross-cultural to the extent possible, but the use of multiple languages is not necessary.

- It may be useful to place several types of markers (detailed versus simple markers, markers made of different materials).
- Markers should communicate basic cautionary information: “Something man-made is here and it is dangerous.”
- Markers should be large enough to not be easily moved by erosion or theft.
- Markers should be placed on each boundary of the site.



Figure 9-2 Example of Simple Warning Marker

Some of the markers should be simple, like the example in Figure 9-2, which combines symbols and simple language to convey that there is danger to anyone digging in the area. More detailed markers may be provided to explain that there is radioactive material buried in the area and it should not be disturbed. However, over time, detailed text may degrade or may no longer be understood.

Figure 9-3 shows an example from Plot M at a closed burial area of the world’s first nuclear reactor, near Argonne National Laboratory in Illinois. A marker was also used at the location of the Project Gnome nuclear test in New Mexico. Each of these markers has undergone vandalism since placement. The Project Gnome marker has also been moved several meters over time, because grazing cattle use it as a scratching post.

9.1.3 Financial Assurance

Financial assurance is the mechanism through which a licensee demonstrates that funds will be available to cover the necessary costs of conducting licensed activities over the planned operating life of the project. Subpart E of 10 CFR Part 61 contains the financial assurance requirements for land disposal of radioactive waste. Licensees must demonstrate that they possess the necessary funds or have reasonable assurance of obtaining the necessary funds

(or a combination of the two). Though the requirement applies to the full life cycle of the project, this section focuses on funding for closure and stabilization.

The requirements for assuring funding of disposal site closure and stabilization operations are found in 10 CFR 61.62. The assurances are to be based on Commission-approved cost estimates reflecting the Commission-approved plan for closure and stabilization. The cost estimates must account for the total capital costs that would be incurred if an independent contractor were hired to perform the closure and stabilization work. The applicant should provide a detailed cost breakdown including an explanation of the assumptions used in the cost estimate. The licensee should develop a closure plan that eliminates the need for ongoing active maintenance to the extent practicable and ensures that only minor custodial care, surveillance, and monitoring are necessary.



Figure 9-3 Marker at Plot M in Illinois

Additionally, 10 CFR 61.62 contains the surety arrangement requirements necessary for closure and stabilization. The applicant should identify the source(s) for the funds necessary to pay for closure and stabilization.

The regulator should review the surety annually to verify that sufficient funds are available for an independent contractor to complete the closure plan. The surety amount should change based on inflation, increases in the amount of disturbed land, changes in engineering plans, and any other conditions affecting costs. Because the surety should be based on the facility life cycle, it should reflect not only the amount of land disturbed to date, but the current estimate for the total disturbance (and total waste volumes and activities to be disposed of) over the lifetime of the facility. The surety will depend on the types of engineered designs used for the facility closure plan. A facility that is highly reliant on engineered designs to meet the regulatory requirements will need relatively higher surety than one less reliant on engineered designs.

The regulation at 10 CFR 61.63, “Financial assurances for institutional controls,” contains the relevant requirements. Before issuance of the license, the applicant is required to provide for review and approval a binding agreement between the applicant and the disposal site owner to ensure that sufficient funds will be available to cover the cost of monitoring and maintenance

during the institutional control period. Monitoring includes both radiological and nonradiological monitoring to confirm that the barriers being relied upon are functioning as well as, or better than, represented in the technical analyses. The technical analyses may rely on an institutional control period of up to 100 years. However, in practice, a licensee may adopt a longer institutional control period. The licensee must provide financial assurance that is consistent with the institutional control period used. The regulator will periodically review the binding agreement to confirm that it reflects changes in inflation, technology and disposal facility operations.

9.2 Performance Confirmation

Performance confirmation is the program of tests, experiments, and analyses that licensees conduct to evaluate and verify the accuracy of information used to demonstrate that 10 CFR Part 61 performance objectives are met before disposal site closure. Licensees must update their technical analyses under 10 CFR 61.13 using details of the final closure plan and the waste inventory. Although the regulation does not use the terminology “performance confirmation,” the NRC staff believes that the elements of a performance confirmation program are supported by 10 CFR Part 61, as shown in Table 9-1.

Elements of performance confirmation may be completed during active operation and up to the time of license transfer. The following are the main elements of a performance confirmation program:

- verification that site conditions encountered during construction were within limits assumed during licensing
- verification that engineered barriers and other defense-in-depth protections were constructed as designed and will perform within limits assumed during licensing
- verification of the performance of natural barriers that were relied upon by the licensee in licensing to achieve compliance with the performance objectives
- monitoring of facility performance
- verification of the safety case

The performance confirmation program should be integrated with disposal system design, development, and construction, to provide confidence that the disposal system will perform as intended. Performance confirmation can be used to supplement information collected to satisfy the requirements of 10 CFR 61.28, 10 CFR 61.52, and 10 CFR 61.53. The more information collected for performance confirmation, the stronger the basis will be for approval of the final closure of the site.

Table 9-1 Regulatory Requirements Supportive of Performance Confirmation

Section	Requirement
10 CFR 61.12(g)	A description of the disposal site closure plan, including those features that facilitate closure and eliminate the need for maintenance
10 CFR 61.28	Contents of application for closure
10 CFR 61.52	Land disposal facility operation and disposal site closure
10 CFR 61.53(c)	Environmental monitoring during construction and operation
10 CFR 61.53(d)	Environmental monitoring, post operational surveillance

Under 10 CFR 61.24, licensee shall evaluate whether updates to the technical analyses are warranted (1) if significant changes have occurred at the site, and (2) before receiving new waste streams not analyzed in the most recent approved technical analyses. Licensees are also required under 10 CFR 61.28 to update technical analyses at closure. A performance confirmation program can be used to proactively generate information to support all of these updates. The licensee should focus on the most risk-significant and uncertain elements of the technical analyses.

In addition, 10 CFR 61.58(e) requires licensees to annually review the content and implementation of their waste acceptance program. The purpose of this review, as described further in Section 8.4, is to ensure that the content of the waste acceptance program continues to be adequate and that the program continues to protect public health and safety. As part of this annual review, licensees should evaluate and document whether the waste acceptance criteria continue to provide reasonable assurance of compliance with the 10 CFR Part 61, Subpart C, performance objectives. If the evaluation finds that the waste acceptance criteria do not provide reasonable assurance of compliance with the performance objectives, the licensee should submit revised criteria and supporting technical analyses to demonstrate that the proposed criteria meet the performance objectives, as part of a license amendment request.

Licensees should identify and implement appropriate operational practices (e.g., allowable limits) to reduce the potential need for more challenging mitigation activities (e.g., installing new barriers, removing waste). Conditions that could trigger a decision to update waste acceptance criteria, and therefore to update technical analyses, include the following:

- substantially different inventory than anticipated (e.g., in quantity, concentration, or form)
- new site information (e.g., environmental conditions, characterization data)
- new information on engineered barrier performance or other defense-in-depth protections
- monitoring data that are inconsistent with current analyses
- substantial changes to relevant scientific understanding
- use of updated dosimetry

Under 10 CFR 61.24 and 10 CFR 61.58, licensees must document the basis for their decision on whether to update the waste acceptance criteria and supporting technical analyses, and to retain these evaluations as records. New information does not always necessitate an update of the waste acceptance criteria and supporting technical analyses. The operating period of a disposal facility may extend over decades, and the NRC staff expects that new information will be developed during this time. Regular updating is advised so that licensees can evaluate relevant information generated since the previous update. Regular updating will also reduce the likelihood of unforeseen developments that could significantly impact facility performance. If significant new information becomes available, the licensee should determine, in consultation with the pertinent regulator, whether the new information warrants an update to the technical analyses.

Monitoring of environmental media is required by 10 CFR 61.53. Such monitoring provides early warning of the release of radionuclides from the disposal site before they leave the site boundary. Most monitoring systems focus on sampling environmental media, such as groundwater or the atmosphere, at some distance from the facility—for example, within the buffer zone surrounding the facility or at the site boundary. It may be useful for a licensee to identify and use performance indicators. A performance indicator is a measure of the performance of one or more subsystems of the disposal system that may be a precursor to the overall performance of the disposal system. Performance indicators provide a less direct measure of overall performance than do environmental concentrations of radioactivity, but they have the advantage of giving early warning of changes in system performance. For example, monitoring of soil moisture underneath an engineered cover may indicate increased infiltration that could lead to increased release of radioactivity from the disposal facility.

Licensees can use the technical analyses supporting demonstration of compliance with the performance objectives to determine what types of information would be most useful to monitor and when that information would be significant. Monitoring data typically have moderate to significant variability. If monitoring data are used in performance confirmation, the NRC staff recommends that the licensee describe the anticipated variability in the data to help prevent misinterpretation of observational data. In some cases, a monitoring observation may appear to

be an outlier. Licensees should use caution in such cases and should not reject data as outliers without a statistical or physical basis for doing so. If a statistical basis cannot be found, additional information should be collected. Reviewers should evaluate the technical basis for the dismissal of any outliers and determine whether it is adequate. In some cases, a technical explanation may be available (e.g., the sample was contaminated). However, in other cases, observations initially appearing to be outliers (e.g., due to measurement error) may represent complex, unanticipated phenomena.

9.3 Mitigation

In some cases, the licensee operating a land disposal facility may learn of new information that indicates that previously disposed waste may, if not mitigated, result in the performance objectives not being met in the future. For example, the new information may indicate a significant reduction in the expected performance of engineered or intruder barriers or of the site characteristics to limit radionuclide release and migration. Licensees may need to consider mitigation to reduce the impact on humans or the environment to ensure that waste disposal continues to meet the performance objectives with reasonable assurance and provides defense-in-depth protections. Mitigation could take many different forms, such as modification of the facility design or remediation of the disposed waste. This section describes when licensees should consider mitigation, what information they should provide to the NRC to demonstrate that mitigation has been implemented, and what a reviewer should evaluate to verify that mitigation has occurred.

As licensees periodically update the waste acceptance criteria and technical analyses, new information may be learned that could indicate that waste previously accepted for disposal may present an unacceptable risk to public health and safety or the environment. Additionally, under 10 CFR 61.28, prior to final closure of the disposal site, the licensee shall apply to amend the license for closure, upon which the Commission shall make a determination if there is reasonable assurance that the performance objectives of 10 CFR Part 61 will be met and that defense-in-depth protections have been provided. A component of the application for closure should include final technical analyses and a final revision of the safety case to demonstrate that the performance objectives will continue to be met after closure. If a licensee determines, either during updates to the waste acceptance criteria or technical analyses, or during the closure process, that the facility is no longer projected to meet the performance objectives, then mitigation is one way to bring the facility into compliance.

Before engaging in mitigation, licensees should describe the actions they propose to take, including the basis for those actions. The proposed actions could include modification of the disposal facility design or remediation of the disposed waste. Examples of design modifications include installation of higher-performance engineered covers, permeable treatment walls, diversion ditches, and sheet piling. The licensee should consider the results of the technical analyses in developing design modifications for mitigation. Remediation of the disposed waste may involve actions such as in situ stabilization of wasteforms (e.g., grouting) and removal of some or all of the unacceptable waste inventory for offsite disposal. The licensee should provide a technical basis demonstrating that the requirements of 10 CFR 61.43 (i.e., protection of individuals during operations) will be met during remediation. It should also perform a cost-benefit analysis to inform the selection of remedial actions.

The reviewer should evaluate the information provided by the licensee, including the technical basis for the selected actions and the alternatives considered. If design modifications are proposed, the reviewer should evaluate the technical basis for the performance of engineered barriers and determine whether the desired performance is likely to be achieved. The reviewer should also determine whether the design modifications were developed in accordance with the results of the technical analyses. If remediation is selected, the reviewer should evaluate the basis for the licensee's determination of how much waste to remove and how to remove it, as well as the basis demonstrating that the requirements of 10 CFR 61.43 will be met. Finally, the reviewer should evaluate the licensee's cost-benefit analyses to verify that the benefits of the proposed remedial actions will outweigh their negative impacts.

10.0 USE OF OTHER NRC GUIDANCE DOCUMENTS

The following tables are intended to provide references that may be useful to licensees in developing their technical analyses. Tables 10-1, 10-2, 10-3, and 10-4 present references according to the performance objectives in 10 CFR Part 61. Table 10-5 provides a list of general topics and associated references. Table 10-6 and Table 10-7 provide references for wasteforms and release and transport modeling, respectively. NUREG-1573 includes a bibliography of technical references applicable to LLW disposal (as of 2000) in its Appendices B and C that may also be useful to licensees and reviewers.

Table 10-1 Guidance Crosswalk for Performance Objective 10 CFR 61.41, Performance Assessment

Document	Description
NUREG-1573, "A Performance Assessment Methodology for Low-Level Radioactive Waste Disposal Facilities" (NRC, 2000a)	Section 2, regulatory framework for 10 CFR Part 61
	Section 3.1, performance assessment approach
NUREG-1200, "Standard Review Plan for the Review of a License Application for a Low-Level Radioactive Waste Disposal Facility," Rev 3 (NRC, 1994)	Review procedures for technical analysis
NUREG-1757, "Consolidated Decommissioning Guidance Characterization, Survey, and Determination of Radiological Criteria," Vol. 2, Rev. 2 (NRC, 2022c)	Volume 2, Section 3.5, evaluation of engineered barriers
NUREG/CR-5512, "Residual Radioactive Contamination from Decommissioning, Technical Basis for Translating Contamination Levels to Annual Total Effective Dose Equivalent" (NRC, 1992)	Volume 1, Appendix E, Table E.6, solubility (inhalation) classes
NUREG-1854, "NRC Staff Guidance for Activities Related to U.S. Department of Energy Waste Determinations: Draft Final Report for Interim Use" (NRC, 2007a)	Section 4.2, guidance on review of performance assessments for waste determinations, general technical review procedures

Table 10-2 Guidance Crosswalk for Performance Objective 10 CFR 61.42, Inadvertent Intruder Assessment

Document	Description
NUREG-1854, "NRC Staff Guidance for Activities Related to U.S. Department of Energy Waste Determinations: Draft Final Report for Interim Use" (NRC, 2007a)[1]	Section 5, Guidance on review of intruder analyses for waste determinations
NUREG-0782, "Draft Environmental Impact Statement on 10 CFR Part 61 Licensing Requirements for Land Disposal of Radioactive Waste" (NRC, 1981a) NUREG-0945, "Final Environmental Impact Statement on 10 CFR Part 61 Licensing Requirements for Land Disposal of Radioactive Waste" (NRC, 1982b) NUREG/CR-1759, "Data Base for Radioactive Waste Management, Review of Low-Level Radioactive Waste Disposal History" (directly supports NUREG-0782) (NRC, 1981b)	Describes the intruder assessment methodology used to develop waste classification tables in 10 CFR 61.55, including generic scenarios
"Concentration Averaging and Encapsulation Branch Technical Position," Revision 1 (NRC, 2015a)	Acceptable methods of concentration averaging
"Final Waste Classification and Waste Form Technical Position Papers," Rev. 0 (NRC, 1983d)	Guidance on classifying waste Various methods to determine radionuclide concentrations
"Technical Position on Waste Form (Revision 1)" (NRC, 1991b)	Guidance on wasteforms to comply with waste characteristics requirements in 10 CFR 61.56

[1] Many of these references also apply to performance objective 10 CFR 61.41.

Table 10-2 Guidance Crosswalk for Performance Objective 10 CFR 61.42, Inadvertent Intruder Assessment

Document	Description
NUREG-1573, "A Performance Assessment Methodology for Low-Level Radioactive Waste Disposal Facilities: Recommendations of NRC's Performance Assessment Working Group" (NRC, 2000a)	Section 3.3, model abstractions for a performance assessment, generally applicable to intruder assessment
NUREG-1200, "Standard Review Plan for the Review of a License Application for a Low-Level Radioactive Waste Disposal Facility," Rev. 3 (NRC, 1994)	Section 6.1, evaluation of pathways in total dose calculation
NUREG-1757, "Consolidated Decommissioning Guidance: Characterization, Survey, and Determination of Radiological Criteria," Vol. 2, Rev. 2 (NRC, 2022c)	Volume 2, Section 3.5, degradation mechanisms, capabilities of engineered barriers, summary of existing guidance and reference information for application of engineered barriers at disposal facilities
	Appendix I, Section I.6, selection of site-specific input parameters for models and providing a technical basis Appendix I, Section I.5, selection of codes/models and approaches for NRC acceptance of the codes/models
NUREG/CR-4370, "Update of Part 61 Impacts Analysis Methodology" (NRC, 1986a)	Provides scenarios and calculation approach to estimate intruder doses

Table 10-3 Guidance Crosswalk for Performance Objective 10 CFR 61.43, Operational Safety Assessment

Document	Description
NUREG-0782, "Draft Environmental Impact Statement on 10 CFR Part 61 Licensing Requirements for Land Disposal of Radioactive Waste" (NRC, 1981a) NUREG-0945 "Final Environmental Impact Statement on 10 CFR Part 61 Licensing Requirements for Land Disposal of Radioactive Waste" (NRC, 1982b) NUREG/CR-1759, "Data Base for Radioactive Waste Management, Review of Low-Level Radioactive Waste Disposal History" (directly supports NUREG-0782) (NRC, 1981b)	Describes the intruder assessment methodology and calculations used to develop waste classification tables in 10 CFR 61.55, including basic accident scenarios such as fires
NUREG-2215, "Standard Review Plan for Spent Fuel Dry Storage Systems and Facilities, Final Report" (NRC, 2020) [2]	Provides guidance on shielding calculations
NUREG-1520, "Standard Review Plan for Fuel Cycle Facilities License Applications, Final Report" Rev. 2 (NRC, 2015b)[2] NUREG-1513, "Integrated Safety Analysis Guidance Document, Final Report" (NRC, 2001)[2]	Provides guidance on the identification of hazards and initiating events

[2] This reference is located at the end of Appendix G and not listed in Chapter 11.0.

Table 10-3 Guidance Crosswalk for Performance Objective 10 CFR 61.43, Operational Safety Assessment

Document	Description
NUREG/CR-2300, "PRA Procedures Guide—A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants" (NRC, 1983)[2]	Provides guidance that may be applicable to operational safety assessment
NUREG/CR-6410, "Nuclear Fuel Cycle Facility Accident Analysis Handbook" (NRC, 1998)[2]	Provides guidance on how to estimate the releases of radioactive materials from nuclear facilities

Table 10-4 Guidance Crosswalk for Performance Objective 10 CFR 61.44, Site Stability Assessment

Document	Description
NUREG-1623, "Design of Erosion Protection for Long-Term Stabilization" (NRC, 2002b)	Design of erosion protection at uranium mill tailings sites Procedure for determining the suitability of a rock source
NUREG-1804, "Yucca Mountain Review Plan," Rev. 2 (NRC, 2003c)	Seismic events in waste disposal
NUREG-1200, "Standard Review Plan for the Review of a License Application for a Low-Level Radioactive Waste Disposal Facility," Rev. 3 (NRC, 1994)	Site stability analysis for long-lived waste
NUREG-1757, "Consolidated Decommissioning Guidance: Decommissioning Process for Materials Licensees," Vol. 1, Rev. 2 (NRC, 2006) NUREG-1757, "Consolidated Decommissioning Guidance: Characterization, Survey, and Determination of Radiological Criteria," Vol. 2, Rev. 2 (NRC, 2022c)	Section 3.5, risk-informed approach to engineered barriers
	Analogs for wasteform stability
	Durability of earthen covers
	Section 3.5.5 (NRC, 2006) and Section P.3 (NRC, 2022c), reference information regarding engineered cover design and performance
	Appendix P, evaluations of rock durability
NUREG/CR-2642, "Long-Term Survivability of Riprap for Armoring Uranium Mill Tailings and Covers: A Literature Review" (NRC, 1982c)	Table 6.7, comparative data on natural materials
	Appendix A, information on rock weathering, durability, examples of analogs
"Technical Position on Waste Form (Revision 1)" (NRC, 1991b)	Specific test procedures and criteria to evaluate wasteform stability
NUREG-0902, "Site Suitability, Selection and Characterization, Branch Technical Position – Low-Level Waste Licensing Branch" (NRC, 1982e)	Provides additional information on processes to be avoided that may affect site stability

Table 10-5 Guidance Crosswalk by Topic

Topic	Document
Analysis Timeframe	NUREG-1573, "A Performance Assessment Methodology for Low-Level Radioactive Waste Disposal Facilities: Recommendations of NRC's Performance Assessment Working Group" (NRC, 2000a) "Technical Analysis Supporting Definition of Period of Performance for Low-Level Waste Disposal" (NRC, 2011c)
Radon Diffusion and Barriers	Regulatory Guide 3.64, "Calculation of Radon Flux Attenuation by Earthen Uranium Mill Tailings Covers" (NRC, 1989a) NUREG/CR-4370, "Update of Part 61 Impacts Analysis Methodology" (NRC, 1986a)
Sensitivity/Uncertainty Analysis	NUREG-1757, "Consolidated Decommissioning Guidance: Characterization, Survey, and Determination of Radiological Criteria," Vol. 2, Rev. 2, Appendix I, Section 1.7 (NRC, 2022c) NUREG-1573, Section 3.3.2 (NRC, 2000a)
Expert Elicitation and Judgment	NUREG-1563, "Branch Technical Position on the Use of Expert Elicitation in the High-Level Radioactive Waste Program" (NRC, 1996) NUREG-2255, "Guidance for Conducting Expert Elicitation in Risk-Informed Decision-making Activities" (NRC, 2022b)

Table 10-5 Guidance Crosswalk by Topic

Topic	Document
Modelling Climate and Infiltration	NUREG-1573, Section 3.3.3 (NRC, 2000a) NUREG-1854, "NRC Staff Guidance for Activities Related to U.S. Department of Energy Waste Determinations: Draft Final Report for Interim Use," Section 4.3.1 (NRC, 2007a)
Waste Acceptance Criteria	"Branch Technical Position on Concentration Averaging and Encapsulation," Rev. 1 (NRC, 2015a) "Final Waste Classification and Waste Form Technical Position Papers," Rev. 0 (NRC, 1983d) "Technical Position on Waste Form," Rev. 1 (NRC, 1991b) NUREG-1200, "Standard Review Plan for the Review of a License Application for a Low-Level Radioactive Waste Disposal Facility," Rev. 3, Section 4.1 and 6.11 (NRC, 1994)

Table 10-5 Guidance Crosswalk by Topic

Topic	Document
Waste Characterization	See documents listed under “Waste Acceptance Criteria” (NRC, 1983d; NRC, 2015a; NRC, 1991b) NUREG/CR-6567, “Low-Level Radioactive Waste Classification, Characterization, and Assessment: Waste Streams and Neutron-Activated Metals” (NRC, 2000b) NUREG-1575, Supplement 1, “Multi-Agency Radiation Survey and Assessment of Materials and Equipment (MARSAME)” (NRC, 2009) NUREG-1576, “Multi-Agency Radiological Laboratory Analytical Protocols Manual (MARLAP)” (NRC, 2004b) NUREG-1200, Section 4, Facility Operations (NRC, 1994)
Waste Certification	NUREG-1200, Section 4, Facility Operations (NRC, 1994) NUREG/BR-0204, “Instructions for Completing NRC’s Uniform Low-Level Radioactive Waste Manifest,” Rev. 3 (NRC, 2020a)
Source Term Modeling	NUREG-1573, Section 3.3.5 (NRC, 2000a) NUREG-1854, Section 4.3.3 (NRC, 2007a)
Inventory Abstraction	NUREG-1573, Sections 3.3.5.1, 3.3.5.2, and 3.3.5.7 (NRC, 2000a) NUREG-1854, Sections 3.1 and 4.3.3.1.1 (NRC, 2007a)
Chemical Environment and Abstraction	NUREG-1573, Section 3.3.5.6 (NRC, 2000a)

Table 10-5 Guidance Crosswalk by Topic

Topic	Document
	NUREG-1854, Section 4.3.3.1.4 (NRC, 2007a) NUREG-1804, "Yucca Mountain Review Plan," Rev.2, Section 2.2.1.3.3 (NRC, 2003c)
Waste Containers, Wasteform, and Waste Type	NUREG-1573, Section 3.3.5.3, 3.3.5.4 (NRC, 2000a)
Wasteforms and Degradation	NUREG-1854, Section 4.3.3.1.2 (NRC, 2007a)
Engineered Barriers	NUREG-1573, Section 3.3.4 (NRC, 2000a) NUREG-1854, Section 4.3.2 (NRC, 2007a) NUREG-1804, Revision 2, Sections 2.2.1.3.1 and 2.2.1.3.2 (NRC, 2003c)
Aqueous Release Models	NUREG-1854, Section 4.3.4 (NRC, 2007a) NUREG-1804, Revision 2, Section 2.2.1.3.4 (NRC, 2003c)
Gaseous Release Screening, Processes Generating Gases	NUREG-1573, Sections 3.3.5.7.1 and 3.3.5.7.2 (NRC, 2000a)
Biosphere Characteristics and Dose Modeling	NUREG-1573, Section 3.3.7 (NRC, 2000a) NUREG-1854, Section 4.3.5 (NRC, 2007a)
Transport Modeling	NUREG-1573, Section 3.3.6 (NRC, 2000a) NUREG-1854, Section 4.3.4 (NRC, 2007a)

Table 10-5 Guidance Crosswalk by Topic

Topic	Document
Groundwater Transport	NUREG-1573, Section 3.3.6.1 (NRC, 2000a) NUREG-1854, Section 4.3.4.1.3 and 4.3.4.1.4 (NRC, 2007a)
Surface Water Transport	NUREG-1573, Section 3.3.6.2 (NRC, 2000a) NUREG-1854, Section 4.3.4.1.2 (NRC, 2007a)
Atmospheric Transport	NUREG-1573, Sections 3.3.6.3 (NRC, 2000a) NUREG-1854, Section 4.3.4.1.1 (NRC, 2007a)
Development of a License Application for an LLW Disposal Facility	NUREG-1199, "Standard Format and Content of a License Application for a Low-Level Radioactive Waste Disposal Facility," Rev. 2 (NRC, 1991a) Regulatory Guide 4.18, "Standard Format and Content of Environmental Reports for Near-Surface Disposal of Radioactive Waste" (NRC, 1983e) NUREG-1300, "Environmental Standard Review Plan for the Review of a License Application for a Low-Level Radioactive Waste Disposal Facility" (NRC, 1987)
Defense-in-Depth	NUREG/KM-009, "Historical Review and Observations of Defense-in-Depth" (NRC, 2016a) "Risk-Informed Decisionmaking for Nuclear Material and Waste Applications," Rev. 1 (NRC, 2008)

Table 10-5 Guidance Crosswalk by Topic

Topic	Document
Environmental Monitoring	NUREG-1388, “Environmental Monitoring of Low-Level Radioactive Waste Disposal Facility” (NRC, 1989e)
Alternate Disposal Methods	NUREG-1241, “Licensing of Alternative Methods of Disposal of Low-Level Radioactive Waste” (NRC, 1986c)
Site Selection	NUREG-0902, “Site Suitability, Selection and Characterization, Branch Technical Position—Low-Level Waste Licensing Branch” (NRC, 1982e) Regulatory Guide, 4.19, “Guidance for Selecting Sites for Near-Surface Disposal of Low-Level Radioactive Waste” (NRC, 1988b)
Scenario Development	NUREG/CR-5927 (SAND91-2802), Vol. 1, “Evaluation of a Performance Assessment Methodology for Low-Level Radioactive Waste Disposal Facilities: Evaluation of Modeling Approaches” (NRC, 1993a) NUREG-1199, “Standard Format and Content of a License Application for a Low-Level Radioactive Waste Disposal Facility,” Rev. 2 (NRC, 1991a) NUREG-1623, “Design of Erosion Protection for Long-Term Stabilization” (NRC, 2002b)

Table 10-6 Additional References on Wasteforms and Release

Description	Reference
Provides acceptable methods to demonstrate waste stability requirements. Performance confirmation and qualification testing for cement stabilized wasteforms are described.	"Technical Position on Waste Form (Revision 1)" (NRC, 1991b)
Provides an examination of the geochemical aspects of radioactive waste disposal. Chapter 12 provides a summary of different types of wasteforms including compiled performance data.	*Geochemical Aspects of Radioactive Waste Disposal* (Brookins, 1984)
Provides a literature review and assessment of the factors affecting release from grouted systems.	CNWRA 2009-001, "Review of Literature and Assessment of Factors Relevant to Performance of Grouted Systems for Radioactive Waste Disposal" (Pabalan et al., 2009)
Provides data and discussion on release rates and test methods to estimate the long-term performance of cement and slag wastes (test method discussion and examples are applicable to many types of wasteforms that may be used for LLW disposal).	NUREG/CR-7025, "Radionuclide Release from Slag and Concrete Waste Materials Part 1: Conceptual Models of Leaching from Complex Materials and Laboratory Test Methods" (NRC, 2010a)
Provides discussion and comparison of laboratory and field leach test results with consideration of the factors relevant to translating from different time and spatial scales.	NUREG/CR-7105, "Radionuclide Release from Slag and Concrete Waste Materials Part 2: Relationship Between Laboratory Tests and Field Leaching" (NRC, 2011)
Provides a literature review of radionuclide chelating agents that may be found in LLW.	NUREG/CR-6124 (PNL-8856), "Characterization of Radionuclide-Chelating Agent Complexes Found in Low-Level Radioactive Decontamination Waste" (NRC, 1996a)
Discusses the potential for microbial degradation of LLW.	NUREG/CR-6341 (INEL-95/0215), "Microbial Degradation of Low-Level Radioactive Waste" (NRC, 1996b)

Table 10-6 Additional References on Wasteforms and Release

Description	Reference
Provides a test method to measure the leaching of solidified wastes (results can be sensitive to test interval).	ANSI/ANS-16.1-2003 (reapproved 2008), "Measurement of the Leachability of Solidified Low-Level Radioactive Wastes by a Short-term Test Procedure" (ANS, 2008)
Provides a test method to measure diffusive releases from solidified wastes.	ASTM Standard C1308-08, "Standard Test Method for Accelerated Leach Test for Diffusive Releases from Solidified Waste and a Computer Program to Model Diffusive, Fractional Leaching from Cylindrical Waste Forms" (ASTM, 2009)
Provides a summary of wasteform testing in Chapter 5 and a discussion of various test method in an appendix to Chapter 5. Discussion of wasteform performance (focused on HLW) is provided in Chapter 7.	"Waste Forms Technology and Performance: Final Report" (National Research Council, 2011)
Provides a detailed discussion of glass corrosion and processes including field testing and observations of natural glasses. A list of test methods is provided in Table 1-2. International experience is summarized in Appendix B.	"High-Level Waste Borosilicate Glass: A Compendium of Corrosion Characteristics, Volume II" (DOE, 1994)
Provides discussion of methods used to estimate container performance as well as different equations and methods to represent waste release.	"Waste Container and Waste Package Performance Modeling to Support Safety Assessment of Low and Intermediate-Level Radioactive Waste Disposal" (Sullivan, 2004)
Provides an example of using modern performance assessment software to execute a legacy model of waste release.	SAND2007-1354, "A User's Guide to the GoldSim/BLT-MS Integrated Software Package: A Low-Level Radioactive Waste Disposal Performance Assessment Model" (Mattie et al., 2007)
Provides a comprehensive source of conference papers on all types of wasteform experiments, observations, assessment, and modeling.	Materials Research Society Symposium Proceedings (numerous over multiple decades)

Table 10-7 Additional References on Transport Modeling

Topic[3]	Description	Reference
UZ	Chapter 10 provides a summary of contaminant transport considerations for the unsaturated zone.	*Unsaturated Zone Hydrology for Scientists and Engineers* (Tindall, 1999)
UZ	Section 5.5 covers unsaturated zone flow and transport.	NCRP Report No. 152, "Performance Assessment of Near-Surface Facilities for Disposal of Low-Level Radioactive Waste" (NCRP 2005)
UZ	Discusses the application of groundwater models	*Ground Water Models: Scientific and Regulatory Applications* (NAS/NRC, 1990)
SZ	Chapter 4 discusses analysis of groundwater transport of radionuclides.	NCRP Report No. 76, "Radiological Assessment: Predicting the Transport, Bioaccumulation, and Uptake by Man of Radionuclides Released to the Environment" (NCRP, 1984)
SZ	General text on groundwater hydrology and transport.	*Physical and Chemical Hydrogeology* (Domenico and Schwartz, 1998)
SZ	Provides a compilation of distribution coefficients for four major soil types.	"Default Soil Solid/Liquid Partition Coefficients, K_d, for Four Major Soil Types: A Compendium," *Health Physics*, 59: 471–482 (Sheppard and Thibault, 1990)
SZ	Text describing groundwater hydrology.	*Hydraulics of Groundwater* (Bear, 1979)
SZ	Provides parameter estimation of saturated zone properties. Provides a comparison of numerical modeling with analytical modeling results.	NUREG/CR-6114, Vol. 3, "Auxiliary Analyses in Support of Performance Assessment of a Hypothetical Low-Level Waste Facility: Groundwater Flow and Transport Simulation" (NRC, 1994a)

[3] UZ = unsaturated, SZ = saturated, SW = surface water, AT = atmospheric, BT = biotic

Table 10-7 Additional References on Transport Modeling

Topic[3]	Description	Reference
SW	Covers multiple concepts and provides observations of transport of radionuclides in surface water.	"Surface Water Transport of Radionuclides," in *Radiological Risk Assessment and Environmental Analysis*, pp. 147–207 (Onishi, 2008)
SW	Provides screening models for surface water transport of radionuclides.	NCRP Report No. 123, Vol. I, "Screening Models for Releases of Radionuclides to Atmosphere, Surface Water, and Ground" (NCRP, 1996a) NCRP Report No. 123, Vol. II, "Screening Models for Releases of Radionuclides to Atmosphere, Surface Water, and Ground—Worksheets" (NCRP, 1996b)
SW	Provides analytical methods for estimating dispersion of radionuclides in different types of surface water bodies.	Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I" (NRC, 1977b)
AT	Covers multiple concepts and provides observations of transport of radionuclides in the atmosphere.	"Atmospheric Transport of Radionuclides," in *Radiological Risk Assessment and Environmental Analysis*, chapter 3 (Crawford et al., 2008)
AT	Provides comparisons of different atmospheric modeling codes and methods.	NUREG/CR-6853, "Comparison of Average Transport and Dispersion Among a Gaussian, a Two-Dimensional, and a Three-Dimensional Model" (NRC, 2004)
AT	Provides a description of a radiological assessment tool used for nuclear accidents.	NUREG-1741, "RASCAL 3.0: Description of Models and Methods" (NRC, 2001)
AT	Provides a description of a radiological assessment tool used for nuclear accidents.	NUREG-1940, "RASCAL 4.0: Description of Models and Methods" (NRC, 2012a)

Table 10-7 Additional References on Transport Modeling

Topic[3]	Description	Reference
BT	Discusses the importance of biotic pathways with respect to radioactive waste disposal.	NUREG/CR-2675 (PNL-4241), Vol. 1, "Relevance of Biotic Pathways to the Long-Term Regulation of Nuclear-Waste Disposal: A Report on Tasks 1 and 2 of Phase 1" (NRC, 1982d)
BT	Provides numerous references and discussion of biological processes impacting release and transport.	NCRP Report No. 152, "Performance Assessment of Near-Surface Facilities for Disposal of Low-Level Radioactive Waste" (NCRP, 2005)

Additional Resources

The following web links provide additional information on radionuclide transport that may be useful to stakeholders. As of April 2026, these website addresses are valid; over time, these addresses may no longer be accurate due to the fluid nature of the Internet. If the actual link does not work, the NRC staff recommends going to the highest level of these web links to find the information (e.g., www.usgs.gov).

http://groundwater.ucdavis.edu/Materials/Vadose_Zone_Modeling__Web-Links/ (UZ modeling)

http://water.usgs.gov/software/lists/groundwater/ (software list)

http://water.usgs.gov/software/lists/surface_water/ (surface water software)

https://ramp.nrc-gateway.gov/ (radiological protection codes and analysis)

11.0 REFERENCES

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12.0 GLOSSARY

Accident scenario: See *Scenario*.

Alternative conceptual model: An additional and different model on how the system might work that is consistent with available supporting information. For example, a site scenario may have two plausible conceptual models: a fracture flow conceptual model and an alternative matrix flow conceptual model. The model outputs from each may yield significantly different results.

Alternative site scenario: See *Scenario*.

Analysis timeframe: The timeframe over which a licensee should assess the projected performance of the disposal facility factoring in the characteristics of the waste, engineered barriers, disposal site, and associated uncertainties. The analysis timeframe is divided into two phases: a compliance period and a performance period.

Assessment context: Provides a framework for the performance assessment and covers the following key aspects: purpose, regulatory framework, assessment endpoints, assessment philosophy, disposal system (or facility) characteristics, and timeframes.

Buffer zone: Portion of the disposal site that is controlled by the licensee and that lies under the disposal units and between the disposal units and the boundary of the site. The buffer zone provides controlled space to establish monitoring locations that are intended to provide an early warning of radionuclide movement, which allows a licensee to perform mitigative or corrective action that might be necessary.

Central site scenario: See *Scenario*.

Code: A set of software commands used to solve mathematical equations representing phenomena of the conceptual model.

Compliance period: The time from the completion of site closure to 1,000 years after site closure for disposal sites that do not contain significant quantities of long-lived radionuclides. For disposal sites that contain significant quantities of long-lived radionuclides, the compliance period ends 10,000 years after closure of the disposal site.

Computational model: See *Numerical model.*

Conceptual model: A well-defined, connected sequence of phenomena describing the behavior of the system of concern.

Critical group*:* A group of individuals reasonably expected to receive the greatest exposure to releases over time, given the circumstances under which the analysis would be carried out. The average member of the critical group is assumed to represent that individual in the most likely exposure situation, based on cautious but reasonable exposure assumptions and parameter values.

Defense-in-depth: The use of multiple independent and, where possible, redundant layers of defense against release of radioactive material such that no single layer, no matter how robust, is exclusively relied upon.

Degradation: A process of gradual reduction in the capability of materials used in the construction of LLW disposal facilities to limit water infiltration and the release of radionuclides; the decline of an engineered barrier following the service life when important characteristics of an engineered barrier progress from an expected design value to the degraded condition.

Deterministic analysis: An analysis using a single set of values for key assumptions or parameters to calculate a single value of model output.

Disposal: The removal of radioactive wastes from the biosphere inhabited by a person and containing the person's food chains by emplacement in a land disposal facility.

Disposal site: That portion of a land disposal facility which is used for disposal of waste. It consists of disposal units and a buffer zone.

Disposal unit: A discrete portion of the disposal site into which waste is placed for disposal (e.g., a trench).

Distribution coefficient (K_d): An empirical constant employed in mathematical expressions representing sorption isotherms that relate the mass of solute on the solid phase to the concentration of solute in solution as a function of temperature and pressure. The distribution coefficient (K_d) represents an empirical constant for a linear sorption isotherm, the validity of which requires that the reactions that cause the partitioning are fast and reversible (e.g., chemical equilibrium is achieved) and the sorption isotherm is linear.

Dose: Generically refers to radiation dose, absorbed dose, dose equivalent, effective dose equivalent, committed dose equivalent, committed effective dose equivalent, or total effective dose equivalent.

Dosimetry: The process or method of measuring the dosage of ionizing radiation.

Engineered barrier: A man-made structure or device that is intended to improve the disposal site's ability to meet the performance objectives in 10 CFR Part 61, Subpart C. Examples of engineered barriers include wasteforms, intruder barriers, resistive covers, and evapotranspiration or water balance covers.

Exposure pathway: The route by which radioactivity travels through the environment to produce radiation exposure to a person or group.

Event: A phenomenon or change that has the potential to affect the performance of the disposal system and that occurs during an interval that is short compared to the analyses timeframes. Examples of events that cause relative rapid change are earthquakes, floods, storms, well drilling, and excavation.

Feature: An object, structure, or characteristic that has a potential to affect the performance of the disposal system. Examples may include a wasteform, a compacted clay layer, or the rocks within an erosion layer of an engineered cover.

FEP: Stands for feature, event, or process. Features, events, or processes (FEPs) have the potential to affect the performance of the disposal system.

FEP screening: The process of using regulatory, probability, and consequence criteria to eliminate or exclude FEPs from further consideration due to their insignificant impact on the performance of the disposal system or because they are otherwise excluded by regulation.

Hazard: An FEP that can cause harm. In waste disposal, the radiological inventory represents a hazard, but, if contained, does not present risk to the public.

Inadvertent intruder: A person who might occupy the disposal site after closure and engage in agricultural and residential activities and other reasonably foreseeable pursuits that might unknowingly expose the person to radiation emitted or released from the waste in the disposal units.

Inadvertent intruder assessment: An analysis that: (1) assumes an inadvertent intruder occupies the disposal site and engages in activities and other reasonably foreseeable pursuits consistent with expected activities in and around the disposal site at the time of the assessment and that might unknowingly expose the person to radiation emitted or released from the waste in the disposal units; (2) examines the capabilities of intruder barriers to inhibit an inadvertent intruder's contact with the waste in the disposal unit or to limit the inadvertent intruder's exposure to radiation from the disposal unit; and (3) estimates an inadvertent intruder's potential annual dose resulting from radiation emitted or released from the waste in the disposal unit, including an evaluation of the uncertainties. Intruder assessments are generally constrained to a limited set of receptor scenarios to avoid excessive speculation about future human behavior. An intruder assessment is used to demonstrate compliance with 10 CFR 61.42(a) and 10 CFR 61.42(b).

Intruder scenario: See *Scenario*.

Institutional controls: Measures to control access to a site and minimize disturbances to engineered measures established by the licensee to control the residual radioactivity. Institutional controls include administrative mechanisms (e.g., land use restrictions) and may include, but are not limited to, physical controls (e.g., signs, markers, landscaping, and fences).

Land disposal facility: The land, building, structures, disposal sites, and equipment that are intended to be used for, or to support, the disposal of radioactive wastes. A land disposal facility includes both a near-surface disposal facility and a specialized land disposal facility.

Licensee: A person possessing a license to dispose of waste in a land disposal facility. This document uses the term "licensees" to refer to licensees or license applicants, as applicable. In instances where a regulatory requirement is being referenced, the reader should refer to the regulation to determine whether the guidance applies in a particular case.

Long-lived waste: Waste containing radionuclides (1) where more than ten percent of the initial activity of the radionuclide remains after 1,000 years, (2) where the peak activity from

progeny occurs after 1,000 years, or (3) where more than ten percent of the peak activity of a radionuclide (including progeny) within 1,000 years remains after 1,000 years. The first part of the definition represents radionuclides with approximately a 300-year or longer half-life. Examples of isotopes that are short-lived but produce long-lived progeny are Am-241 and Cm-242.

Low-level (radioactive) waste (LLW): Items that have become contaminated with radioactive material or have become radioactive through exposure to radiation. The radioactivity in these wastes can range from just above natural background levels to much higher levels, such as seen in parts from inside the reactor vessel in a nuclear power plant. LLW is defined by what it is not, so that an understanding of the definitions of HLW, spent nuclear fuel, byproduct material, and naturally occurring radioactive material is necessary to determine whether a subject waste is LLW. The ILLRWD rule removes the exclusion of TRU from the definition of LLW.

Maintenance: Any remedial, monitoring, or custodial activity performed at the site. Maintenance can be grouped into two types of activities. First, active maintenance includes remedial activities needed to maintain reasonable assurance that the performance objectives in 10 CFR 61.41 and 10 CFR 61.42 are met. Examples of active maintenance activities, assumed to be carried out before site closure, include pumping and treatment of water from a disposal unit or one-time measures such as replacement of a disposal unit cover. Second, necessary maintenance and custodial activities performed after site closure can include surveillance, monitoring, and custodial activities, including revegetation, minor additions to soil cover, minor repair of disposal unit covers, and general disposal site upkeep such as mowing grass, repairing fencing, or repairing or replacing monitoring equipment. No maintenance should be necessary after the institutional control period ends.

Mathematical model: A representation of a conceptual model in mathematical terms (i.e., a governing equation or set of equations intended to represent important processes). Mathematical models can be solved analytically or numerically.

Member of the public: An individual in a controlled or unrestricted area. However, an individual is not a member of the public during a period in which the individual receives an occupational dose.

Model: A conceptual or mathematical representation of a system used to project future performance.

Model abstraction: The process of abstracting a conceptual model representing a disposal site in the physical world into a mathematical model governed by equations that is implemented within a numerical model.

Model integration: The connection of models, submodels, and abstractions at the level of detail necessary to represent the conceptual model. For example, a model simulating precipitation may be integrated with models of infiltration and erosion.

Model simplification*:* The process of simplifying a complex numerical model into a reduced numerical model while still maintaining the validity of the simulation results.

Model support: Data and information that technically support the development of the numerical models or assessments and provide confidence in their results. Model support that involves multiple sources and types of information is generally more robust and can include laboratory or field tests, comparison to analogous systems, natural analogs, formal independent peer review, and comparison to monitoring data.

Model uncertainty *(also referred to as "conceptual model uncertainty")*: The uncertainty in the conceptualization of the system, the uncertainty in its mathematical representation, and the uncertainty in the solution of the mathematical representation.

Monitoring: The collection of field observations and measurement data to evaluate the performance and characteristics of the disposal site.

Near-surface disposal facility: A land disposal facility in which radioactive waste is disposed generally within the upper 30 m of the earth's surface.

Numerical model: A model to solve the equations of the mathematical model using codes or modeling software. The results of the simulations can represent, for example, potential radiological exposures and their associated uncertainties.

Operational safety assessment: An assessment used to provide reasonable assurance that exposures will be controlled to meet the requirements of 10 CFR Part 20, thereby meeting the performance objective for the protection of individuals during operations set forth in 10 CFR 61.43. An operational safety assessment is more detailed and comprehensive as the level of hazard posed by the waste increases.

Parameter uncertainty: Uncertainty associated with the input to the numerical model being used in the analysis including uncertainty in the actual values and the statistical and spatial distributions of data use to infer model parameters. Parameter uncertainty is highly dependent on the quality of the data.

Pathway: Route or means of release of contaminants from a disposal facility, transport of contaminants in the environment, or exposure of humans.

Performance assessment: An analysis used to demonstrate compliance with 10 CFR 61.41(a) and 10 CFR 61.41(b) that identifies the features, events, and processes that could affect the performance of the disposal site; and estimates the potential dose as a result of releases caused by all significant features, events, and processes including an evaluation of the uncertainties.

Performance period: The timeframe after the compliance period during which waste that contains significant quantities of long-lived radionuclides remains a radiological hazard..

Performance period analyses*:* Analyses used to demonstrate compliance with 10 CFR 61.41(b) and 10 CFR 61.42(b) by providing information, consistent with available data and current scientific understanding, that demonstrates that releases of long-lived radioactive waste from a disposal site are reduced to the extent reasonably achievable during the performance period. Performance period analyses are required if the disposal facility is accepting significant quantities of long-lived waste.

Phenomenon: Either a process or an event. Typically, a phenomenon acts upon a feature.

Probabilistic analysis: Refers to computer codes or analyses that use a sampling method to select parameter values from a distribution. Results of the calculations are also in the form of a distribution of values or time series of different values.

Process: A phenomenon or change that has the potential to affect the performance of the disposal system and that occurs during all or a significant part of the analyses timeframe. Examples of processes that cause relative gradual change are radionuclide transport, differential settlement, leaching, and erosion.

Qualified specialist: A person, by reason of training or experience, who possesses expertise in a particular field or scientific study (e.g., geomorphologist, seismologist, or chemist).

Radionuclide inventory: The isotopic distribution of radioactive materials by waste class, wasteform, and waste container disposed of in the facility and potentially available for release to the environment.

Receptor: The exposed individual relative to the exposure pathway considered.

Receptor behaviors and activities*:* The FEPs associated with the behaviors and activities of the people who may be exposed to radiation.

Reviewer: This document uses the term “reviewer” to include NRC staff reviewers as well as Agreement State reviewers.

Risk: The combined answer to the three questions that consider: (1) what can go wrong; (2) how likely it is; and (3) what the consequences might be. In the context of radioactive waste disposal risk refers to probability-weighted radiological doses.

Safety assessment: A systematic analysis of the ability of the site and design to provide the safety functions and meet technical requirements.

Safety case: A high-level evaluation of the information and analyses that support the licensee’s demonstration that the land disposal facility will be constructed and operated safely. The safety case, which is a component of the application, provides a summary of the safety basis that the disposal site will be capable of isolating waste and limiting releases to the environment; describes the strength and reliability of the technical analyses; and includes consideration of defense-in-depth protections and safety relevant aspects of the site, the facility design, and the managerial, engineering, regulatory, and institutional controls.

Safety function: Defined as a function through which a barrier component of the disposal system contributes to safety and achieves its safety objective throughout the analyses timeframe. Some types of layers such as controls and personnel can also act as safety functions before and during the institutional control period.

Scenario: A possible development of processes, sequence of events, or both, and their effect on features or components of the disposal system and, subsequently, on humans. Types of scenarios used in this guidance document are defined below:

- ***Site scenario*** *(for 10 CFR 61.41)*: A type of postclosure scenario composed of features, events, and processes that exist, or will exist, at the disposal site and their interactions with one another. Used to identify the probable future evolution of the disposal site and its future state in which a receptor carries out activities. The central scenario is used to determine if the performance objective for 10 CFR 61.41 has been met unless dose assessment results for a receptor in a less likely, but plausible, alternative site scenario produce higher dose rates.

  - ***Central scenario*** *(also referred to as a "central site scenario")*: A site scenario that the licensee can best support as to the probable future evolution of the disposal site. As a result of the site selection process for low-level waste disposal, the central scenario generally will not include disruptive events but will include disruptive processes.

  - ***Alternative site scenario***: A site scenario that is not as probable as the central scenario, but still a plausible, future evolution of the disposal site and its future state in which a receptor could carry out activities. Alternative site scenarios may include disruptive events if those features, events, and processes are relevant at the disposal site.

- ***Intruder scenario****, or inadvertent intruder scenario (for 10 CFR 61.42)*: A type of stylized scenario composed of one or more exposure pathways and features, events, and processes associated with the behaviors and activities of the humans who inadvertently intrude. The future state of the postclosure disposal site in which an inadvertent intrusion occurs is the central scenario. Intruder scenarios are used to determine if the performance objective for 10 CFR 61.42 has been met.

- ***Accident scenario*** *(for 10 CFR 61.43)*: A type of stylized scenario composed of accidents, disruptive events, and abnormal occurrences, involving the failure of the encapsulation media during disposal operations and result in release of radionuclides and exposure to onsite receptors. Accident scenarios are used to determine if the performance objective for 10 CFR 61.43 has been met. Accidents can include fires, disruptive events during operations, drops, construction/installation defects, earthquakes, tornadoes, etc.

Scenario development *(also referred to as "site scenario development")*: The process of incorporating a site's current and future features, events, processes, and their interactions into a site scenario. Frequently, a top-down or bottom-up approach is used, or a mixture of the two.

Scenario uncertainty *(also referred to as "site scenario uncertainty")*: Uncertainty about the future state or environment of the site due to the inherent lack of knowledge about how the site will evolve in time.

Sensitivity analysis: An examination of how the behavior of a system varies with change, usually in the values of the governing parameters. An analysis to investigate the dependencies of the result of the assessment on the alternative input elements (i.e., data, assumption, etc.).

Significant quantities: An amount (volume or mass) and concentration of long-lived radionuclides accepted for disposal after the [effective date of the revisions to 10 CFR Part 61]

that could, if released, result in the performance objectives of subpart C of this part not being met.

Site characterization: Studies that enable the licensee to sufficiently describe the conditions of the site to evaluate the acceptability of the decommissioning plan.

Site closure and stabilization: Those actions that are taken upon completion of operations that prepare the disposal site for custodial care and that ensure, to the extent practical, that the disposal site will remain stable and will not need ongoing active maintenance.

Site scenario: See *Scenario*.

Site stability assessment: An assessment used to demonstrate compliance with 10 CFR 61.44 by providing reasonable assurance that long-term stability of the disposal site can be ensured and that maintenance following site closure will not be needed. Long-term stability of the disposal site includes the ability of the site to maintain structural stability from within the disposal units and to maintain stability of the surface of the site. A site stability assessment is tailored to the types of waste disposed, thereby influencing the areal extent of the assessment. In addition, such analyses consider the potential effects of degradation of mechanical properties of containers or other stabilizing man-made features. Stability analyses may be design-based or model-based and may or may not be based on risk considerations.

Solubility limit: The maximum amount of a radionuclide (solute) that can be dissolved per unit of liquid (solvent) under specified conditions (e.g., temperature, pH).

Source term: A conceptual representation of the radionuclide inventory in a disposal site. The quantity of radionuclides expected to be released over time out of a clearly identified boundary (such as the wasteform, container, disposal unit, or facility).

Specialized land disposal facility: A land disposal facility that is designed to prevent an inadvertent intruder from disrupting or contacting emplaced waste while it remains a radiological hazard. A specialized land disposal facility is not a near-surface disposal facility.

Stability: The capability of the disposal site (e.g., wasteform, disposal containers, and disposal units) to maintain its shape and properties to an extent that will not prohibit the demonstration that the disposal site will meet the performance objectives in 10 CFR 61.41 and 10 CFR 61.42 of this part and will, to the extent practical, eliminate the need for active maintenance after site closure and for maintenance in any form after license termination.

Surveillance: Observation of the disposal site for purposes of visual detection of need for maintenance, custodial care, evidence of intrusion, and compliance with other license and regulatory requirements.

System description: A description of the characteristics and interactions, including features and phenomena, of the disposal site and surrounding area to ensure information used to develop the technical analyses and describing the overall disposal system performance have been adequately described.

Technical analyses: The analyses described in 10 CFR 61.13 and includes the performance assessment, the intruder assessment, the operational safety assessment, and the site stability

assessment, in addition to, under certain circumstances, the performance period analyses, needed to demonstrate compliance with the performance objectives of subpart C of 10 CFR Part 61.

Total effective dose equivalent (TEDE): The sum of the deep-dose equivalent (for external exposures) and the committed effective dose equivalent (CEDE) (for internal exposures) (see 10 CFR 20.1003).

Upscaling: The modification of data for use at a different scale. Most commonly upscaling transforms data from fine-scale observations for use at a coarser scale.

Uncertainty analysis: A method of formally assessing, reducing, managing, and documenting the inherent uncertainties of a system. The uncertainties include uncertainty about the future of the site, model uncertainty (which spans conceptual model uncertainty and mathematical model uncertainty), and parameter uncertainty (i.e., uncertainty in values used in the numerical model).

Validation (model): The process of determining the degree to which a model is an accurate representation of the real world from the perspective of the intended uses of the model.

Validation (software): Verification that the governing equations accurately describe the physical processes that occur.

Verification (software): Comparison of the numerical solution generated by the computational model with one or more analytical solutions or with other numerical solutions. Verification of the code ensures that the computer program accurately solves the equations that constitute the mathematical model.

Waste acceptance criteria: The requirements developed through technical analyses or other methods to ensure that waste disposed in a facility will meet the established performance objectives set forth in 10 CFR Part 61. The criteria include allowable activities and concentrations of specific radionuclides, acceptable wasteform characteristics and container specifications, and restrictions or prohibitions on waste, materials, or containers that might affect the facility's ability to meet the performance objectives in Subpart C.

Waste incidental to reprocessing (WIR): Wastes that are incidental to the reprocessing of nuclear fuel that can be managed as LLW.

Waste stream: The origin of a LLW type or combination of waste types with a particular radionuclide content and distribution independent of its physical characteristics.

Waste type: Radioactive materials such as cloth, wood, plastic, glass, or metal, or other substances obtained from radioactive waste treatment systems, industrial processes, or research experiments. Some examples of waste types are dry solids, dry active waste, ion exchange resins, sorbed liquids, filter cartridges, and activated metals.

Wasteform: Radioactive waste in its physical and chemical form including any stabilizing or encapsulating material within which it is incorporated.

APPENDIX A
COMPARISON OF APPLICABLE PERFORMANCE OBJECTIVES

	10 CFR Part 61 regulations for Sites meeting Criteria in 10 CFR 61.1(b)	**Current (Revised) 10 CFR Part 61 Regulations**
Protection of the general population from releases of radioactivity (10 CFR 61.41)	- 10 CFR 61.41c - Pathway analysis - Undefined period of performance - 0.25 millisievert (mSv) (25 millirem (mrem)) annual whole body dose limit for the protection of the general population from releases of radioactivity	- 10 CFR 61.41(a) and (b) - Performance assessment that estimates peak annual dose within the compliance period - 0.25 mSv (25 mrem) annual dose limit for the protection of the general population from the releases of radioactivity that occurs within the compliance period - Performance period analyses that only apply for disposal sites containing significant quantities of long-lived radionuclides and that demonstrate how the disposal site has been designed to reduce long-term releases
Protection of individual from inadvertent intrusion (10 CFR 61.42)	- 10 CFR 61.42(c) - Comply with 10 CFR 61.55 low-level waste classification and segregation requirements - Provide adequate barriers to protect against inadvertent intrusion - Undefined compliance period - No annual dose limit	- 10 CFR 61.42(a) and (b) - Inadvertent intruder assessment that estimates peak annual dose that occurs within the compliance period - 5 mSv (500 mrem) annual dose limit - Analyses that demonstrate exposures will be reduced to the extent reasonably achievable for the protection of inadvertent intruders beyond the compliance period. - Performance period analyses that only apply for disposal sites containing significant quantities of long-lived radionuclides and that demonstrate how the disposal site has been designed to reduce long-term exposures to an inadvertent intruder
Protection of individuals during operations (10 CFR 61.43)	- 10 CFR 61.43 - 0.25 mSv (25 mrem) annual dose limit for the protection of the general population from the releases of radioactivity that occurs during operations - Operational safety assessment with content based on hazard of waste	

	10 CFR Part 61 regulations for Sites meeting Criteria in 10 CFR 61.1(b)	Current (Revised) 10 CFR Part 61 Regulations
Analyses of long-term stability of the disposal site after closure (10 CFR 61.44)	- 10 CFR 61.44 - Site stability assessment that demonstrates that long-term stability of the disposal site can be ensured - Demonstration that there will not be a need for ongoing active maintenance of the disposal site during the institutional control period	

APPENDIX B
HAZARD MAPS

The U.S. Nuclear Regulatory Commission (NRC) staff created hazard maps related to the features and phenomena addressed in the criteria in Title 10 of the *Code of Federal Regulations* (10 CFR) 61.50, "Disposal site suitability requirements for land disposal." The hazard maps presented in this appendix illustrate the features, events, and processes (FEPs) related to 10 CFR 61.50 site suitability criteria. The maps cannot be displayed in this document at sufficient size to be used to determine whether one of these phenomena would impact any specific location. Rather, the figures provide an illustration of potentially impacted areas.

Regulators should not use the figures to prohibit disposal because the resolution of the maps and the precision and accuracy of the techniques used to generate them may not be sufficient for site-specific evaluations. However, regulators should use the maps to determine when greater review effort and more technical basis should be expected for the applicant's site-specific evaluation. The data used to produce these maps could be applied, through Geographic Information System (GIS) software, to perform screening-level analyses of FEPs.

Preparation of the Hazard Maps:

ArcGIS was used to process the data from the data sources and produce the maps. ERDAS Imagine was used to process image data used for the groundwater depth (Figure B-4) and erosion (Figure B-8) maps.

Figure B-1 – The source is elevation data compiled from various sources and provided by Environmental Systems Research Institute (ESRI). The NRC staff created an indicator plot for areas less than 5 m (16 ft) above the current sea level using data from ESRI (2008b).

Figure B-2 – Figure is based on wetlands land use classes from U.S. Geological Survey (USGS) land use/land cover data (USGS, 2011).

Figure B-3 - Where available, one percent annual chance flood event risk zones (100-year floodplain) from the Federal Emergency Management Agency (FEMA) Digital Flood Insurance Rate Map Database (DFIRM) are shown (FEMA, 2012). When DFIRM data were not available, one percent annual chance flood event risk zones (100-year floodplain) from the FEMA National Flood Insurance Program Q3 Flood Data are shown (FEMA, 1998). When DFIRM or Q3 FEMA data were not available, the source is NRC staff calculations performed on data compiled and provided by ESRI (ESRI, 2008a; ESRI, 2008b). A slope model (a grid where each cell is assigned maximum slope between it and the neighboring cells) was created from a digital elevation model (DEM) of the continental United States. From the slope model, a flow direction grid was generated using the direction of maximum slope out of the cell. From the flow direction grid, a flow accumulation grid was generated based on how many cells lay upstream of the grid cell. Cells with a very low slope that accumulated flow over a certain threshold are displayed as black (prone to flooding), all others are white. On top of this image a hydrology layer was added that shows ponds, lakes, reservoirs, and large rivers as black.

Figure B-5 – The source is hydrology data compiled from various sources and provided by ESRI. The figure is based on the categories provided in the referenced data sources. There

could be other data categories not in the ESRI data source that might be areas of previous flooding (ESRI, 2008a; ESRI, 2008b).

Figure B-6 through B-9 - The data were available for download from the websites listed in the references cited on the figure caption. Thirty-five tiles of data that covered the continental United States were used. Each tile of data contained files representing the extent of glaciation at various times during the Quaternary. There were four general time periods covered in most cases: the Younger Dryas, the Late Weichselian (Wisconsinian), the Early-Middle Weichselian, and the maximum limit of Pleistocene glaciation. For some tiles there were separate files for various features: ice sheets, mountain glaciers, and basin glaciers. Sometimes they were combined into one file with a field in the attribute table which signified which type of feature it was. Sometimes there were separate data files for the work from more than one author. Not all authors studied the entire continental United States. The files representing like datasets were merged and those files were clipped to the boundary of the continental United States plus the Great Lakes. Areas in each file that represent ice covered areas for that time are displayed as black on the map. By stacking the files from various authors and features, the maximum extent of glaciation for the period covered by these files is represented in the figure.

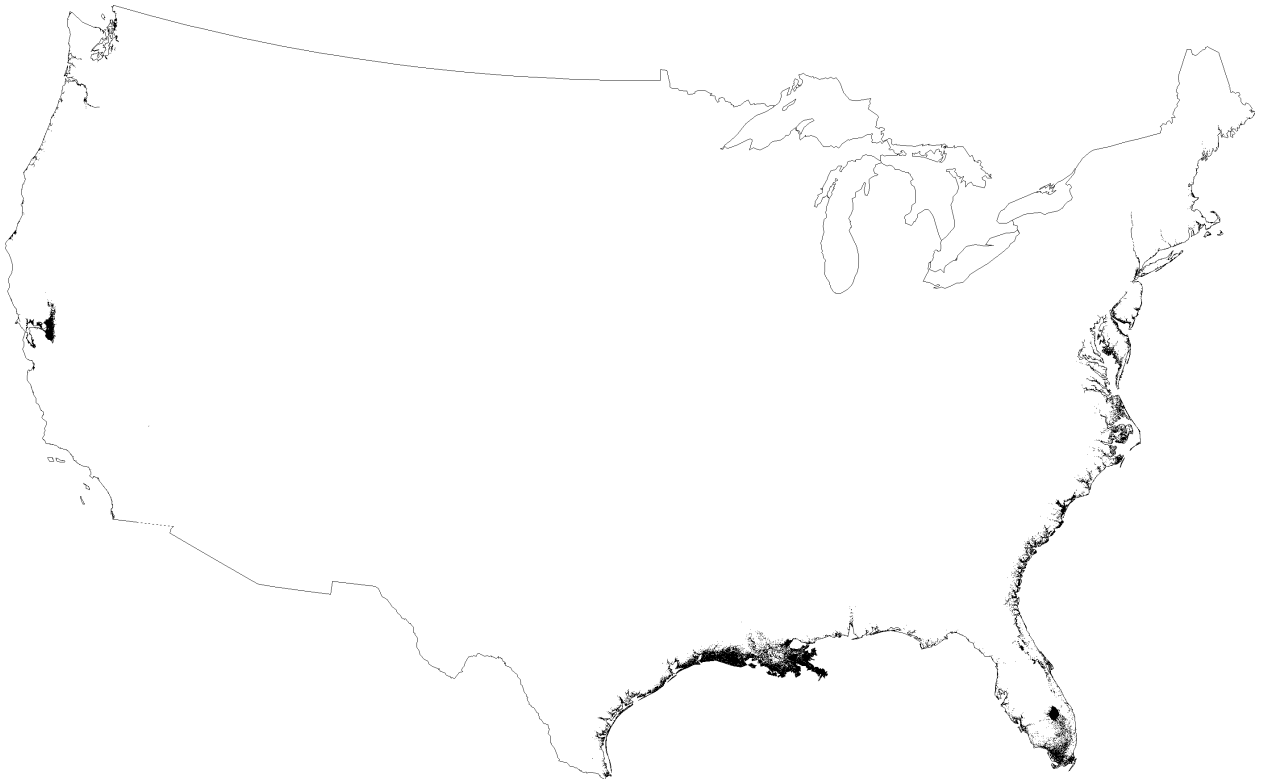


Figure B-1: Approximate area of land that might flood if mean sea level rose by 5 m (16 ft). Proposed sites located near these areas may require additional analysis and evaluation (ESRI, 2008b)

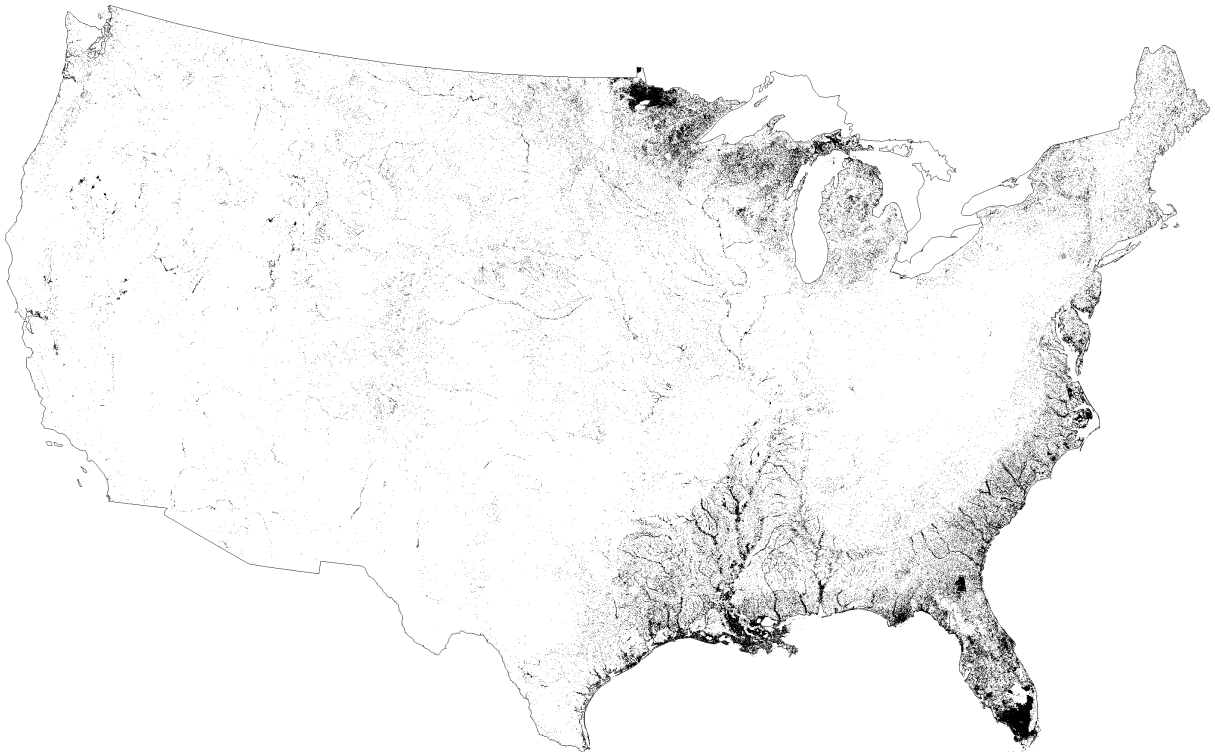


Figure B-2: Approximate area of current wetlands. Proposed sites located near these areas may require additional site characterization and analysis. Wetland areas in the future may change (USGS, 2011)



Figure B-3: Areas of potential flooding that may require additional site characterization and analysis (FEMA, 2012; FEMA, 1998; ESRI, 2008a; ESRI, 2008b)

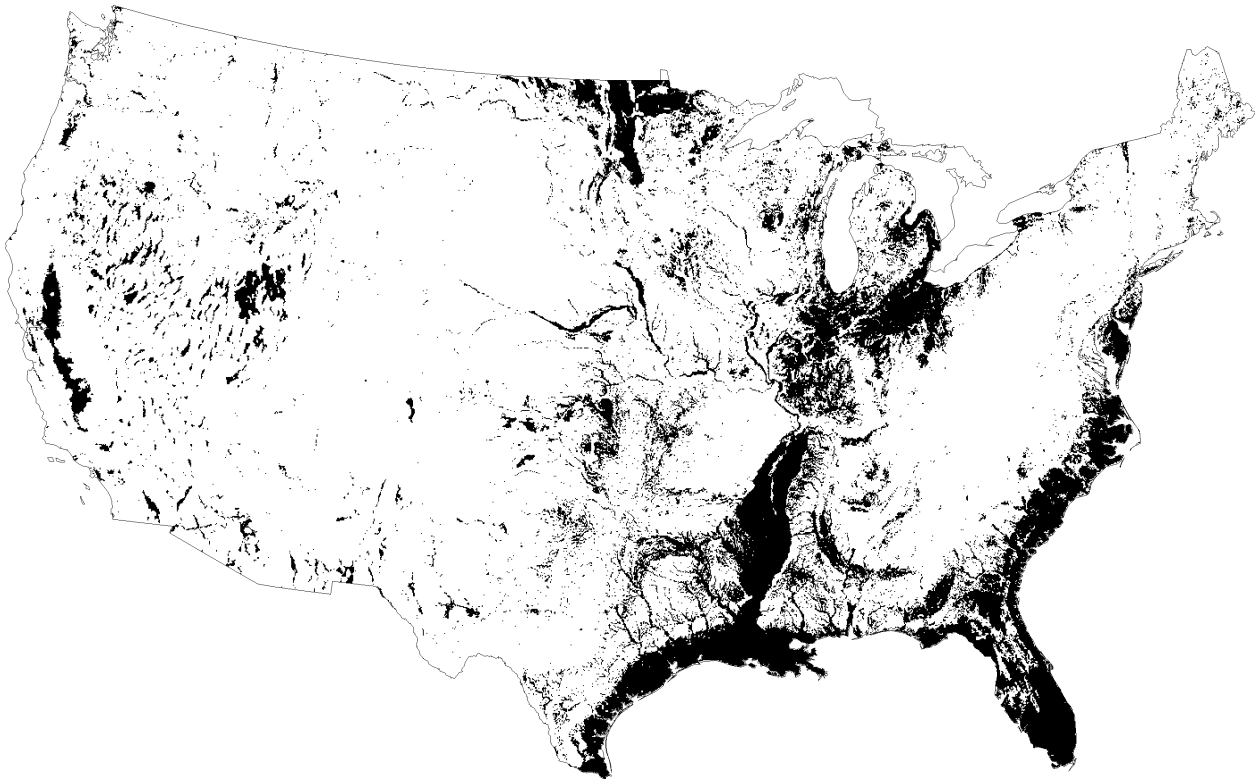


Figure B-4: Approximate area of simulated current water tables shallower than 30 m (100 ft). Proposed sites located in these areas may require additional site characterization and analysis. Future near-surface groundwater areas may change (Kreakie et al, 2012)



Figure B-5: Approximate areas that may have frequently flooded in the past (e.g., dry lake beds, salt flats, areas below sea level). Proposed sites located in these areas may require additional site characterization and analysis (ESRI, 2008a; ESRI, 2008b)

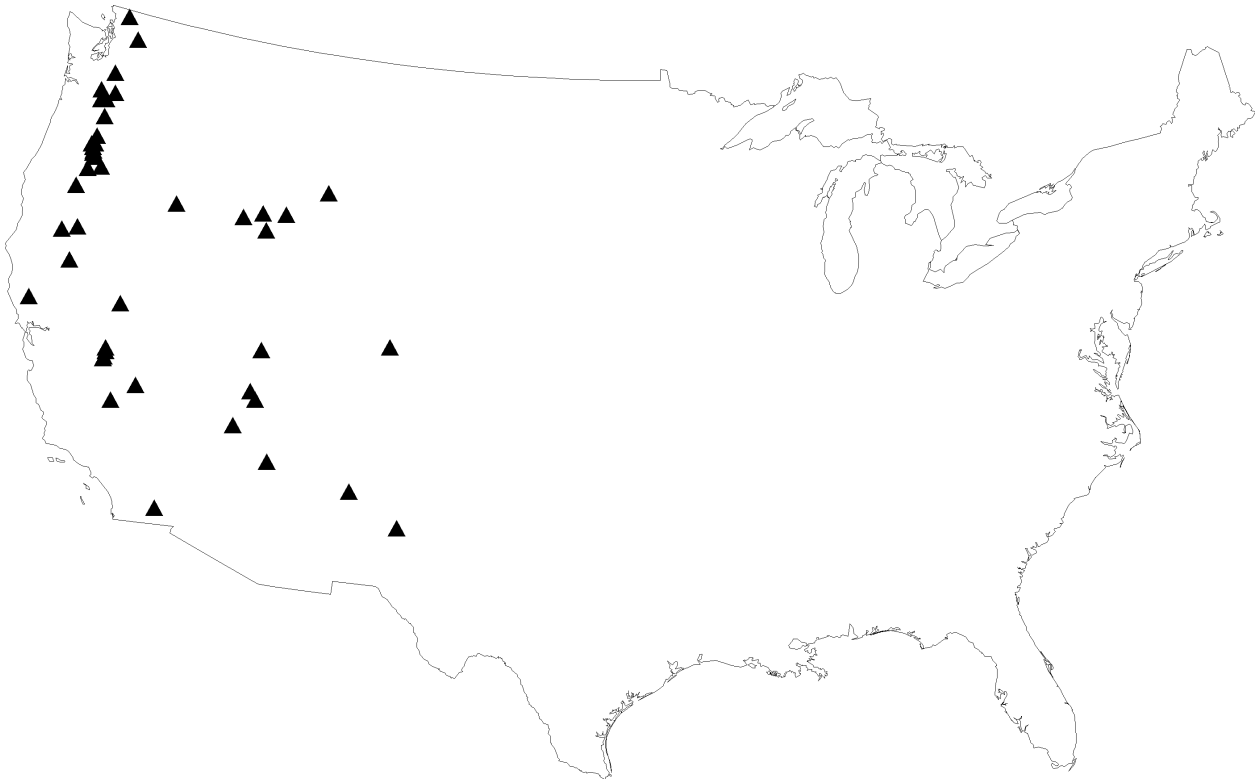


Figure B-6: Approximate locations of Holocene volcanic activity. Proposed sites located near these locations may require additional analysis and evaluation (Siebert and Simkin, 2002)

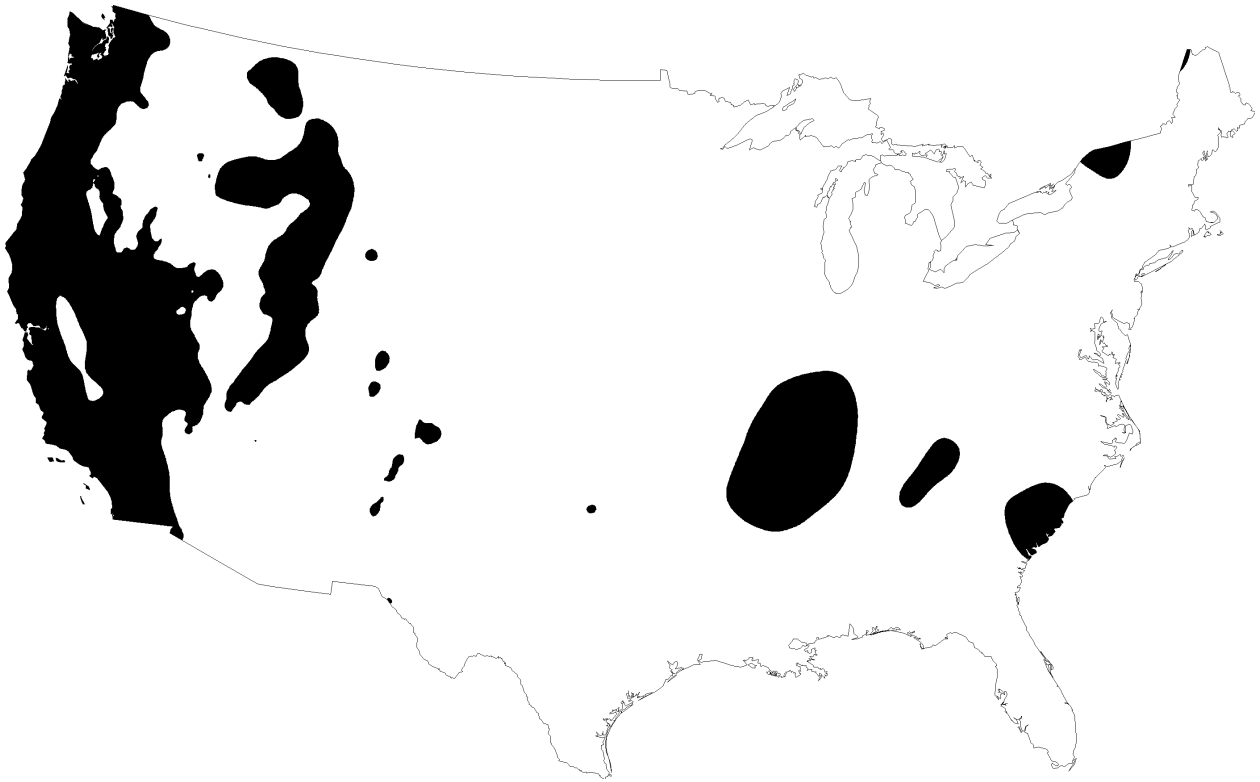


Figure B-7: Current approximate areas of higher potential seismic hazard. Proposed sites located near these areas may require additional analysis and evaluation (Petersen et al, 2011)

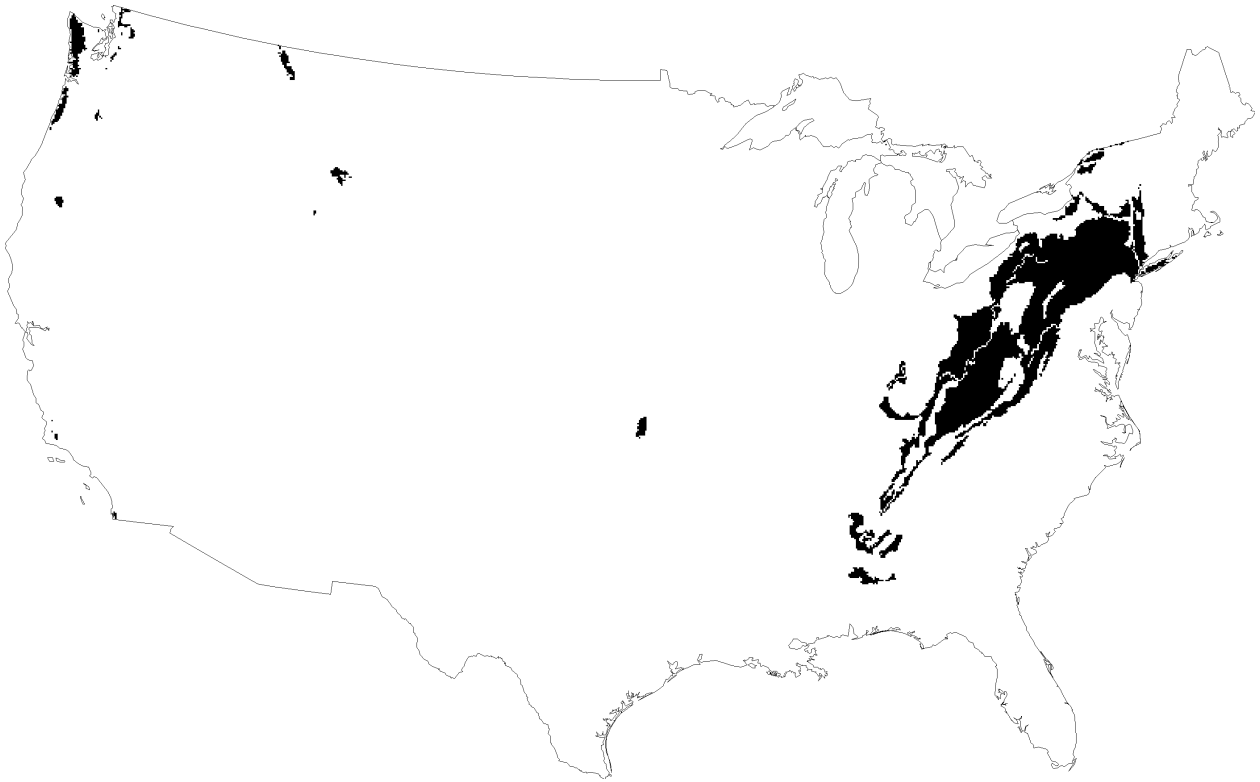


Figure B-8: Current approximate areas of higher vulnerability to water erosion. Proposed sites located in these areas may require additional site characterization and analysis. Areas of high vulnerability to water erosion in the future may change (USDA, 1988)



Figure B-9: Approximate area covered by glaciers during the last three glacial periods of the current Quaternary ice age, i.e., the Wisconsin, Illinoisan, and Pre- Illinoisan glacial periods. Glaciers can cause very disruptive surface geologic processes and potential sites located in areas created by previous glacial processes could require additional analysis and careful evaluation (Ehlers et al., eds., 2011)

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APPENDIX C
GENERIC FEATURES, EVENTS, AND PROCESSES LIST FOR NEAR-SURFACE DISPOSAL OF LOW-LEVEL RADIOACTIVE WASTE

The U.S. Nuclear Regulatory Commission (NRC) staff has developed a list of generic features, events, and processes (FEPs) that can be used by reviewers, applicants, licensees, and other interested stakeholders involved with preparation and review of technical analyses conducted to support licensing of a near-surface disposal facility for low-level radioactive waste (LLW). The NRC staff consulted numerous references to develop (1) a comprehensive list of FEPs and (2) a smaller set of FEPs (or "starter list") that should be analyzed and screened based on requirements in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 61, "Licensing Requirements for Land Disposal of Radioactive Waste." The smaller list is usually considered to be essential to development of a performance assessment (PA). The starter FEP list is not a complete FEP list. Rather, it is a core list of FEPs that an analyst should consider when developing an LLW PA. There are various methods of screening FEPs, as described in Chapter 2.0 of the main document. The technical analysis needed to justify exclusion of a required FEP or to evaluate the impact of an included (or "screened in") FEP (or set of FEPs) can vary. The effort expended on disposition or evaluation of the FEP(s) should be commensurate with the FEP's expected risk-significance.

A complete set of site-specific FEPs can only be developed based on an adequate understanding of the disposal system. Understanding is developed through site characterization and review of facility designs. The development and evaluation of FEPs follows an iterative process, beginning first with evaluation of a central scenario that incorporates the key FEPs that best represent the dynamic system being studied. Following initial analyses, alternative site scenarios that might include potential disruptive events should be considered to ensure that potential vulnerabilities in disposal system performance are identified and adequately addressed. As-emplaced conditions (including final engineered barrier configurations and waste allocation), monitoring, and other data developed following initial PA preparation should be considered to ensure that the results of the PA adequately assess the risk of the disposal facility. Table C-1 gives the starter list of FEPs that should be addressed in LLW PAs.

Table C-1 Starter List of FEPs for Near-Surface Disposal in the United States

Type	FEP
Feature	Radiological inventory
	Waste inventory
	Wasteform (e.g., design, properties, characteristics)
	Waste container
	Free liquids
	Colloids
	Backfill
	Disposal unit
	Disposal site
	Buffer zone
	Engineered barriers (e.g., intruder barriers, engineered cover):
	Presence of agents in waste that may increase mobility (e.g., chelating agents) or lead to degradation of engineered barriers (e.g., corrosive agents)
	Material defects
	Geologic units and materials: Surface soils and sediments Stratigraphy and lithology Hydrogeologic units
	Surface water
	Preferential pathways (anthropogenic, natural)
	Perched water
	Wetlands
	Biosphere: Humans Ecology Flora and fauna (including insects)
	Receptors – surrounding population
	Exposure pathways
	Land use
	Institutional control
	Natural resources
Process	Climate and meteorology
	Natural climate cycling (e.g., glaciation)
	Radioactive decay and in-growth
	Waste interactions
	Gas generation
	Radon emanation
	Waste release (e.g., leaching, dissolution)
	Pyrophoricity
	Criticality
	Degradation: Corrosion – all forms Creep Fatigue

Table C-1 Starter List of FEPs for Near-Surface Disposal in the United States

Type	FEP
	Abrasion (wind, water) Temperature cycling or extremes Freeze thaw cycling Frost action Wet/dry cycling Salt action Oxidation Acid attack Weathering Fracturing (via mechanical or chemical/reaction) Cementitious material degradation Geosynthetic degradation processes Plugging of drainage layers Degradation of clays (e.g., desiccation) Polymer degradation Seismic-induced degradation
	Internal degradation processes, to waste and disposal units: Biodegradation Reaction – incompatible materials Thermodynamic instability Excessive void space – subsidence Radiation damage Dissolution Internal stress generation
	External degradation processes, interaction of system with environment
	Geochemical evolution (disposal unit, disposal site, surrounding environment)
	Geochemistry – speciation, solubility, sorption, etc.
	Interactions of environment and disposal system
	Biological driven release (plant uptake, burrowing animals, bioturbation, etc.)
	Ecological succession
	Water balance processes (e.g., evapotranspiration, runoff, recharge)
	Infiltration
	Near-field flow processes (e.g., flow bypassing, film flow)
	Episodic flow
	Groundwater flow: Advection Dispersion Water table fluctuation Discharge to surface Matrix diffusion Dilution Diffusion Density-driven flow Capillary rise
	Erosion
	Deposition

Table C-1 Starter List of FEPs for Near-Surface Disposal in the United States

Type	FEP
	Instability
	Geomorphology
	Surface geologic processes – mass wasting, subsidence, slope failure, etc.
	Dynamic change to geology (e.g., sinkhole formation)
	Loading and differential settlement
	Pedogenesis
	Tectonic processes
	Groundwater transport
	Gas transport
	Soil transport (fluvial, Aeolian)
	Colloid transport
	Dose assessment processes (e.g., drinking water consumption, soil build-up, resuspension)
	Dynamic processes (e.g., natural temporal variability, seasonal effects, episodic changes, barometric pumping)
Event	Explosion
	Fire
	Inadvertent human intrusion (habitation, drilling, resource exploration)
	Accident – operational or external
	Dam failure
	Aircraft crash
	Tectonic events: Seismic (including earthquakes) Volcanic Tsunami
	Tornado
	Hurricane

As discussed above, the NRC staff developed a comprehensive list of generic FEPs appropriate for near-surface disposal of LLW for use by reviewers, applicants, licensees, and other interested stakeholders. The staff consulted numerous references during development of this list (NEA, 2000; NEA, 2006; IAEA, 2003, BIOMOVS II, 1996; Arlt, 2013; Neptune, 2011; SRS, 2012a; SRS, 2012b). Some of the reference FEPs are not expected to be risk-significant during the time periods of interest for disposal of most LLW or are considered outside the scope of the 10 CFR Part 61 regulatory framework. An effort was made to identify classes of these FEPs. For those FEP sources that screened FEPs in or out (e.g., BIOMOVS and site-specific FEP lists), the staff considered the rationale for exclusion of the FEP in determining whether to list the FEP in the list of generic FEPs. It is important to note that although a project-specific FEP may have been "screened out," if the FEP was considered potentially applicable to near-surface LLW disposal, the staff included it in the generic FEP list. Although screening approaches and the results of screening are summarized below, it is important to note that the staff did not evaluate the adequacy of FEP screening processes or the completeness of project-specific FEP lists.

The generic FEP list is structured after the International Atomic Energy Agency (IAEA) Improvement of Safety Assessment Methodologies for Near Surface Disposal Facilities (ISAM)

and Nuclear Energy Agency (NEA) FEP lists described in more detail below. FEPs that are unlikely during the compliance period but may become increasingly more likely over time are flagged as "long-term" FEPs. If a FEP is designated a "long-term" FEP, licensees should consider the applicability of the FEP for a specific site when performance period analyses are required. The generic FEP list may not explicitly include some more detailed FEPs from the reference sources, but those may need to be considered unless clearly linked to a category of FEPs that are not considered important for near-surface waste disposal. Reference sources used to create the generic FEPs list include databases developed by international organizations as well as several site-specific applications for low- and high-level radioactive waste disposal facilities (or repositories) in the United States and Europe[1]. A brief description of each of these data sources is provided.

BIOMOVS II

A structured, generic biosphere FEP list was developed by the Biospheric Model Validation Study Phase II (BIOMOVS II) Reference Biospheres Working Group. BIOMOVS is an international study to test models designed to predict the environmental transfer and bioaccumulation of radionuclides and other trace substances. The BIOMOVS II FEP list was developed specifically for application to the calculation of annual individual doses from the long-term release of radionuclides to groundwater. The list is relevant to a wide range of assessments. The developers recognized that the list may not include sufficient detail for a specific project or assessment. Additionally, definitions may not be universally applicable. The structure of the FEP list is provided in Figure C-1. The expanded FEP list includes approximately 140 FEPs.

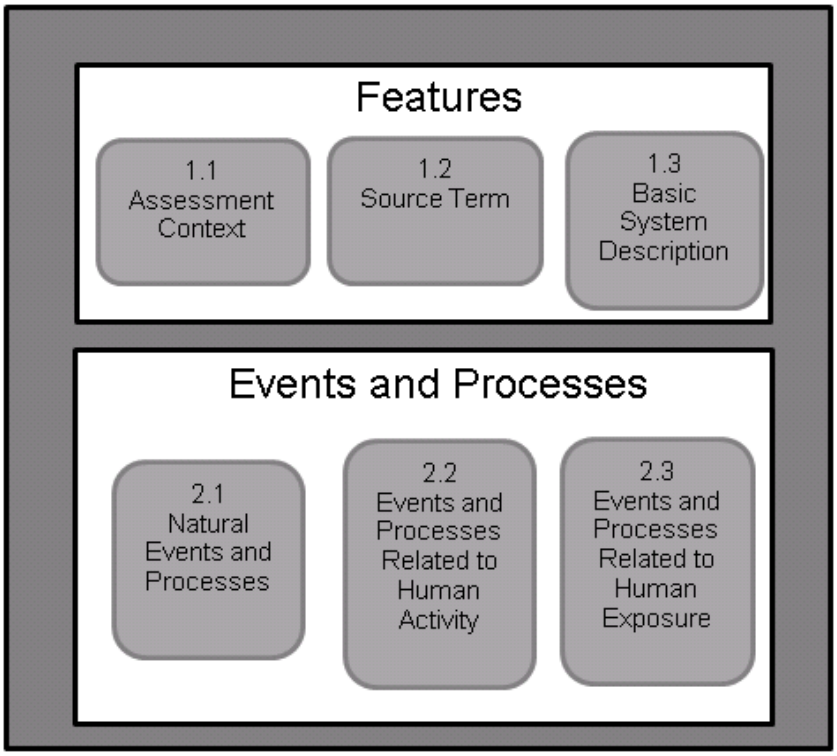


Figure C-1 Structure of BIOMOVS FEP List

[1] Several domestic FEP lists incorporate FEPs from European assessments.

Nuclear Energy Agency

The NEA developed an international FEP list (NEA, 2000) relevant to the postclosure safety of repositories for radioactive waste disposal. The NEA FEP list was intended to (1) give FEPs to be considered when determining the scope of a new assessment, (2) provide FEPs against which completed assessments could be audited or reviewed, and (3) give confidence in the comprehensiveness of a completed assessment. The NEA FEP list includes 134 FEPs.

A database consisting of two parts was created to facilitate use of the international FEP list:

1. International FEP List—the structured list of factors, or FEPs, relevant to the assessment of the long-term safety of nuclear waste repositories

2. Project Databases—a collection of FEP lists and databases from specific project studies, along with their references

The database was developed by the NEA FEP Database Working Group that included representatives from seven Organisation for Economic Cooperation and Development (OECD)/NEA countries. Version 2.1 of the database contains over 1650 project specific FEPs[2] from 10 projects (NEA, 2006). Two additional project-specific FEP lists were added: (1) SCK-CEN Catalogue of Events, Features and Processes for the Mol Site in Belgium and (2) Encyclopedia of FEPs for the Swedish SFR and Spent Fuel Repositories. Additional details on the project-specific FEP lists, and additional functionality, were added to the database.

The NEA international FEP list is a generic, high-level FEP list from which project-specific FEPs could be developed or categorized. Project-specific FEPs available in the database are cross-walked to the categorical FEPs comprising the international FEP list. Figure C-2 shows the major categories of the NEA international FEP list. The international FEP list has four layers with the three inner layers in figure C-2 further subdivided into additional categories. Figure C-2 does not list individual FEPs for each subcategory.

International Atomic Energy Agency Improvement of Safety Assessment Methodologies

In 1997, the IAEA launched ISAM, a coordinated research project to critically evaluate, enhance, and provide confidence in the approaches and tools used for postclosure safety assessments of near-surface radioactive waste disposal facilities. As part of the ISAM project, the NEA international FEP list was modified for near surface disposal facilities. For example, some of the NEA FEP definitions and comments associated with the FEPs were altered to be more representative of near surface conditions. The ISAM FEP list was intended to be a user-friendly, comprehensive, initial list from which FEPs applicable to a specific site could be developed. Because the NEA list was extensively reviewed for completeness for geologic systems and the ISAM FEP list was based on the NEA FEP list, the developers of the ISAM FEP list reasoned that users should have confidence in the comprehensiveness of the ISAM FEP list. The hierarchy of the ISAM FEP list is similar to that of the NEA international FEP list in Figure C-2[3].

[2] These project FEPs are not unique, and many of them overlap.

[3] This is true with the exception that “1.5 Other External Factors” is not included in the ISAM FEP list.

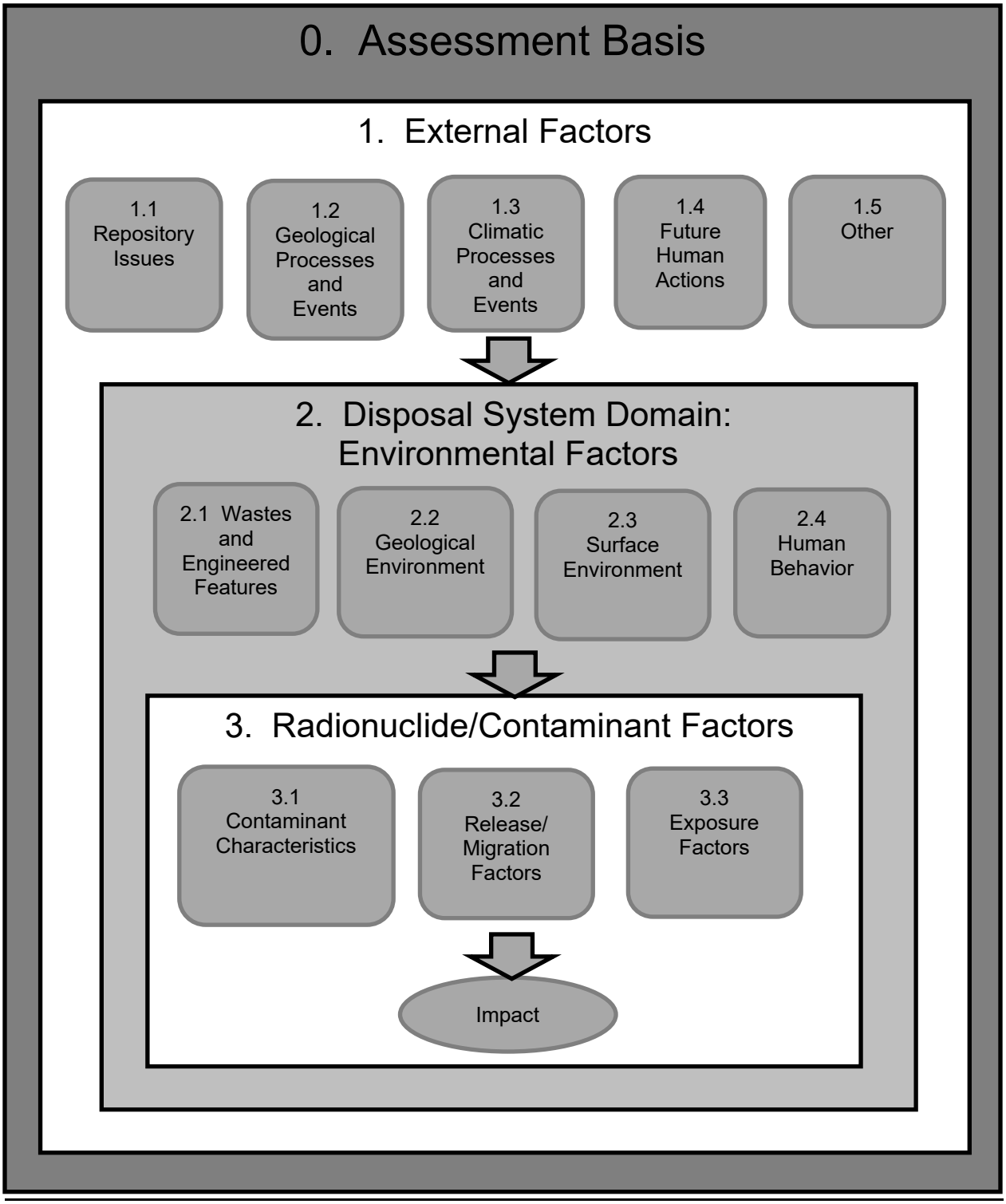


Figure C-2 Structure of the NEA FEP list

BIOMASS

Changes such as the following were made to the organization of the BIOMOVS II list to develop the BIOMASS FEP list:

- providing a clearer distinction between FEPs related to basic elements of the assessment context and those related to the biosphere system, radionuclide transport, and radiation exposure
- expressing the intrinsic phenomena relating to the biosphere system in terms of system characteristics rather than the behavior of radionuclides within the system
- incorporating experience gained with the application of the reference biosphere methodology since BIOMOVS II, which helped to amplify certain details of the original list and led to the incorporation of additional FEPs

U.S. Department of Energy Hanford Site

The U.S. Department of Energy (DOE) has developed PAs to support the closure of underground storage tanks containing radioactive waste resulting from nuclear weapons production activities. A total of 177 underground tanks are located in the 200 Area of the site and used to store reprocessed, liquid high-level waste (HLW) from reactor operations and other site activities. The DOE has initiated the process of retrieving, treating, and disposing of radioactive mixed waste from 149 underground single-shell tanks that do not have secondary containment. DOE Hanford prepared a PA to support closure of these tanks. In 2009, DOE Hanford initiated a scoping process to assist with updating the PA to support tank closure. An extensive FEP list was developed as part of this scoping process. Due to lack of funding, the scoping process was curtailed; however, the NRC staff considered the draft list of FEPs developed during Hanford PA scoping in developing this guidance document.

Clive, Utah Site

Energy*Solutions* operates an LLW disposal facility in Clive, Utah. Energy*Solutions* prepared a PA to evaluate the risk of disposal of depleted uranium (DU) waste. To support PA development, Energy*Solutions* developed a list of FEPs. Neptune and Company, Inc. documented and examined FEPs that may apply to the disposal of DU waste at the Clive facility (Neptune, 2011). The identification of FEPs for use in the Clive facility PA was an iterative process that began with compilation of an exhaustive list of candidate FEPs that could affect the long-term performance of the LLW disposal facility. Table C-2 lists the reference sources considered in developing the initial list of FEPs. All potentially relevant FEPs were collected from a variety of reference sources (e.g., the Yucca Mountain Project, the Waste Isolation Pilot Plant (WIPP), and several foreign radioactive waste projects). The initial list from external sources was modified as additional FEPs were identified that are specific to the Clive facility. Approximately 980 FEPs were identified.

This compilation of FEPs led to significant redundancy because some FEPs were included in numerous original sources. Redundancy was addressed by modifying the candidate list of FEPs through normalization (i.e., removal of redundant FEPs) and assignment of FEP categories (i.e., groupings of common FEPs). This consolidation process reduced the total number to 135 unique FEP groupings that were binned into 18 major categories.

Of the 135 FEP groupings, 67[4] were retained for further consideration and 68[5] were dismissed from inclusion in the PA model. All FEP groupings considered and retained for inclusion in the conceptual site model and site scenarios are reported in Table C-3 (see unshaded FEPs). FEPs that were dismissed from consideration in the PA include those that do not fall within the scope of the PA or were characterized as extremely unlikely to occur or have a low consequence if they were to occur. Some FEPs were dismissed based on site-specific considerations. FEP groupings that were excluded from the PA are listed in Table C-3 (see grey-shaded FEPs).

U.S. Department of Energy Savannah River Site

Operation at the DOE Savannah River Site (SRS) began in 1951. The primary use for the site was the production of nuclear material for national defense. Between 1954 and 1986, the DOE generated significant quantities of radioactive waste from the reprocessing of spent nuclear fuel and to a lesser extent from the production of targets for nuclear weapons. This waste was stored in 51 underground tanks located in two tank farms. The DOE plans to clean the tanks and stabilize the waste residuals in a cementitious wasteform. The DOE also plans to dispose of relatively low-activity salt waste retrieved from the tanks in the saltstone disposal facility. The DOE has prepared PAs to demonstrate that the stabilized waste remaining in the tank farms and the waste disposed of in the saltstone disposal facility can meet performance objectives for LLW disposal. The DOE prepared an *ex post facto* FEPs analysis to provide support for its compliance demonstrations.

The initial SRS FEP list included 245 FEPs drawn from five different reference sources listed in Table C-4. The initial 245 FEPs were then binned into the six categories listed in Table C-5.

In addition to consolidation of the five FEP list reference sources, the DOE included 17 additional FEPs evaluated in the SRS PAs leading to a total of 262 FEPs. Next, the list of 262 FEPs was screened.[6] The SRS FEP screening team performed screening in two phases. During the first phase, team members independently applied the FEP screening criteria[7] using a survey. The independent survey results were collected, and a subset of FEPs was "screened in" or "out" based on the results. Those FEPs with relatively more ambiguous results (i.e., survey results indicated that the FEP should be considered further) were discussed in the second phase. At the start of Phase 2, 142 FEPs remained. The FEPs that were "screened out" in Phases 1 and 2 are listed in Table C-6.

A total of 230 FEPs remained after Phase 2 screening. SRR-CWDA-2012-00022, "Evaluation of Features, Events, and Processes in the F-Area Tank Farm Performance Assessment," Revision 0, issued February 2012 (SRS, 2012b), crosswalks remaining FEPs to the F-Area Tank Farm PA to ensure all relevant FEPs were addressed.

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4 Neptune (2011) indicates that 90 FEP groupings were retained but Table B of the same document only lists 67 FEP groupings.

5 Neptune (2011) indicates that 45 groupings were excluded but Table C of the same document lists 68 FEP groupings.

6 This is true except for 46 programmatic FEPs that would not be subject to screening. These FEPs are listed in Table 4.0-1 of SRR-CWDA-2012-00011, Revision 0.

7 Criteria were based on the perceived probability of occurrence within 10,000 years and the perceived consequence relative to final PA results.

Table C-2 List of Clive Facility FEPs Reference Sources

Project or Facility	Reference[a]
SKI/SKB	Andersson, J., T. Carlsson, T.F. Kautsky, E. Soderman, and S. Wingefors, 1989. "The Joint SKI/SKB Scenario Development Project," SKB-TR8 9-35, Swedish Nuclear Fuel and Waste Management Co., Stockholm, Sweden.
WIPP	Burkholder, H.C., 1980. "Waste Isolation Performance Assessment—A Status Report," in *Scientific Basis for Nuclear Waste Management*, Vol. 2, C.J.M. Northrup, Jr. (ed.), Plenum Press, New York, New York, pp. 689–702.
WIPP	Guzowski, R.V., 1990. "Preliminary Identification of Scenarios That May Affect the Escape and Transport of Radionuclides from the Waste Isolation Pilot Plant, Southeastern New Mexico," SAND89-7149, Sandia National Laboratories, Albuquerque, New Mexico.
Nevada Test Site	Guzowski, R.V., and G. Newman, 1993. "Preliminary Identification of Potentially Disruptive Scenarios at the Greater Confinement Disposal Facility, Area 5 of the Nevada Test Site," SAND93-7100, Sandia National Laboratories, Albuquerque, New Mexico.
Nevada Test Site	Hertzler, C.L., and C.L. Atwood, 1989. "Preliminary Development and Screening of Release Scenarios for Greater Confinement Disposal of Transuranic Waste at the Nevada Test Site," EGG-SARE-8767, EG&G Idaho, Inc., Idaho Falls, Idaho.
Hypothetical Columbia Plateau Repository	Hunter, R.L., 1983. "Preliminary Scenarios for the Release of Radioactive Waste from a Hypothetical Repository in Basalt of the Columbia Plateau," SAND83-1342 (NUREG/CR-3353), Sandia National Laboratories, Albuquerque, New Mexico.
WIPP	Hunter, R.L., 1989. "Events and Processes for Constructing Scenarios for the Release of Transuranic Waste from the Waste Isolation Pilot Plant, Southeastern New Mexico," SAND89-2546, Sandia National Laboratories, Albuquerque, New Mexico.
High-Level Waste Repository	IAEA, 1983. "Concepts and Examples of Safety Analyses for Radioactive Waste Repositories in Continental Geological Formations," Safety Series Report No. 58, IAEA, Vienna, Austria.
High-Level Waste Repository	Koplik, C.M., M.F. Kaplan, and B. Ross, 1982. "The Safety of Repositories for Highly Radioactive Wastes," *Reviews of Modern Physics*, Vol. 54, No. 1, pp. 269–310.
High-Level Waste Repository (Canada)	Merrett, G.J., and P.A. Gillespie, 1983. "Nuclear Fuel Waste Disposal: Long-Term Stability Analysis," AECL-6820, Atomic Energy of Canada Limited, Pinawa, Manitoba, Canada.
NEA Working Group	NEA, 1992. "Systematic Approach to Scenario Development. A report of the NEA Working Group on the Identification and Selection of Scenarios for Performance Assessment of Radioactive Waste Disposal," NEA, Paris, France.
NEA Working Group	NEA, 2000. "Features, Events, and Processes (FEPs) for Geologic Disposal of Radioactive Waste: An International Database," NEA, Paris, France.

[a] References for Andersson et al. (1989), Burkholder (1980), Guzowski (1990), Hertzler and Atwood (1989), Hunter (1983), Hunter, (1989), IAEA (1983), Koplik et al. (1982), Merrett and Gillespie (1983), and NEA (1992) were found in Guzowski and Newman (1993).

Table C-3 Clive Facility FEP Groupings and FEPs

ID	FEP Category	FEP Groupings
1	Model Settings	Model Parameterization
2		Period of Performance
3		Regulatory Requirements
4		Spatial Domain
5	Geological	Diagenesis
6		Gas or Brine Pockets
7		Landslide
8		Local Subsidence
9	Geochemical	Geochemical Effects
10	Hydrogeological	Denudation
11		Erosion
12		Erosional Transport
13		Hydrogeological Effects
14		Sedimentation
15		Subrosion
16	Hydrology	Groundwater Transport
17		Hydrological Effects
18		Inundation
19		Flooding
20		Surface Water Transport
21	Meteorology	Frost Weathering
22		Meteorology
23		Resuspension
24		Atmospheric Dispersion
25		Tornado
26	Marine	Coastal Processes
27		Hurricanes
28		Insolation
29		Marine Effects
30		Tsunami
31	Climate Change	Climate Change
32		Lake Effects
33		Wave Action
34		Glacial Effects
35		Permafrost
36	Tectonic/Seismic/Volcanic	Geophysical Effects
37		Breccia Pipes
38		Diapirism
39		Discontinuities
40		Earthquake
41		Faulting

Table C-3 Clive Facility FEP Groupings and FEPs

ID	FEP Category	FEP Groupings
42		Fracturing
43		Geological Intrusion
44		Hydraulic Fracturing
45		Intrusion Into Accumulation Zone in the Biosphere
46		Isostatic Effects
47		Lava Tubes
48		Orogeny
49		Regional Subsidence
50		Seismic Effects
51		Tectonic Effects
52		Volcanism
53	Celestial	Meteorite Impact
54		Microbial Effects
55	Natural Processes	Radiological Effects
56		Wildfire
57		Ecological Changes
58		Gas Generation
59		Pedogenesis
60	Other Natural Processes	Radioactive Decay and Ingrowth
61		Radon Emanation
62		Reconcentration
63		Anthropogenic Climate Changes
64		Community Development
65		Excavation
66		Explosions
67		Human-Induced Processes
68		Human-Induced Transport
69		Inadvertent Human Intrusion
70		Inhabitation
71		Institutional Control
72		Land Use
73	Human Processes	Postclosure Subsurface Activities
74		Accidents During Operations
75		Climate Control
76		Closure Failure
77		Fire
78		Fisheries
79		Geothermal Energy Production
80		Injection Wells
81		Intentional Intrusion
82		Investigation

Table C-3 Clive Facility FEP Groupings and FEPs

ID	FEP Category	FEP Groupings
83		Irrigation
84		Monitoring
85		Nuclear Testing
86		Operational Effects
87		Operational Error
88		Quality Control
89		Resource Extraction
90		Sabotage
91		Unplanned Events
92		War
93		Waste Recovery
94		Water Resource Management
95		Weapons Testing
96	Containerization	Containment Degradation
97		Corrosion
98	Engineered Features	Compaction Error
99		Engineered Features
100		Material Properties
101		Repository Design
102		Source Release
103		Subsidence of Repository
104		Waste
105	Waste	Nuclear Criticality
106		Other Waste
107	Source Release	Electrochemical Effects
108		Explosions
109	Contaminant Migration	Biotically-Induced Transport
110		Colloid Transport
111		Contaminant Transport
112		Diffusion
113		Dilution
114		Dispersion
115		Dissolution
116		Dust Devils
117		Gas Transport
118		Infiltration
119		Local Geology
120		Preferential Pathways
121		Gas Intrusion
122		Convergence of Opening
123		Design Error

Table C-3 Clive Facility FEP Groupings and FEPs

ID	FEP Category	FEP Groupings
124		Material Defects
125		Mechanical Effects
126		Release of Stored Energy
127		Repository Seals
128	Exposure	Animal Ingestion
129		Dosimetry
130		Exposure Media
131		Human Behavior
132		Human Exposure
133		Ingestion Pathways
134		Inhalation Pathways
135		Agriculture

Table C-4 SRS FEP List Reference Sources

FEPs List	Source Document	Total Number of FEPs
ISAM	"Safety Assessment Methodologies for Near-Surface Disposal Facilities, Results of a Coordinated Research Project," Volume 1, ISBN 92-0-104004-0.	141
UFD	"Features, Events, and Processes for the Disposal of Low-Level Radioactive Waste FY2011 Status Report," FCRD-USED-2011-000297.	449
YMP	"Features, Events, and Processes for the Total System Performance Assessment: Analyses," ANL-WIS-MD-000027, Rev. 0.	374
DGR	"Deep Geologic Repository for OPG's Low and Intermediate Level Waste, Post-Closure Safety Assessment (Volume 1): Features, Events and Processes," NWMODGR-TR-2009-05.	299
SKI	"Encyclopedia of Features, Events and Processes (FEPs) for the Swedish SFR and Spent Fuel Repositories," SKI Report 02:35.	120

Table C-5 SRS FEPs List Categories

Category	Group
1.0 Assessment Basis	1.1 General
	1.2 Regulations and Controls
	1.3 Models and Calculations
	1.4 Other Assessment Factors
2.0 External Factors	2.1 Human Characteristics
	2.2 Land and Water Management
	2.3 Future Human Activity
	2.4 Biological Factors
	2.5 Geologic Features
	2.6 Geologic Processes
	2.7 Climate
	2.8 Water Cycle
3.0 Closure System	3.1 General Closure System
	3.2 Pre-Closure Activities
	3.3 Closure System Components
	3.4 Closure System Hydrology
	3.5 Chemical Processes
	3.6 Thermal Processes
	3.7 Material Degradation
	3.8 Other Closure System Factors
4.0 Contaminant Factors	4.1 Contaminant Description
	4.2 Contaminant Properties
	4.3 Concentrations
	4.4 Exposure Factors
	4.5 Other Contaminant Factors
5.0 Flow and Transport	5.1 Flow Factors
	5.2 Hydraulic Effects on Flow
	5.3 Release and Transport
6.0 Disruptive Events	6.1 Intrusions
	6.2 Seismic Events
	6.3 Igneous Events
	6.4 Other Events

Table C-6 FEPs Excluded from SRS PAs

FEP	ID	Phase
Large Scale Salt Processes (Diapirism, Dissolution, and Creep)	2.6.11	Phase 1
Ashfall	6.3.03	Phase 1
Extraterrestrial Events	6.4.08	Phase 1
Changes in the Earth's Magnetic Field	6.4.09	Phase 1
Changes to Earth's Tidal Processes	6.4.10	Phase 1
Pollution (Soil, Groundwater, Air, etc.)	2.2.07	Phase 2
Ozone Layer Failure	2.3.07	Phase 2
Species Evolution	2.4.05	Phase 2
Stress Regimes	2.5.09	Phase 2
Orogeny	2.6.03	Phase 2
Diagenesis and Pedogenesis	2.6.05	Phase 2
Sedimentation	2.6.06	Phase 2
Creeping of the Rock Mass	2.6.10	Phase 2
Acid Rain	2.7.04	Phase 2
Costs of Construction, Operation, Closure	3.2.04	Phase 2
Chelating Agent Effects	3.5.12	Phase 2
Thermal Processes and Conditions the Engineered System	3.6.01	Phase 2
Thermal Processes and Conditions the Natural System	3.6.02	Phase 2
Thermo-Mechanical Stresses Alter Characteristics of Engineered Barrier System Components	3.6.04	Phase 2
Recrystallization of Vitrified Wastes	3.6.05	Phase 2
Effects of System Heat on the Biosphere	3.6.06	Phase 2
Creep of Metallic Materials in the Engineered System	3.7.04	Phase 2
Oxygen Embrittlement of Engineered System Metals	3.7.05	Phase 2
Localized Interactions Between Emplaced Wastes	4.2.05	Phase 2
Nuclear Criticality	4.5.01	Phase 2
Chemically-Induced Density Effects on Groundwater Flow	5.1.06	Phase 2
Hydrothermal Activity	5.2.02	Phase 2
Seismicity Associated with Igneous Activity	6.2.04	Phase 2
Igneous Intrusion into the Closure Facility	6.3.01	Phase 2
Volcanic Eruptions and Magmatic Activity	6.3.02	Phase 2
Releases Prior to Closure	6.4.01	Phase 2
Impacts from Meteorites or Space Debris	6.4.07	Phase 2
Extraterrestrial Events	6.4.08	Phase 2
Changes in the Earth's Magnetic Field	6.4.09	Phase 2
Changes to Earth's Tidal Processes	6.4.10	Phase 2

Generic FEP List

The ISAM and NEA international FEP list structure illustrated in Figure C-2 was retained in constructing a comprehensive, generic FEP list applicable to near-surface disposal of LLW. Because the ISAM FEP list modified the NEA FEP list to be applicable to near-surface disposal facilities, the ISAM FEP category titles are specifically listed in the generic FEP list. However, some of the ISAM FEPs are not expected to be risk-significant during the time periods of interest for disposal of LLW or are considered outside the scope of the 10 CFR Part 61 regulatory framework. These FEPs that are considered less relevant for near-surface disposal of LLW are listed in Table C-7. The remaining ISAM FEPs represent the major FEP groupings in the generic FEPs list (see column 1 of Tables C-9 and Table C-10).

Table C-8 gives factors associated with the assessment context (e.g., purpose of the assessment), operational factors, and site characterization or monitoring activities that may dictate the types and scope of FEPs considered in a technical assessment and help with the development of project-specific FEPs. The two tables of generic FEPs that follow include (1) FEPs to analyze or screen to construct central and alternative site scenarios (Table C-9) and (2) FEPs to analyze or screen for site-specific intruder scenarios and FEPs relevant to receptor behaviors, receptor activities, and pathway analyses (Table C-10).

For FEPs that may be considered unlikely during the compliance period but that may become more likely over time, the “long-term” column (column 3) is flagged in Tables C-9 and Table C-10 (no assessment context factors are marked “long-term”). If the “long-term” column is flagged, the applicability of the FEPs for a specific site should be considered if performance period analyses are required. Although some FEPs from the reference sources are not explicitly listed in Tables C-7 through C-10, one can assume that the FEP should be considered unless it is clearly linked to a category of FEPs that are deemed not applicable for near-surface waste disposal (listed in Table C-7). Reviewers, applicants, licensees, and other interested stakeholders should find the detail provided in the tables sufficient to understand the scope of FEPs that should be considered for a site, with examples provided in the last column (column 4); however, the generic FEP list should not be considered an exhaustive list that would encompass every potentially applicable FEP that may be important for a site. Likewise, a licensee would not need to consider every FEP listed in Tables C-8 through C-10.

Table C-7 FEPs Less Relevant for Near-Surface Disposal of LLW in the United States

Excluded FEP	ISAM ID	Rationale
Future human action assumptions[1]	0.05	The scope of this FEP is limited. Future societal and technology development will occur but is difficult to predict. Therefore, unnecessary speculation about future human actions should be avoided. The uncertainty associated with future human actions is accounted for in the technical analyses by using reasonably conservative receptor scenarios.
Future human behavior (target group) assumptions[1]	0.06	The scope of this FEP is limited. Future human behavior is difficult to predict. Unnecessary speculation about future human behavior should be avoided.
Retrievability	1.1.13	LLW disposal facilities in the United States are not designed for retrievability.
Orogeny and related tectonic processes at plate boundaries	1.2.01	Orogeny and related tectonic processes at plate boundaries are only expected to be important for near-surface disposal facilities if very long performance periods are evaluated.
Anorogenic and within-plate tectonic processes (Deformation, elastic, plastic, and brittle)	1.2.02	Anorogenic and within-plate tectonic processes are only expected to be important for near-surface disposal facilities if very long performance periods are evaluated.
Metamorphism	1.2.05	Metamorphism is not expected to be important for near-surface disposal facilities or in the timeframes of interest.
Diagenesis[2]	1.2.08	Diagenesis is not expected to be important for near-surface disposal facilities or in the timeframes of interest.
Human influences on climate including ozone depletion, global warming, and greenhouse effect.	1.4.01	Future anthropogenic impacts on climate change are difficult to predict and are covered under natural climate change.
Motivation and knowledge issues (inadvertent/deliberate human actions)	1.4.02	The scope of this FEP is limited. Advertent intruders are not protected under 10 CFR Part 61.

Table C-7 FEPs Less Relevant for Near-Surface Disposal of LLW in the United States

Excluded FEP	ISAM ID	Rationale
Pollution (as it impacts site performance, radionuclide mobility, or monitoring)	1.4.07	The scope of this FEP is limited. Unnecessary speculation about future pollution should be avoided. Current and reasonably foreseeable pollution should be considered.
Social and institutional developments	1.4.11	The scope of this FEP is limited. Unnecessary speculation regarding future human actions and behavior should be avoided such as changes in demography, land use, controls, and regulatory requirements that may not need to be evaluated. Loss of control of a site due to loss of records or societal memory should be included and is considered in the inadvertent intruder analysis.
Technological developments	1.4.12	The scope of this FEP is limited. Technological developments are likely to occur but difficult to predict. Unnecessary speculation regarding future technological advances should be avoided (e.g., cure for cancer, technological advances in food production).
Explosions and crashes	1.4.14	The scope of this FEP is limited. For example, deliberate or malicious human actions may not need to be considered.
Meteorite impact[3]	1.5.1	Considered unlikely.
Species evolution[3]	1.5.2	Expected to be of limited significance in the timeframes of interest and with unknown impact.
Miscellaneous and FEPs of uncertain relevance[3]	1.5.3	Items in this category are not considered likely or significant to near-surface, LLW disposal (e.g., extraterrestrial activity, dust, changes in magnetic field, and change in tidal processes).
Non-radiological toxicity/effects	3.3.08	Non-radiological effects are not considered.

[1] FEP of limited scope.
[2] Table C-9 includes a portion of this FEP (pedogenesis) that may need to be considered.
[3] NEA FEPs that are excluded from the ISAM FEP list.

Table C-8 Assessment Context, Site Characterization, or Operational Factors Considered in Developing FEPs

IAEA ISAM FEP List	IAEA ISAM ID	Long-Term	Example Factors (Project IDs) BMA=BIOMASS, BMO=BIOMOVS, CLV=Clive, HAN=Hanford, SRS=Savannah River Site
Assessment endpoints (including intermediate outputs or results)	0.01		Points of assessment (e.g., well location, horizontal distance from source) (SRS 5.3.18)
			Annual Individual Dose (BMA 2.1.2, 2.1.2.1, BMO 1.1.2.1, HAN 0.4.06.01)
			Radionuclide Flux or Concentration (BMO 1.1.2.8, BMA, 2.1.2.8., SRS 1.1.09, SRS 5.3.12, HAN 0.4.06.08, HAN 0.4.06.09, HAN 0.4.06.10)
			Results, Presentation of (e.g., multiple lines of reason, barrier analysis, documentation, use of simpler models) (HAN 0.4.08)
Timescales of concern	0.02		Timeframes (BMA 2.1.7)
			Postclosure period (HAN 0.2.02.03)
			Assessment Timeframe (e.g., institutional control period, compliance period, >10,000 years, peak impact) (HAN 0.4.01, SRS 1.1.06, CLV 2)
			Safety Effects Beyond Periods of Control (beyond institutional control period) (SRS 1.1.07)
Spatial domain of concern[8]	0.03		Assessment Domain/Spatial Domain of Concern (HAN 0.4.02, SRS 1.1.08, CLV 4)
Facility[9] assumptions (e.g., assumptions regarding the success of facility closure and any changes to design, construction, or waste emplacement)	0.04		Repository System (BMA 2.1.4)
			Site, Context (BMA, 2.1.5, BMO 1.1.4)
			Facility Type (BMO 1.1.3)
			Disposal Facility Assumptions (HAN 0.4.09)
			Facility Factors (life cycle of disposal facility) (SRS 3.1.03)
Dose response assumptions[10]	0.07		Dose Response Assumptions (HAN 0.4.07)
Assessment purpose	0.08		Assessment Purpose (e.g., site selection and characterization, design, compliance, WAC, corrective action, confidence) (BMA 2.1.1, BMO 1.1.1, HAN 0.1, SRS 1.1.02)

8 Spatial extent over which disposed waste presents a risk to human health.
9 The ISAM FEP name was changed from “repository assumptions” to “facility assumptions” to avoid confusion related to terminology.
10 One example is the linear relationship of human health effect to dose with no threshold dose below which no effects are observed.

Table C-8 Assessment Context, Site Characterization, or Operational Factors Considered in Developing FEPs

IAEA ISAM FEP List	IAEA ISAM ID	Long-Term	Example Factors (Project IDs) BMA=BIOMASS, BMO=BIOMOVS, CLV=Clive, HAN=Hanford, SRS=Savannah River Site
			Assessment Audience (HAN 0.4.05)
Regulatory requirements and exclusions	0.09		Regulatory Requirements and Criteria (e.g., radiological protection standards, optimization) (HAN 0.2, CLV 3)
			Protection of Human Health and the Environment (SRS 1.2.02)
			Performance Requirements and Criteria (SRS 1.2.03)
			ALARA (SRS 1.2.05)
			Functional and Technical Requirements (e.g., containment, isolation, characterization, design, construction, quality assurance (QA), waste acceptance criteria (WAC), monitoring, analog, peer review) (HAN 0.2.03, SRS 1.2.04)
			Waste Acceptance Criteria (SRS 1.2.07)
Model and data issues (e.g., uncertainty, model abstraction)	0.10		Uncertainties, or confidence (BMA, 2.1.2.10)
			Confidence, Model (e.g., calibration, verification, validation) (HAN 0.3.04, SRS 1.3.12)
			Uncertainties, treatment of (e.g., subjective, future, conceptual, mathematical, model, parameter) (HAN 0.3.02, SRS 1.3.10)
			Sensitivity Analysis, Performance of (HAN 0.3.03, SRS 1.3.11)
			Model and Data Issues (including conceptual and mathematical model uncertainty, discretization, boundary conditions, coupled processes; parameter development and correlations; and scale issues) (SRS 1.3.01, CLV 1)
			Software Codes (SRS 1.3.02)
Assessment Philosophy or Approach	New		Assessment Philosophy (e.g., assessment and modeling approach, treatment of uncertainty, sensitivity analysis, and confidence building) (BMA 2.1.3, HAN 0.3)
			Assessment Approach (e.g., iterative, systemic, realistic, conservative, transparent) (HAN 0.3.01, SRS 1.3.04, SRS 1.3.05, SRS 1.3.06, SRS 1.3.07, SRS 1.3.08, SRS 1.3.09)
			Modeling Approach (e.g., screening, bounding, deterministic, probabilistic) (HAN 0.3.05, SRS 1.3.03)

Table C-8 Assessment Context, Site Characterization, or Operational Factors Considered in Developing FEPs

IAEA ISAM FEP List	IAEA ISAM ID	Long-Term	Example Factors (Project IDs) BMA=BIOMASS, BMO=BIOMOVS, CLV=Clive, HAN=Hanford, SRS=Savannah River Site
			Alternative Simplified Modeling Approach (SRS 1.3.13)
			Transparency of Assessment Approach (SRS 1.1.05)
			Documentation and Presentation of Results (SRS 1.1.04)
Site investigation	1.1.01		Technical Requirements (e.g., site characterization) (HAN 0.2.03)
			Investigations, Site (e.g., geology, hydrogeology, geochemistry, tectonic and seismicity, surface environment, meteorology and climatology, geography and demography, natural resources and land use) (HAN 1.1.01, CLV 82)
			Site Characterization and Investigations (SRS 3.1.01)
Schedule and planning	1.1.09		Schedule and Planning (e.g., construction, operation, and closure scope and schedule; alternative schedule) (HAN 1.1.03, SRS 3.2.02)
Administrative control, facility site[11]	1.1.10		Administrative control, Disposal facility (from pre- to post-closure and failures) (HAN 1.1.08, SRS 1.2.06)
			Institutional control (CLV 71)
Monitoring of facility[12]	1.1.11		Operation, Disposal Facility (e.g., monitoring) (HAN 1.1.05, CLV 84)
			Radionuclide Fluxes to the Biosphere (to monitor barrier performance) (SRS 5.3.12)
			Releases Prior to Closure (SRS 6.4.01)

[11] The ISAM FEP name was changed from “repository assumptions” to “facility assumptions” to avoid confusion related to terminology.
[12] The ISAM FEP name was changed from “repository assumptions” to “facility assumptions” to avoid confusion related to terminology.

Table C-9 Generic FEP List for Near-Surface Disposal in the United States (FEPs independent of human action and exposure assumptions)

IAEA ISAM FEP List	IAEA ISAM ID	Long-Term	Example FEPs (Project IDs) BMA=BIOMASS, BMO=BIOMOVS, CLV=Clive, HAN=Hanford, SRS=Savannah River Site
Design, facility[13]	1.1.02		Design, Disposal Facility (e.g., description, documentation, functional requirements, features, alternative designs) (HAN 1.1.02, CLV 101)
			Design error (CLV 123)
Construction, facility[14]	1.1.03		Construction, Disposal Facility (e.g., process, performance and verification, alternative conditions) (HAN 1.1.04)
			Construction (includes factors related to the excavation, stabilization, and the installation and assembly of structural elements) (SRS 3.2.05)
			Material defects (CLV 124)
Emplacement of wastes and backfilling	1.1.04		Operation, Disposal Facility (e.g., waste emplacement and repackaging; backfill preparation; handling and emplacement) (HAN 1.1.05)
			Operation (waste emplacement, backfilling, monitoring and surveillance, remedial activities) (SRS 3.2.06)
			Void Space Formation (SRS 3.8.06)
			Waste Type Classification (as it impacts disposal requirements for different classes of waste) (SRS 4.1.01)
			Waste Form Characteristics (SRS 4.1.02)
			Waste allocation and emplacement (SRS 4.1.04)
			Localized Interactions Between Emplaced Wastes (SRS 4.2.05)
Closure, facility[15]	1.1.05		Closure, Disposal Facility (e.g., closure plan, performance requirements, failure mechanisms, construction, confirmation, remedial alternatives) (HAN 1.1.06)
			Disposal Unit and/or Facility Closure (includes activities undertaken to prevent human access into and limit the migration of contaminants from the individual waste tanks) (SRS 3.2.08)

[13] The ISAM FEP name was changed from “repository assumptions” to “facility assumptions” to avoid confusion related to terminology.
[14] The ISAM FEP name was changed from “repository assumptions” to “facility assumptions” to avoid confusion related to terminology.
[15] The ISAM FEP name was changed from “repository assumptions” to “facility assumptions” to avoid confusion related to terminology.

Table C-9 Generic FEP List for Near-Surface Disposal in the United States (FEPs independent of human action and exposure assumptions)

IAEA ISAM FEP List	IAEA ISAM ID	Long-Term	Example FEPs (Project IDs) BMA=BIOMASS, BMO=BIOMOVS, CLV=Clive, HAN=Hanford, SRS=Savannah River Site
			Closure System Features and Materials (SRS 3.3.01)
			Compaction Error (CLV 98)
Waste allocation (projected inventory, waste acceptance criteria)	1.1.07		Operation, Disposal Facility (e.g., waste acceptance, waste allocation) (HAN 1.1.05)
			Radionuclide inventory disposed and remaining in the disposal facility (HAN 0.4.6.11)
			Activity limits in disposed waste (HAN 0.4.6.12, SRS 4.2.02)
			Waste allocation and emplacement (SRS 4.1.04)
			Homogeneity (SRS 4.1.05)
			Waste Acceptance Criteria (SRS 1.2.07)
Quality control	1.1.08		Quality control (CLV 88)
			Technical Requirements (e.g., QA) (HAN 0.2.03)
			Quality Assurance (e.g., all stages including research and development, procurement, manufacturing, siting, design, construction, commissioning, operation, decommissioning, QA/QC failures) (HAN 1.1.07)
			Procurement of Items and Services (quality assurance) (SRS 3.2.03)
			Manufacturing and Commissioning of Components (including defects) (SRS 3.3.02)
			Inadequate Quality Assurance/Control and Deviations from Design (SRS 3.8.04)
Accidents and unplanned events[16]	1.1.12		Accidents and unplanned events (e.g., human-induced and naturally occurring) (HAN 1.1.09, SRS 6.4.05, CLV 99, CLV 91)
Seismicity	1.2.03		Seismicity and effects (e.g., soil liquefaction) (HAN 1.2.03, SRS 6.2.01, CLV 50)
			Seismicity Associated with Igneous Activity (SRS 6.2.04)
			Earthquakes (CLV 40)
			Tsunami (CLV 30)

[16] For example, earlier than expected engineered barrier system failure, unexpected event, or unexpected waste.

Table C-9 Generic FEP List for Near-Surface Disposal in the United States (FEPs independent of human action and exposure assumptions)

IAEA ISAM FEP List	IAEA ISAM ID	Long-Term	Example FEPs (Project IDs) BMA=BIOMASS, BMO=BIOMOVS, CLV=Clive, HAN=Hanford, SRS=Savannah River Site
			Faulting (CLV 41)
Volcanic and magmatic activity	1.2.04	Long-term	Volcanic and magmatic activity (HAN 1.2.04, SRS 6.3.02, CLV 52)
			Igneous intrusion into the closure facility (SRS 6.3.01)
			Geological intrusion (CLV 43)
Hydrothermal activity	1.2.06	Long-term	Hydrothermal activity (HAN 1.2.06, SRS 5.2.02, CLV 13)
Erosion and sedimentation	1.2.07		Denudation and Deposition (large-scale, including erosion, corrasion, weathering, fluvial, aeolian, glacier, coastal, mass-wasting, sedimentation, deposition, events triggering mass wasting) (HAN 1.2.07, CLV 10, CLV 11)
			Geomorphologic response to geological changes (e.g, structural, weathering, erosional, and depositional landforms) (HAN 1.2.13, CLV 13)
			Depositional Environments and Landforms (including beaches, deltas, flood plains, and glacial moraines) (SRS 2.5.03)
			Sedimentation (SRS 2.6.06, CLV 14)
Pedogenesis[17]	1.2.08		Pedogenesis (factors related to the development and origin of soils, may also effect caps) (HAN 1.2.09, SRS 2.6.05, CLV 59)
Salt diapirism and dissolution	1.2.09	Long-term	Salt diapirism and dissolution (intrusion or upwelling of a salt formation into overlying strata (such as salt domes). Salt dissolution can occur when any soluble mineral is removed by flowing water) (HAN 1.2.10, SRS 2.6.11, CLV 9, CLV 38)
Hydrological/ hydrogeological response to geological changes	1.2.10	Long-term	Hydrological/hydrogeological response to geological changes (e.g., change in boundary conditions, surface water flow path, geochemical properties, and hydraulic properties; and preferential pathways) (HAN 1.2.12, SRS 5.2.01, CLV 13, CLV 120)
			Unconsolidated soft zones (as it affects site stability and contaminant flow and transport) (SRS 2.5.07)

[17] Diagenesis was originally included with this FEP; however, diagenesis is excluded in Table C-6.

Table C-9 Generic FEP List for Near-Surface Disposal in the United States (FEPs independent of human action and exposure assumptions)

IAEA ISAM FEP List	IAEA ISAM ID	Long-Term	Example FEPs (Project IDs) BMA=BIOMASS, BMO=BIOMOVS, CLV=Clive, HAN=Hanford, SRS=Savannah River Site
Climate change, global[18]	1.3.01	Long-term	Description of Climate Change (BMA 2.2.1.1, BMO 1.3.1.2)
			Climate-driven changes (BMO 2.1.1.2)
			Climate change, global (e.g., climate reconstruction, climate change theories) (HAN 1.3.01, SRS 2.7.07, CLV 31)
			Isostatic effects (CLV 46)
Climate change, regional and local	1.3.02		Description of Climate Change (BMA 2.2.1.1, BMO 1.3.1.2, SRS 2.7.02)
			Climate-driven changes (BMO 2.1.1.2)
			Climate change, regional and local (e.g., climate fluctuations, volcanic eruptions, global climate induced changes, lake effects) (HAN 1.3.02, SRS 2.7.07, CLV 31, CLV 32)
Sea level change	1.3.03	Long-term	Sea level changes (HAN 1.3.03)
			Isostatic Effects (CLV 46)
Periglacial effects (physical processes in cold but ice sheet free environments)	1.3.04	Long-term	Cold weather effects (permafrost, freeze/thaw cycles, frost heaving, gelifluction) (SRS 2.7.06, CLV 35)
			Periglacial (HAN 1.3.04, CLV 34)
			Solifluction (CLV 12)
			Glacial and ice sheet effects, local (HAN 1.3.05, CLV 34)
Glacial and ice sheet effects, local	1.3.05	Long-term	Warm climate effects (HAN 1.3.06, SRS 2.7.05)
Warm climate effects (tropical and desert)[19]	1.3.06		Hydrological/hydrogeological response to climate changes (e.g., change in driving forces of flow and flow) (HAN 1.3.07)
Hydrological/	1.3.07	Long-term	Climate Driven Changes (BMO 2.1.1.2)

[18] Does not include anthropogenic impacts on climate change.

[19] Facilities in tropical climates may experience extreme weather patterns such as monsoons, hurricanes, flooding, storm surges, and high winds; arid climates may be dominated by infrequent storm events.

Table C-9 Generic FEP List for Near-Surface Disposal in the United States (FEPs independent of human action and exposure assumptions)

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hydrogeological response to climate changes			Ecological changes (BMA 2.3.1.1.3, CLV 57)
Ecological response to climate changes	1.3.08		Climate Driven Changes (BMO 2.1.1.2)
			Ecological response to climate changes (e.g., soil, water, atmosphere, solar radiation, living organisms, ecological adaptation) (HAN 1.3.08, CLV 28)
Other geomorphological changes	1.3.10		Geomorphologic responds to climate changes (e.g., periglacial landforms, warm climate) (HAN 1.3.10)
			Surface excavations (e.g., construction, remediation, geotechnical investigation, waste disposal) (HAN 1.4.07, CLV 65)
Pollution (as it impacts site performance, radionuclide mobility, or monitoring)[20]	1.4.07		Artificial soil fertilization (BMA 2.3.2.1.1, BMO 2.2.3.3)
			Chemical pollution (BMA 2.3.2.1.2, BMO 2.2.1.1)
			Acid rain (BMA 2.3.2.1.3, BMO 2.2.1.1.1, SRS 2.7.04)
			Pollution (e.g., soil, groundwater, and air that may increase mobility of radionuclides) (HAN 1.4.12, SRS 2.2.07)
Inventory, radionuclide and other material	2.1.01		Inventory, waste (e.g., waste stream, volume, homogeneity, radiological content, non-radiological content (as it impacts radionuclide mobility), classification, uncertainty) (HAN 2.1.01, SRS 4.1.03)
			Source Term/Radionuclide Content (BMA 2.1.6, BMO 1.2.3.1)
Wasteform materials, characteristics and degradation processes	2.1.02		Wasteform, characteristics and degradation processes (HAN 2.1.02, CLV 100)
			Removal or Stabilization of Waste (stabilization of waste) (SRS 3.2.07)
			Waste characteristics (e.g., metallic, inorganic/non-metallic, organic, volatile, combustible) (SRS 4.1.07, 4.1.08, 4.1.09, 4.1.10, CLV 77, CLV 104)
			Waste degradation (inorganic and organic) (SRS 4.2.06, SRS 4.2.07)

[20] Unnecessary speculation about future pollution should be avoided. Current and reasonably foreseeable pollution should be considered.

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Container materials, characteristics, and degradation processes	2.1.03		Container materials, characteristics, and degradation processes (e.g., corrosion) (HAN 2.1.03, CLV 96, CLV 97, CLV 100)
			Waste tank, container, or package characteristics (physical, chemical, and mechanical properties) (SRS 3.3.04)
			Waste Container, Package, or Over-Pack Failure (e.g., corrosion, manufacturing defects, improper seal) (SRS 3.7.06)
Buffer/backfill materials, characteristics, and degradation processes	2.1.04		Buffer/backfill, characteristics, and degradation processes (HAN 2.1.04, CLV 100)
			Closure System Buffer (Engineered Surface Cover, Backfill, and Near-Field Soil) Properties (e.g., dehydration of zeolites, mineralogical dehydration, geothermal fluid impacts, sorption, use of bentonite and vermiculite) (SRS 3.3.05, SRS 3.3.06)
			Swelling of Backfill and Emplacement Materials (e.g., bentonite and vermiculite degradation) (SRS 3.3.06, SRS 3.7.08)
Engineered barriers system (EBS), characteristics and degradation processes	2.1.05		Other engineered features, characteristics, and degradation processes (e.g., final or interim covers or multi-layer cap designs) (HAN 2.1.05, CLV 100)
			Multi-Barrier Safety Function (combination of natural and engineered barriers) (SRS 3.1.04)
			Design Basis for Engineered Components (SRS 3.2.01)
			Consolidation of System Components (consolidation of engineered barrier system components that may affect the chemical environment and release) (SRS 3.3.03)
			Chemical Degradation of Engineered System Metals (SRS 3.7.01)
			Corrosion (e.g., stress corrosion cracking, hydride cracking) (SRS 3.7.02, SRS 3.7.03)
			Creep of Metallic Materials in the Engineered System (SRS 3.7.04)
			Oxygen Embrittlement of Engineered System Metals (SRS 3.7.05)
			Concrete Shrinkage/Expansion (may impact hydraulic properties) (SRS 3.7.09)
			Cementitious material degradation (e.g., sulfate and chloride attack, carbonation, (SRS 3.7.10, SRS 3.7.11)
			Seismic-Induced Damage or Changes to System Components (SRS 6.2.02)
			Frost Weathering (engineered cover degradation mechanism) (CLV 21)

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			Geophysical effects on degradation (CLV 36)
Other engineered features materials, characteristics and degradation processes	2.1.06		Other engineered features, characteristics and degradation processes (e.g., vault structure, site cut off walls or fire breaks, engineered drainage system) HAN 2.1.05
			Engineered Barrier Thickness and Other Material Properties (e.g., engineered surface cover, vaults, basemat) (SRS 3.3.07, SRS 3.3.08, SRS 3.3.09, CLV 100)
			Ancillary Equipment and Piping/Transfer Lines (SRS 3.3.10)
			Degradation of Non-Metal Solids: Backfill, Rock, Grout, Cement, etc. (SRS 3.7.07)
			Polymer Degradation (SRS 3.7.12)
			Material Volume Changes (e.g., expansion may lead to cracking of materials) (SRS 3.8.07)
Mechanical processes and conditions (in wastes and EBS)	2.1.07		Mechanical Effects at EBS Component Interfaces (e.g., mechanical and static loading) (SRS 3.8.09)
			Effects of Subsidence (including increased infiltration) (SRS 6.2.03, CLV 8, CLV 103)
			Cave-In, Collapse, or Rockfall (SRS 6.4.04)
			Compaction Error (CLV 98)
Hydraulic/hydrogeological processes and conditions (in wastes and EBS)	2.1.08		Hydrological Processes and Conditions (processes affecting flow through waste and engineered features) (SRS 3.4.01, CLV 99)
			Hydrostatic Pressure on the Closure System (hydrostatic pressure (or suction) of saturated waste and engineered system components) (SRS 3.4.02)
			Condensation on Closure System Surfaces (SRS 3.4.03)
			Resaturation and Desaturation (equilibration of engineered barriers with surrounding materials, may cause hydraulic, thermal, chemical, mechanical changes such as expansion, cooling, corrosion) (SRS 3.4.04)
			Groundwater Flow and Movement (Near-Field) (e.g., preferential flow) (SRS 5.1.01)
			Focusing of Flow Along Preferred Flow Paths (Fingers, Weeps, Faults, Fractures, etc.) (SRS 5.1.05)

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			Episodic Or Pulse Flow and Release (SRS 5.1.03)
			Water Influx at the Closure Facility (SRS 5.1.04)
			Flow Diversion and Bypass Flow (focused flow around cap, flow through voids such as roots) (SRS 5.1.07)
			Film/Laminar Flow (through waste zone) (SRS 5.1.08)
			Perched Water (SRS 5.2.06)
			Contaminant Release Pathways (SRS 5.3.02, CLV 102)
			Multi-Phase Transport Processes (SRS 5.3.03)
			Fast Transport/Preferential Pathways (SRS 5.3.13, CLV 120)
			Flooding or drainage system failure (SRS 6.4.02)
			Mechanical effects (CLV 125)
Chemical/geochemical processes and conditions (in wastes and EBS)	2.1.09		Chemical/geochemical-mediated processes, effects on contaminant release and migration (including dissolution, precipitation, speciation, solubility, sorption/desorption, colloids, chemical complexing agents, and reconcentration; interface between near- and far-field; and effects on sorption and permeability) (HAN 3.2.02, SRS 3.5.01, SRS 5.3.01, CLV 9, CLV 102)
			Evolving water chemistry in wasteform, containment system, or near-field (e.g., dissolution of cementitious materials and impacts on contaminant mobility; impacts of leachate on near-field transport; pH and Eh changes; colloid generation) (SRS 3.5.02, SRS 3.5.03, SRS 3.5.05, SRS 3.5.06, SRS 3.5.07, CLV 9, CLV 102)
			Chemical Effects of Waste-Rock Contact (direct contact of waste with rock due to failure of waste package) (SRS 3.5.08)
			Rind (Chemically Altered Zone) Forms in the Near-Field (thermal-chemical processes involving precipitation, condensation, and re-dissolution could alter the properties of the adjacent materials) (SRS 3.5.09)
			Reaction Kinetics (non-equilibrium conditions) (SRS 3.5.11)
			Contaminant Release from the Waste Form and Engineered Barrier System (SRS 5.3.04, CLV 99)

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			Long-Term Release of Radionuclides (SRS 5.3.14)
			Electrochemical effects (CLV 107)
Biological/biochemical processes and conditions (in wastes and EBS)	2.1.10		Microbial/biological-mediated processes, effects on contaminant release and migration (e.g., biological activity that may change radionuclide mobility and microbes as colloids) (HAN 3.2.03, SRS 2.4.02, CLV 54, CLV 102)
Thermal processes and conditions (in wastes and EBS)	2.1.11		Thermal Processes and Conditions the Engineered System (as from cement hydration, radioactive decay) (SRS 3.6.01)
			Thermo-Chemical Alteration, Near-Field (e.g., effects on solubility) (SRS 3.6.03)
			Thermo-Mechanical Stresses Alter Characteristics of Engineered Barrier System Components (e.g., thermal cracking of cementitious materials) (SRS 3.6.04)
			Recrystallization of Vitrified Wastes (SRS 3.6.05)
Gas sources and effects (in wastes and EBS)	2.1.12		Waste Form Characteristics (e.g., gas generation) (HAN 2.1.02, SRS 4.1.02)
			Gas generation (e.g., radon) (CLV 58)
Radiation effects (in wastes and EBS) (e.g., radiolysis, material degradation)	2.1.13		Radiation Effects on the Waste Closure System (SRS 4.5.02)
			Radiolysis Effects (SRS 4.5.07)
			Radiological effects (CLV 55)
Nuclear criticality	2.1.14		Nuclear Criticality (HAN 1.1.11, SRS 4.5.01, CLV 105)
Extraneous materials	2.1.15		No examples were identified.
Disturbed zone, host lithology	2.2.01		Disturbed zone, host lithology (e.g., formation of cracks, interface, hydro mechanical effects, backfilling, contaminant migration) (HAN 2.2.03)
Host lithology	2.2.02		Host lithology (e.g., geological properties, physical characteristics) (HAN 2.2.02, CLV 119, CLV 100)
			Stratigraphy and Host Lithology (including description, flow and transport effects, properties, homogeneity, and potential changes) (SRS 2.5.04)
Lithological units, other	2.2.03		Stratigraphy (e.g., stratigraphic record, formation) (HAN 2.2.01, CLV 119)

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Discontinuities, large scale (in geosphere)	2.2.04		Discontinuities, large scale (in geosphere) (e.g., faults, folds, dykes, aquifer formation, discontinuities affecting boundary conditions) (HAN 2.2.04, SRS 2.5.05, CLV 39)
Contaminant transport path characteristics (in geosphere) (e.g., fracture flow, porous flow, fracture/matrix interactions)	2.2.05		Contaminant migration path characteristics (in geosphere) (e.g., hydrogeological zones, interstitial geometry, bypass flow, fracture infill, weathering) (HAN 2.2.05, CLV 111)
			Unconsolidated soft zones (SRS 2.5.07)
			Fast Transport Pathways (SRS 5.3.13)
Mechanical processes and conditions (in geosphere)	2.2.06		Mechanical processes and conditions (in geosphere) (e.g., changes in stress field, mechanical load, mechanical rapture, changes in rock properties) (HAN 2.206, SRS 2.6.01)
			Stress Regimes (caused by coupled thermal-hydro-mechanical effects; swelling of materials; isostatic rebound (such as when glaciers recede), salt creep, etc., and can lead to changes in flow/directions). (SRS 2.5.09)
Hydraulic/hydrogeological processes and conditions (in geosphere)	2.2.07		Hydraulic/hydrogeological processes and conditions (in geosphere) (e.g., hydrological cycle, groundwater flow, saturated/unsaturated flow, water table fluctuations, boundary conditions/variability, gradients, hydraulic properties/variability, salinity, geothermal gradient) (HAN 2.2.07, SRS 2.5.11, SRS 2.6.12, SRS 5.1.02, SRS 5.1.09, SRS 5.2.03, CLV 16, CLV 100)
			Hydrological Regime and Water Balance (Near-Surface) (SRS 2.8.09)
			Aquifer Properties (including vadose and saturated zone thicknesses) (SRS 5.3.16, 5.3.17)
			Capillary rise (SRS 2.8.05)
			Discharge Zones Within and Outside the Assessment Domain (including discharge to sensitive or remote areas) (SRS 2.8.07, SRS 2.8.08, SRS 5.3.15)
			Interfaces Between Different Waters (SRS 5.2.04)
			Chemically-Induced Density Effects on Groundwater Flow (SRS 5.1.06)
			Unconsolidated soft zones (SRS 2.5.07)
			Hydrological Effects (CLV 17)
			Flooding (CLV 19)

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Chemical/geochemical processes and conditions (in geosphere)	2.2.08		Properties of the Groundwater Plume (BMO 1.2.3.2)
			Chemical/geochemical processes and conditions (in geosphere) (e.g., chemical composition and evolution; geochemistry factors, evolution and geochemical interactions; groundwater recharge; geothermal effects, solubility controls) (HAN 2.2.08, CLV 9)
			Evolving Water Chemistry in the Far-Field (e.g., perturbations due to climate change which can cause infiltration of sea-water or glacial melt waters) (SRS 3.5.04)
			Complexation in the Natural System (SRS 3.5.10)
			Alteration and Chemical Weathering Along Flow Paths (SRS 5.1.10)
Biological/biochemical processes and conditions (in geosphere)	2.2.09		Biological/biochemical processes and conditions (in geosphere) (e.g., biogeochemical changes, generation of chelating agents, microbial species, influence on pH and redox potential, and changes in microbial populations) (HAN 2.2.09, CLV 54)
Thermal processes and conditions (in geosphere)	2.2.10		Thermal processes and conditions (in geosphere) (e.g., sources and distribution of geothermal heat, geothermal gradient, temperature, and geothermal induced changes from fracturing, fracture displacement, solubility changes, or changes in flow directions due to buoyancy) (HAN 2.2.10, SRS 3.6.02)
			Temperature and Thermal Gradient Effects on the Geosphere (temperature can influence rates of chemical and microbiological processes, stress field, groundwater flow, diffusion rates, and radionuclide transport) (SRS 3.6.07)
Gas sources and effects (in geosphere)	2.2.11		Gas sources and effects (in geosphere) (e.g., effects of natural gases, gas induced groundwater flow) (HAN 2.2.11)
			Gas or brine pockets (CLV 6)
Undetected features (in geosphere) (e.g., boreholes, abandoned mines, gas or brine	2.2.12		Undetected features (in geosphere) (HAN 1.2.11, SRS 2.5.08)
			Breccia pipes (CLV 37)
			Lava tubes (CLV 47)

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pockets, geological features [faults, dykes, lava tubes, breccia pipes, lava tubes])			Gas intrusion (CLV 121)
Geological resources	2.2.13		Geological resources (in geosphere) (e.g., methane, water, and other resources; and near-surface and deep deposit exploration) (HAN 2.2.12, CLV 119)
			Natural and Geological Resources and Land Use (resources including water, lumber, oil, gas, minerals, geothermal energy; land use including reclamation, logging, agricultural, urbanization, and waste disposal) (SRS 2.2.01, CLV 79)
			Hydraulic fracturing (CLV 44)
			Resource extraction (CLV 89)
Topography and morphology	2.3.01		Topography and morphology (e.g., landform, changes in landform, topography changes on climate) (HAN 2.3.01, SRS 2.5.02)
Soil and sediment (physical, chemical and biological properties; sedimentation; evolution)	2.3.02		Chemical changes (e.g., chemical changes to soil) (BMA 2.3.1.1.2)
			Geosphere/Biosphere Interface (e.g., deep soil) (BMO 1.2.1)
			Environmental Components (e.g., top soil, deep soil) (BMO 1.3.2.3)
			Alkalinization (BMO 2.1.1.1.2.1)
			Soil conversion (BMO 2.1.1.1.3.1)
			Soil and sediment (composition, structure, profile, type, depth, and impact on contaminant mobility) (HAN 2.3.03, SRS 2.5.10)
			Radionuclide Accumulation (Recycling) in Soils (SRS 4.3.05)
			Sedimentation (CVL 14)
Aquifers and water-bearing features, near surface	2.3.03		Environmental Components (e.g., deep soil, biosphere aquifer) (BMO 1.3.2.3)
			Aquifers and water-bearing features, near surface (including water table depth, aquifer type, geometry [fractures], recharge and discharge zones, alkali flats, sea water intrusion) (HAN 2.3.04, SRS 2.5.06)

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Lakes, rivers, streams and springs	2.3.04		Geosphere/Biosphere Interface (e.g., river) (BMO 1.2.1)
			Environmental Components (e.g., river water, river sediment) (BMO 1.3.2.3)
			Terrestrial surface water bodies (lakes, dams, rivers, streams, springs, wetlands, dilution, and recharge/discharge zones) (HAN 2.3.05)
			Surface water bodies (characteristics of surface-water bodies such as rivers, lakes, wetlands and springs, and their evolution in time) (SRS 2.8.02, CLV 20)
			Lake Effects (including appearance, disappearance of large lake leading to possible sedimentation, wave action, erosion/inundation, isostasy) (CLV 18, CLV 32)
			Wave Action (from large lake) (CLV 33)
			Regional subsidence (effect on lake levels) (CLV 49)
Coastal features	2.3.05		Coastal features and processes (HAN 2.3.06, CLV 26)
Marine features	2.3.06		Marine features and effects (HAN 2.3.07, CLV 29)
Atmosphere (e.g., physical transport of gases, chemical and photochemical reactions, aerosols and dust)	2.3.07		Environmental Components (e.g., atmosphere) (BMO 1.3.2.3)
			Atmosphere (transport of contaminants in gas, vapor, or particulate/aerosol phase; contamination of atmosphere may occur due to water evaporation, degassing from soils or water, transpiration from plants, suspension due to wind erosion, plowing, or fires) (HAN 2.3.08, SRS 2.7.01)
			Effects related to air and vapor flow and evaporation within the system (SRS 5.2.05)
			Gas Transport (CLV 117)
			Resuspension (CLV 23)
			Atmospheric dispersion (CLV 24)
			Tornado (CLV 25)
Vegetation	2.3.08		Vegetation (including such factors as vegetation type, contamination processes, retention, fires, root intrusion, evolution, and hydroponics) (HAN 2.3.09, SRS 2.4.03, CLV 56)

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Animal populations (e.g., diets, external contamination of)	2.3.09		Living components of ecosystems (BMA 2.2.3.2.1)
			Animal populations (including animal types, burrowing animals, scavengers and predators, and pets) (HAN 2.3.10, SRS 2.4.04)
Meteorology	2.3.10		Identification and Characterization of Climate Categories (BMA 2.2.1.2)
			Differentiation of Climate Categories (BMO 1.3.1.1)
			Diurnal variability (BMA 2.3.1.2.1)
			Meteorology (HAN 2.3.11, CLV 22)
			Precipitation (control on the amount of runoff and infiltration, flow in the unsaturated zone, and groundwater recharge) (SRS 2.7.03)
			Events (e.g., flooding, tornados, hurricanes) (CLV 10, CLV 25, CLV 27)
			Insolation (CLV 28)
Hydrological regime and water balance (near surface)	2.3.11		Interannual and longer timescale variability (e.g., seasonal water table fluctuations) (BMA 2.3.1.2.3)
			Hydrological regime and water balance (near-surface) (HAN 2.3.12)
			Water (characteristics of water and its evolution) (SRS 2.8.01)
			Evapotranspiration, surface runoff, infiltration, and recharge (SRS 2.8.03, SRS 2.8.04, SRS 2.8.06, CLV 118)
			Hydrological Effects (CLV 17)
Erosion and deposition	2.3.12		Physical changes (e.g., erosion, sea level change) (BMA 2.3.1.1.1, BMO 2.1.1.1.3)
			Erosion (BMA 2.3.1.4.2.10, BMO 2.1.1.1.3.2, CLV 11)
			Erosion and deposition (HAN 2.3.13)
			Erosion and weathering (SRS 2.6.08)
			Deposition (SRS 2.6.07)

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			Mass Wasting (soil movement as a result of gravitational forces) (SRS 2.6.09)
			Creeping of the Rock Mass (slow movement of the rock along pre-existing discontinuities or in the rock matrix due to differential stress fields; may affect hydraulic properties of the rock) (SRS 2.6.10)
			Subrosion (CLV 15)
Ecological/biological/microbial systems (e.g., features, microbial activity, chemical changes). The NEA database also has the following examples: fire, ecological succession.	2.3.13		Ecosystems (BMA 2.2.3.2)
			Non-living components of ecosystems (BMA 2.2.3.2.2)
			Interannual and longer timescale variability (e.g., natural succession after fire) (BMA 2.3.1.2.3)
			Burning (BMO 2.1.2.1.3)
			Chemical Changes Caused by Micro-Organisms (BMO 2.1.1.1.2.2)
			Ecological/biological/microbial systems (HAN 2.3.14)
			Biomes (e.g., desert, grassland, forest, and mountain biomes) (HAN 2.3.02, SRS 2.4.01)
			Insolation, effects of (CLV 28)
Animal/plant intrusion leading to vault/trench disruption (e.g., root uptake, burrowing animals)	2.3.14		Transport mediated by flora and fauna (e.g., root uptake, transpiration, interception, intake by fauna, bioturbation, burrowing, root development, translocation) (BMA 2.3.1.3.1, BMO 2.1.2.7, CLV 109)
			Animal/Plant intrusion (HAN 2.3.15, SRS 6.1.05)
Radioactive decay and ingrowth	3.1.01		Radioactive decay and ingrowth (HAN 3.1.01, SRS 4.2.01, CLV 60)
			Radionuclide properties, other (HAN 3.1.02)
Chemical/organic toxin stability	3.1.02		Chemical/organic toxin stability (HAN 3.1.04)
Inorganic solids/solutes	3.1.03		Inorganic solids/solutes (HAN 3.1.05)
Volatiles and potential for volatility	3.1.04		Volatiles and potential for volatility (HAN 3.1.06)

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IAEA ISAM FEP List	IAEA ISAM ID	Long-Term	Example FEPs (Project IDs) BMA=BIOMASS, BMO=BIOMOVS, CLV=Clive, HAN=Hanford, SRS=Savannah River Site
Organics and potential for organic forms	3.1.05		Organics and potential for organic forms (HAN 3.1.03)
Noble gases	3.1.06		Noble gases (HAN 3.1.07)
Dissolution, precipitation and crystallisation, contaminant	3.2.01		Release Mechanisms (BMA 2,1.6.2, CLV 102)
			Source Term Mechanisms (BMA 2.1.6.3)
			Dissolution/precipitation (BMA 2.3.1.4.4.1, BMO 2.1.3.1.1, SRS 4.3.02)
			Chemical/geochemical-mediated processes, effects on contaminant release and migration (e.g., dissolution, precipitation, and crystallisation) (HAN 3.2.02)
			Dissolution (CLV 115)
Speciation and solubility, contaminant	3.2.02		Source Term/Release Mechanisms (BMA 2,1.6.2, BMA 2.1.6.3, CLV 102)
			Chemical/geochemical-mediated processes, effects on contaminant release and migration (e.g., speciation and solubility) (HAN 3.2.02, CLV 9)
			Contaminant Solubility, Solubility Limits, and Speciation (SRS 4.2.03)
			Reduction-Oxidation Potential (Redox Fronts) (SRS 4.2.04)
			Solubility and Sorption Changes from Chemical and Temperature Interactions (SRS 4.3.03)
Sorption/desorption processes, contaminant	3.2.03		Source Term/Release Mechanisms (BMA 2,1.6.2, BMA 2.1.6.3, CLV 102)
			Adsorption/desorption (BMA 2.3.1.4.4.2, BMO 2.1.3.1.2, SRS 5.3.11, CLV 111)
			Soil and sediment (e.g., sorptive capability) (HAN 2.3.08)
			Chemical/geochemical-mediated processes, effects on contaminant release and migration (e.g., sorption/desorption processes) (HAN 3.2.02, CLV 9)
			Electrochemical Effects in the Closure System (Including Anion Exclusion) (SRS 3.8.08)
			Solubility and Sorption Changes from Chemical and Temperature Interactions (SRS 4.3.03)
			Radionuclide Interaction with Corrosion Products (SRS 4.5.03)

Table C-9 Generic FEP List for Near-Surface Disposal in the United States (FEPs independent of human action and exposure assumptions)

IAEA ISAM FEP List	IAEA ISAM ID	Long-Term	Example FEPs (Project IDs) BMA=BIOMASS, BMO=BIOMOVS, CLV=Clive, HAN=Hanford, SRS=Savannah River Site
Colloids, contaminant interactions and transport with	3.2.04		Colloid formation (BMA 2.3.1.4.4.3)
			Transport of Colloids (BMO 2.1.2.3.6.2)
			Colloids mediated migration of contaminant (HAN 3.2.10, CLV 110)
Chemical/complexing agents, effects on contaminant speciation/transport	3.2.05		Source Term Content of Other Hazardous Materials[21] (BMO 1.2.3.3)
			Chemical/geochemical processes and conditions (in geosphere) (e.g., naturally occurring complexing agents or complexing agents formed in the near-field) (HAN 2.2.08, CLV 9)
			Chelating Agent Effects (SRS 3.5.12)
Microbial/biological/plant-mediated processes, contaminant	3.2.06		Microbial/biological-mediated processes, effects on contaminant release and migration (HAN 3.2.03, SRS 2.4.02, CLV 54, CLV 102)
Water-mediated transport of contaminants	3.2.07		Water-borne transport (e.g., surface run-off, infiltration, percolation, multi-phase flow, recharge, capillary rise, groundwater transport, discharge) (BMA 2.3.1.4.2)
			Geosphere Aquifer Discharge (BMO 1.2.2.1.1)
			Porous Media Aqueous Transport Processes (e.g., infiltration, percolation, matrix diffusion, diffusion/dispersion, dual flow systems, capillary rise, groundwater transport) (BMO 2.1.2.3, CLV 118)
			Physical Processes (e.g., rainfall, snowfall, evaporation, evapotranspiration) (BMO, 2.1.3.2)
			Surface Water Aqueous Transport Processes (e.g., surface water run-off, transport in water bodies) (BMO 2.1.2.2, CLV 17, CLV 111)
			Water-mediated migration of contaminants (including groundwater advection, dispersion, dilution, imbibition, diffusion, multi-phase, unsaturated flow; surface water transport; currents; and sea spray) (HAN 3.2.04, SRS 5.3.07, SRS 5.3.08, SRS 5.3.09, CLV 16, CLV 112, 113, 114)
			Dilution of Radionuclides in Groundwater (SRS 4.3.04)

[21] This FEP is included to the extent that chemical constituents impact the mobility of radionuclides. Non-radiological impacts are not considered.

Table C-9 Generic FEP List for Near-Surface Disposal in the United States (FEPs independent of human action and exposure assumptions)

IAEA ISAM FEP List	IAEA ISAM ID	Long-Term	Example FEPs (Project IDs) BMA=BIOMASS, BMO=BIOMOVS, CLV=Clive, HAN=Hanford, SRS=Savannah River Site
			Contaminant Release Pathways (water-mediated release from infiltration, breach of engineered barriers or wasteform, and leaching) (HAN 3.2.01, CLV 118, CLV 119)
			Leaching (SRS 3.5.14)
Solid-mediated transport of contaminants	3.2.08		Solid-phase transport (e.g., landslides, rock fall, wash out, sedimentation, resuspension, rain splash) (BMA 2.3.1.4.3, BMO 2.1.2.5, CLV 7)
			Solid-mediated migration of contaminants (including from erosion, denudation, resuspension, sediment migration, saltation, wet deposition, and mass wasting) (HAN 3.2.05, CLV 10, CLV 11, CLV 12)
			Contaminant Release Pathways (e.g., solid-mediated release from intrusion, natural disruption, or animal action) (HAN 3.2.01, CLV 102)
			Solid-Mediated Migration of Contaminants (SRS 5.3.05)
Gas-mediated transport of contaminants	3.2.09		Gaseous Release (BMO 1.2.2.2)
			Gaseous Transport (BMO 2.1.2.1.1.)
			Gas-mediated migration of contaminants (including expelling groundwater, gas-phase transport, and multi-phase flow effects) (HAN 3.2.06)
			Contaminant Release Pathways (gas-mediated release from barometric pressure changes, controlled release of gas, and dissolution; gas release models, transport processes, biosphere entry points, and monitoring may also be discussed) (HAN 3.2.01, CLV 102)
			Gas-Mediated Migration of Contaminants (SRS 5.3.06)
Atmospheric transport of contaminants	3.2.10		Atmospheric transport (e.g., gas, aerosols, evaporation, wet/dry deposition) (BMA 2.3.1.4.1)
			Aerosol Transport (BMO 2.1.2.1.2)
			Wet Deposition (BMO 2.1.2.5.4.5)
			Atmospheric migration of contaminants (including convection, diffusion, turbulence, deposition, saltation, burning, showers and humidifiers) (HAN 3.2.07, CLV 111)
			Dust devils (CLV 116)

Table C-9 Generic FEP List for Near-Surface Disposal in the United States (FEPs independent of human action and exposure assumptions)

IAEA ISAM FEP List	IAEA ISAM ID	Long-Term	Example FEPs (Project IDs) BMA=BIOMASS, BMO=BIOMOVS, CLV=Clive, HAN=Hanford, SRS=Savannah River Site
Animal, plant and microbe mediated transport of contaminants (e.g., microbial-enhanced contaminant transport)	3.2.11		Recycling of bulk solid materials (e.g., recycling materials for compost or mulch) (BMA 2.3.2.3.5, BMO 2.2.3)
			Fodder products (BMA 2.4.1.1.3)
			Animal, plant and microbe mediated migration of contaminants (including animal intrusion, external contamination, carcasses, root uptake, microbes, translocation, leaf deposition) (HAN 3.2.08, SRS 4.4.04, CLV 54, CLV 111)
Radiological toxicity/effects	3.3.06		Radiological toxicity/effects (somatic, genetic, stochastic, non-stochastic) (HAN 3.3.07)
Scope of FEP limited. See Table C-7 for more information on what scope can be excluded from the FEP.			
Additional FEPs not included in the NEA international FEPs database.			

Table C-10 Generic FEPs List for Near-Surface Disposal in the United States (FEPs related to human actions, exposure assumptions, etc.)

IAEA ISAM FEP List	IAEA ISAM ID	Long-Term	Example FEPs (Project IDs) BMA=BIOMASS, BMO=BIOMOVS, CLV=Clive, HAN=Hanford, SRS=Savannah River Site
Future human action assumptions	0.05		Future Human Action Assumptions (e.g., present day technology, past as reflection of future) (HAN 0.4.03, SRS 2.3.01, CLV 67)[22]
Future human behavior (target group) assumptions	0.06		Future Human Behavior (target group) Assumptions (HAN 0.4.04, CLV 67) [23]
			Human Behavior and Habits (Non-Diet Related) (SRS 2.1.03, CLV 131)
Records and markers, facility[24]	1.1.06		System of Records (HAN 0.2.03.09)
			Loss/degradation of societal memory (HAN 1.4.02.06)
			Loss of archives/records (HAN 1.4.02.07)
Natural climate cycling (e.g. glaciation)	1.3.09	Long-term	Human behavior response to climate changes (e.g., irrigation rates) (HAN 1.3.09)
Motivation and knowledge issues[25] (inadvertent human actions)	1.4.02		Motivation and knowledge issues (inadvertent intrusion) (HAN 1.4.03)
			Inadvertent Intrusion (SRS 6.1.01, CLV 69)
			Loss/degradation of societal memory (HAN 1.4.02.06)
			Loss of archives/records (HAN 1.4.02.07)
			Future Knowledge of the Facility (loss of records) (SRS 2.3.02)

[22] The Hanford FEP list includes three sub-categories of future human actions that are explicitly excluded in Table C-7: (1) cure for cancer, (2) malicious acts or acts of war, and (3) deliberate intrusion. If a cure for cancer is assumed, then future generations will not be protected at the same level as current generations. Additionally, credit cannot be given for an assumed future mitigative action to meet regulatory standards today. The regulations in 10 CFR Part 61 do not protect against deliberate acts or advertent intrusion. In general, analysts should avoid unnecessary speculation about future human actions that tend to reduce potential dose impacts. See Table C-7 for additional details.

[23] Use of dose estimates for the average member of the critical group, or that group of individuals reasonably expected to receive the highest dose based on current or reasonably foreseeable (e.g., within the next 100 years) future practices, is an acceptable approach for technical analyses. This FEP should not include unsupported assumptions regarding future human species evolution or changes to radiosensitivity. See Table C-7 for additional details.

[24] The ISAM FEP name was changed from "repository assumptions" to "facility assumptions" to avoid confusion related to terminology.

[25] The original ISAM FEP (and Hanford FEP) was named "Motivation and knowledge issues (inadvertent/deliberate human actions)" but was renamed "Motivation and knowledge issues (inadvertent human actions)." This FEP does not include advertent or deliberate intrusion into the disposal facility. See Table C-6 for additional details regarding exclusion of deliberate or advertent intrusion.

Table C-10 Generic FEPs List for Near-Surface Disposal in the United States (FEPs related to human actions, exposure assumptions, etc.)

IAEA ISAM FEP List	IAEA ISAM ID	Long-Term	Example FEPs (Project IDs) BMA=BIOMASS, BMO=BIOMOVS, CLV=Clive, HAN=Hanford, SRS=Savannah River Site
			Igneous or Seismic Event Precedes Human Intrusion (making it difficult to recognize engineered wasteform) (SRS 6.1.06)
Drilling activities (human intrusion)	1.4.03		Water extraction by pumping (BMA 2.3.2.2.2, BMO 2.2.4.1)
			Water recharge by pumping (BMA 2.3.2.2.3)
Mining and other underground activities (human intrusion)	1.4.04		Drilling activities (e.g., exploration, studies, site characterization, exploitation, construction, water-supply well, reuse of boreholes, waste disposal, remedial action, injection wells, geothermal) (HAN 1.4.04, SRS 6.1.03, CLV 73, CLV 80)
			Mining and other underground activities (e.g., mining, underground construction/dwelling, exploitation, exploration, excavation, solution mining, waste disposal, underground nuclear testing) (HAN 1.4.05, SRS 6.1.04, CLV 65, CLV 73, CLV 85)
Un-intrusive site investigation	1.4.05		Un-intrusive site investigation (HAN 1.4.06)
Surface excavations	1.4.06		Dredging (BMA 2.3.2.3.7, BMO 2.2.4.4.1)
			Ploughing (BMA 2.3.2.3.1, BMO 2.2.3.1)
			Earth Works (BMO 2.2.4.4.2)
Site Development	1.4.08		Land reclamation (BMA 2.3.2.2.5, BMO 2.2.2.3)
			Dam building (BMA 2.3.2.2.4, BMO 2.2.2.2)
			Construction (BMA 2.3.2.2.1)
			Site Development (e.g., construction, road building, dam building, drainage, change in topography, change in land use) (HAN 1.4.08, SRS 3.1.02)
			Community development (e.g., establishment of residences) (CLV 64, CLV 70)
			Land use (CLV 72)
Archaeology	1.4.09		Archaeology (HAN 1.4.09)

Table C-10 Generic FEPs List for Near-Surface Disposal in the United States (FEPs related to human actions, exposure assumptions, etc.)

IAEA ISAM FEP List	IAEA ISAM ID	Long-Term	Example FEPs (Project IDs) BMA=BIOMASS, BMO=BIOMOVS, CLV=Clive, HAN=Hanford, SRS=Savannah River Site
Water management (wells, reservoirs, dams)	1.4.10		Well supply (BMA 2.3.2.3.2, BMO 2.2.4.1.1)
			Dam building (BMA 2.3.2.2.4)
			Other water supply (BMA 2.3.2.3.3)
			Irrigation (BMA 2.3.2.3.4, BMO 2.2.3.1.2, CLV 83)
			Geosphere/Biosphere Interface (e.g., biosphere aquifer with well) (BMO 1.2.1)
			Environmental Components (e.g., well) (BMO 1.3.2.3)
			Water management (including construction of dams, reservoirs, canals, pipelines; and potential for flooding) (HAN 1.4.10, SRS 2.2.02, CLV 94)
			Flooding or drainage system failure (SRS 6.4.02)
Social and institutional developments[26]	1.4.11		Loss/degradation of societal memory (HAN 1.4.02.06)
			Loss of archives/records (HAN 1.4.02.07)
			Social and Institutional Developments (loss of records or society memory) (SRS 2.3.03)
Technological developments[27]	1.4.12		Technological development assumptions (e.g., including retrograde developments and no technological development) (HAN 1.4.14, SRS 2.3.04, SRS 2.3.05)
Remedial actions	1.4.13		Remedial actions (e.g., remediation, additional engineered barrier construction, waste retrieval) (HAN 1.4.13, SRS 3.8.05)
Explosions and crashes[28]	1.4.14	Long-term	Explosions and crashes (e.g., accidental human actions) (HAN 1.4.11, SRS 6.4.06, CLV 66, CLV 108)
Scope of FEP is limited. See Table C-7 for more information on what scope can be excluded from the FEP.			

[26] This FEP is limited in scope to include loss of records and societal memory. Other social and institutional developments may be excluded as they are difficult to predict. See Table C-7 for additional details.

[27] Several aspects of this FEP are excluded from consideration as they are difficult to predict and may be considered speculative, such as a cure for cancer. See Table C-7 for additional details.

[28] Acts of war, terrorism, and sabotage may be excluded.

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APPENDIX D
APPROACHES TO CONSTRUCT SITE SCENARIOS AND CONCEPTUAL MODELS

This appendix provides techniques for constructing site scenarios and conceptual models. Many of the techniques discussed are from examples not related to low-level radioactive waste (LLW) (NEA, 1992; IAEA, 2004; NRC, 1995; SKB, 2008); however, the U.S. Nuclear Regulatory Commission (NRC) staff believes the techniques and associated references are of value to a licensee that is looking for possible approaches to construct alternative site scenarios and alternate conceptual models. Licensees have the flexibility to use any technique and are not required to use the techniques presented in this appendix.

The techniques presented here provide a logical structure for the documentation of the relevant processes and their representation in conceptual models or site scenarios. The techniques are not mutually exclusive and may be used in combination. Whatever techniques are used, the judgment of subject matter experts is important to successfully completing this portion of the performance assessment process.

Event Tree Analyses

Event trees are one of the oldest techniques used to assess the operational safety of nuclear reactors. Probabilities can be systematically treated but the variations are mainly binary (fault or no fault). This method describes system behavior as an event or series of events leading to system performance or failure/loss of function. Application of the technique yields combinations of basic events whose occurrence causes system failure or loss of function. These event combinations are then evaluated by various screening techniques to determine the significance of scenarios. This technique is not used extensively in the context of radioactive waste disposal for several reasons: (1) most processes are generally slow and difficult to define as discrete events, and (2) the tree methods are not suitable to handle interaction and feedback between features, events, and processes (FEPs). In addition, the number of possible combinations in an event tree can very rapidly become unmanageable.

Logic Diagrams

Another technique to assist in scenario development is the use of logic diagrams. The development of site scenarios from combinations of the various release and transport phenomena is illustrated by the example in Figure D-1 (NRC, 1990). Two release phenomena (R1, R2) and three transport phenomena (T1, T2, T3) create 32 possible combinations or site scenarios. The use of a logic diagram, as illustrated in Figure D-1, ensures that all possible combinations of these phenomena are identified. The central scenario represents the initial conceptualization of the disposal system. All components of the engineered barrier system are assumed to perform as designed. The other site scenarios are perturbations to these central conditions (e.g., assuming less than 100 percent barrier performance).

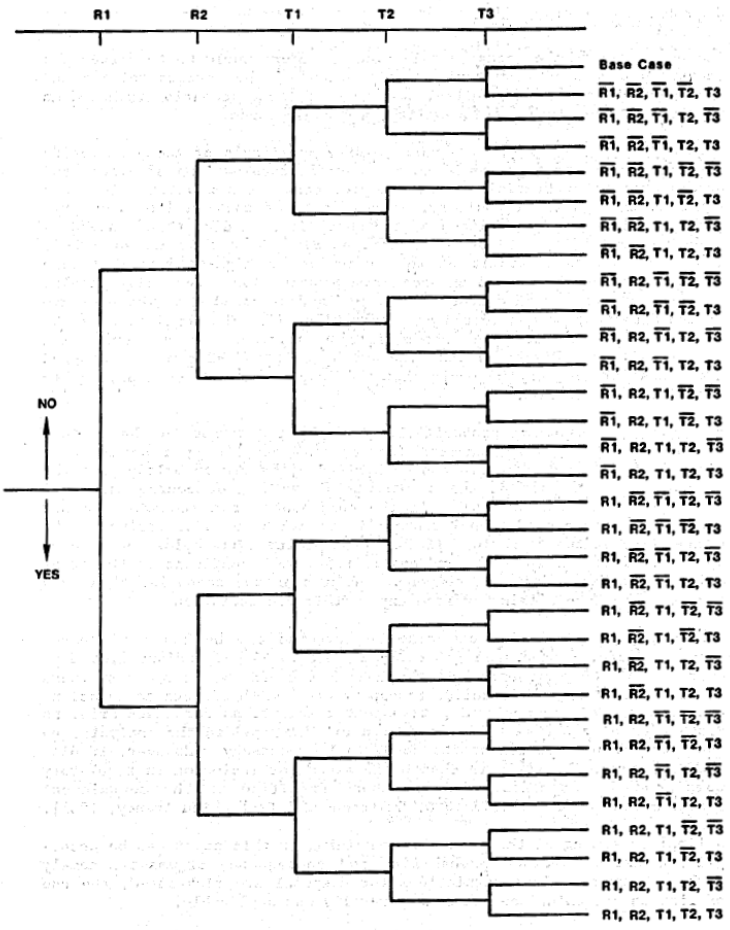


Figure D-1 Potential Combinations of Two Release and Three Transport Phenomena

Alternative site scenarios may be characterized by disruptive events (NRC, 2007). Severe droughts, floods, and introduction of irrigated agriculture are examples of discrete events affecting the hydrologic conditions at a site. These events are often not mutually exclusive (e.g., the occurrence of irrigated agriculture does not preclude the occurrence of a flood). By defining site scenarios as possible combinations of events, the site scenarios can be mutually exclusive. An example for three events is shown in Figure D-2. A "1" in the figure signifies the occurrence of the event in a site scenario and a "0" indicates the absence of that event. Scenario 1 in Figure D-2 has none of the events occurring and is characterized by the continuation of current hydrologic conditions into the future. In Scenario 1, the probability of climate change, flood, and irrigated agriculture not occurring is 0.7, 0.8, and 0.4 (or 1-p), respectively, so that the probability of the site scenario occurring is 0.7 x 0.8 x 0.4 or 0.224. For n events, this procedure will result in 2^n scenarios; some of these site scenarios may be eliminated from further consideration because of low probability or because the site scenarios are not of regulatory concern. Since scenario development for LLW disposal is typically qualitative and exact probability values for events characterizing site scenarios will usually not be available, qualitative science-based appraisals, such as large, medium, and small, would be workable substitutes. Numerical equivalents for high, medium, and low (e.g., 0.75, 0.5, and 0.25, respectively) could be used to derive estimates on the likelihood of a site scenario.

		Events Characterizing Scenarios				
		Climate Change (p=0.3)	Flood (p=0.2)	Irrigated Agriculture (p=0.6)		
Scenarios	1	0	0	0	0.224	Scenario Probability
	2	1	0	0	0.096	
	3	0	1	0	0.056	
	4	1	1	0	0.024	
	5	0	0	1	0.336	
	6	1	0	1	0.144	
	7	0	1	1	0.084	
	8	1	1	1	0.036	

Figure D-2 Example Formulation of Mutually Exclusive Scenarios from Three Scenario-characterizing Events

Interaction Matrices

The interaction matrix methodology starts with a top-down approach, dividing the system into constituent parts. The main components are identified and listed in the leading diagonal elements of the matrix. The interactions between the leading diagonal elements are then noted in the off-diagonal elements. The convention is to allocate off-diagonal elements in the direction of contaminant migration (see Table D-1 and Figure D-3). This allows FEP interactions and pathways to be mapped, which is an important step in developing and defining a conceptual model. Moreover, the systematic process of examining how the system components relate to one another may help to identify new, previously unrecognized relevant characteristics of the system.

When using a reference list of FEPs for populating the matrix, some processes may not be allocated to any of the defined off-diagonal elements. It might be necessary to subdivide some of the leading diagonal elements if these processes are considered important. Introducing more divisions of the leading diagonal elements will result in a more detailed matrix (Avila, 2012). Table D-1 provides an example of a simple interaction matrix.

Table D-1 Example of a Simple Interaction Matrix

Component A LEADING DIAGONAL ELEMENT 1,1	Influence of A on B OFF-DIAGONAL ELEMENT 1,2
Influence of B on A OFF-DIAGONAL ELEMENT 2,1	Component B LEADING DIAGONAL ELEMENT 2,2

IAEA (2004) describes this method in more detail (also see Figure D-3). The first step for constructing an interaction matrix is to identify and define its diagonal elements. This is done by exploring how the state of the system can be described in terms of physical components and spatial and temporal extension of the system. Usually, the diagonal elements represent system state variables (such as chemical composition of water or rock stress) and the off-diagonal elements represent processes affecting the state. For example, a typical diagonal element could be "groundwater composition," which in turn can be divided into concentrations of various constituents, colloid content, and other components.

As the diagonal elements of the matrix are filled in with features or components, the interactions between them are identified and introduced into the appropriate off-diagonal elements of the matrix. Interactions should be binary (i.e., direct interactions between variables in two diagonal elements and not a path through a variable in a third diagonal element). For interactions described in off-diagonal elements, the variable in the diagonal element on the row should affect the variable in the diagonal element on the column. Each off-diagonal element should be checked for interactions and marked "none" where no interaction is found.

In the process of identifying interaction, plausible interacting mechanisms are first considered without evaluating the probability of occurrence or the significance of the effect. Only irrelevant or unreasonable interactions should be discarded. For each off-diagonal element, the question is whether there are potential events or processes affecting the variables assigned to the target diagonal element (found in the column) that are at the same time affected by the variables in the source diagonal element (found in the row). If the answer is yes, a short description of the interaction should be documented, together with the variables in the two diagonal elements that are involved in the interaction.

In addition to the interaction matrix example, relevant information in IAEA (2004) can be found in Appendix B, "Generation of Scenarios for Near Surface Disposal Systems."

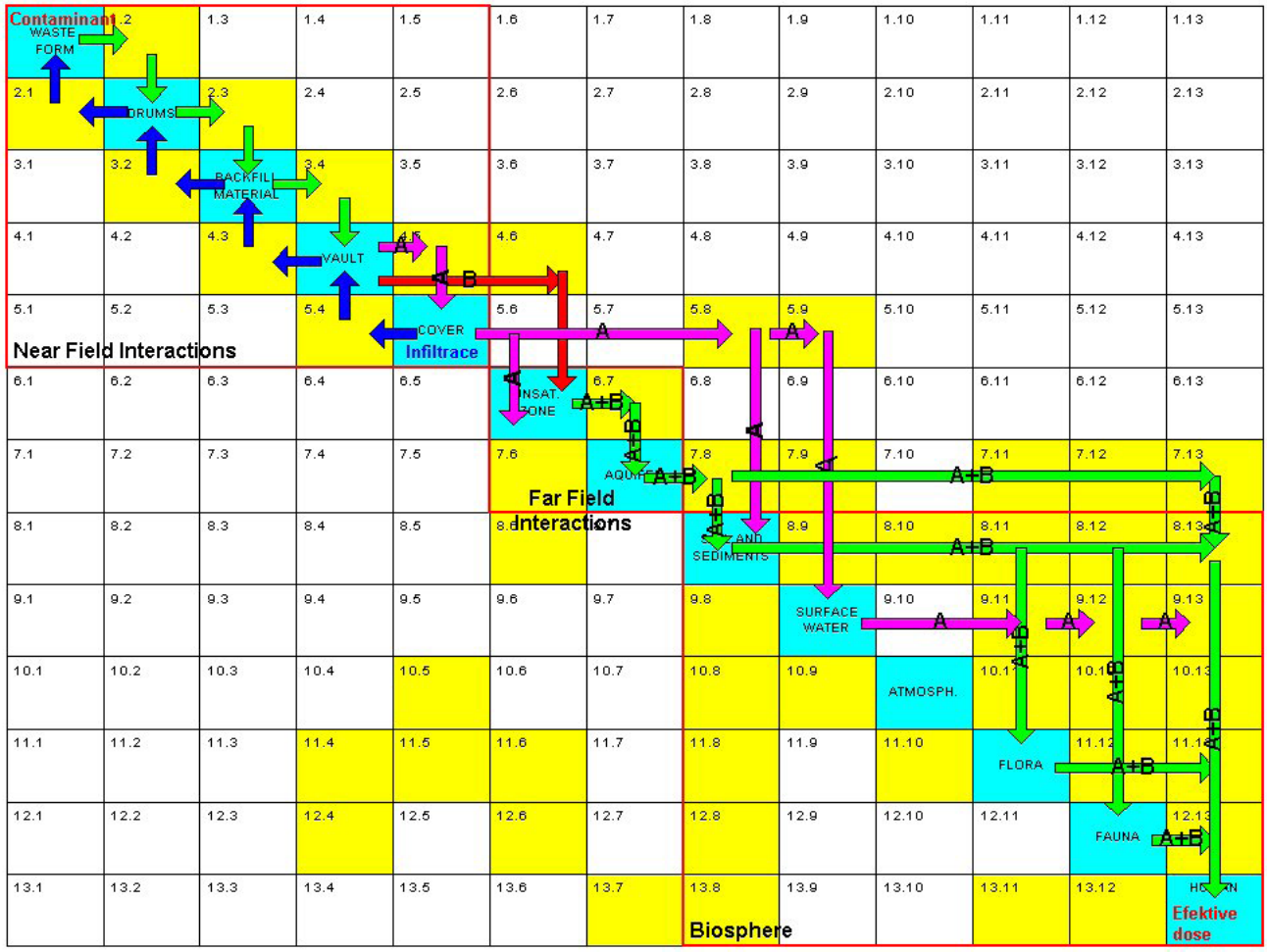


Figure D-3 Example of an Interaction Matrix for a Central Scenario Including the Bathtub Effect (IAEA, 2004)

Influence Diagrams

Influence diagrams are one of several methods to systematically evaluate and visualize FEPs that influence disposal facility performance. The objective of this method is to systematically identify and review FEP interactions and combinations of FEPs that can influence the performance of the disposal system. Each process is described as an influence between features or parameters (nodes) in the system. As the number of nodes rises, these influence diagrams tend to become complicated, but they are usually helpful.

In the influence diagram approach, arrows show the direction of the influence or the interactions between FEPs: one arrow per direct influence and one box per FEP. An example influence diagram is given by Chapman et al. (1995) for the deep repository performance assessment project in Sweden (see Figure D-4).

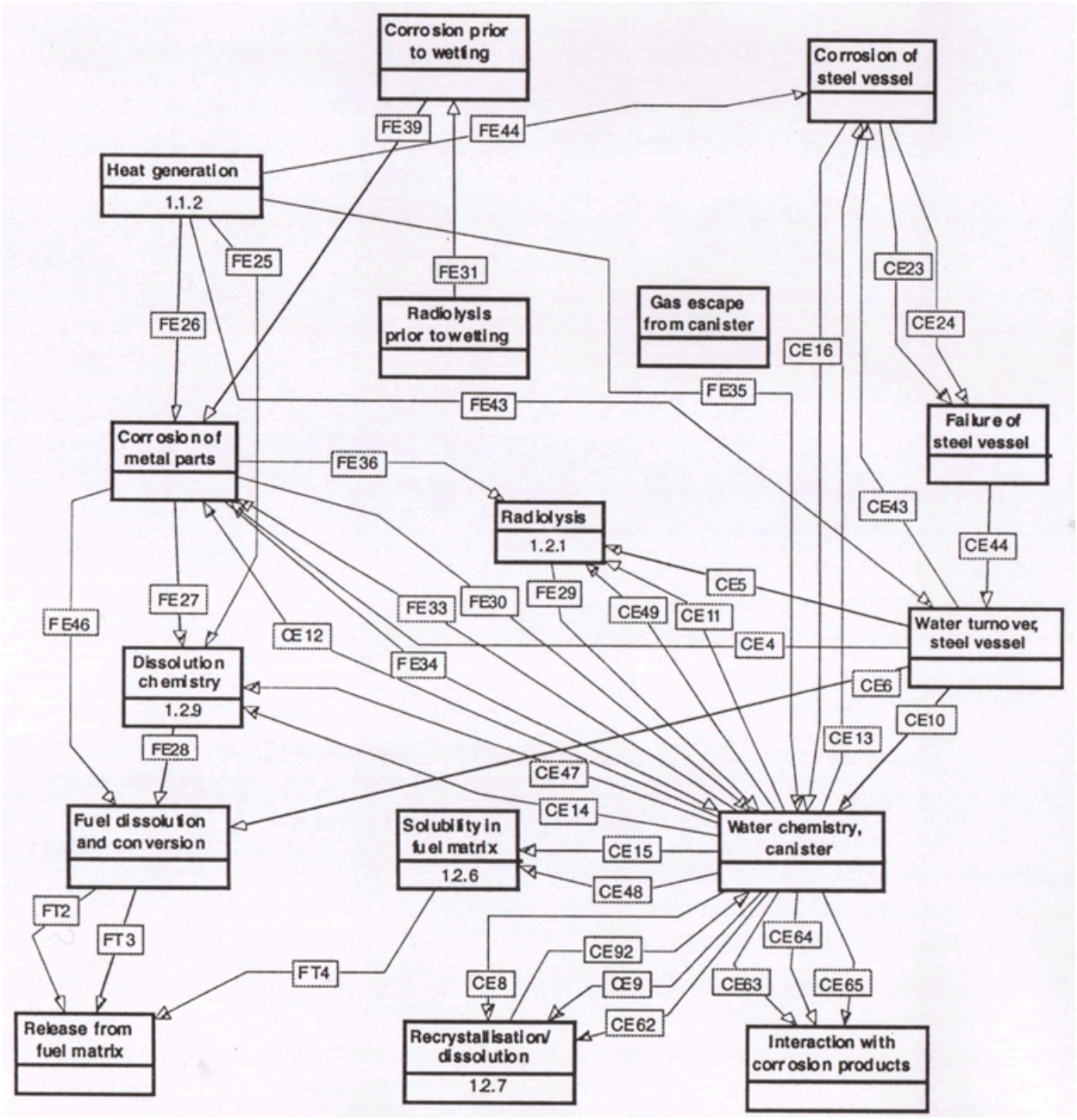


Figure D-4 Example of an Influence Diagram for Proposed Repository PA in Sweden. Many of the specific phenomena reflected in this diagram are not expected be relevant to LLW (Chapman et al., 1995)

The four main steps to building the influence diagram can be summarized as follows (IAEA, 2004):

(1) Definition of the system barriers and selection of FEPs relevant to the defined system. The FEPs can be sorted into FEPs belonging to the system and those external FEPs to the system that initiate a scenario.

(2) Representation of the system FEPs in boxes. If an FEP is relevant for several disposal components, then it should be represented by one box for each of the disposal components.

(3) Identification and representation of the influences between selected FEPs. Each influence in the diagram is marked with a unique code. There are no restrictions on the number of influences between two FEPs.

(4) Documentation of FEPs and influences. A more comprehensive description of each FEP and influence is needed to clarify the representation.

The influence diagram should not include large disruptive events that would alter the system features since this would produce a separate, alternative site scenario and would require a separate conceptual model. For the influence diagram approach, process-to-process influences should be avoided since it is assumed that the processes do not influence each other directly. Influences of this type should be broken in feature-process or process-feature influences. The development of influence diagrams is an iterative process. Two FEPs may be combined into one, and one FEP may be split into several FEPs to obtain an improved representation of the system. New influences between FEPs can be identified and classified using a significance scale. This can be used to build a reduced influence diagram by removing the influences with lower significance. Various influence diagrams can be developed based on the significance of the FEPs.

Judgmental Approaches

Scenarios can be formed in several ways and human judgment is an important element of scenario formation. After systematic screening, the number of retained FEPs is normally too large to allow detailed consideration of all possible combinations. There are, however, scenario formation procedures that allow the judgment and knowledge of qualified specialists to be integrated with quantitative considerations so that they result in a manageable number of representative site scenarios.

In the judgmental method, the assessment team members, or invited experts, select the phenomena or conditions that they believe are most important and define possible release situations. A list of phenomena or FEPs can be used as a starting point. Documentation and transparency should not be lacking for the sake of expediency; formal and rigorous scenario development procedures are encouraged. Documents of this procedure should be sufficiently detailed to provide pertinent information to the regulatory authorities and the public. Comprehensiveness and traceability in both the assessment and documentation are very important when using a judgmental approach.

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APPENDIX E
SITE STABILITY ASSESSMENT EXAMPLES

As discussed in Chapter 5.0, a licensee may use a design-based, model-based, or combined approach to site stability assessment. The combined approach incorporates elements of both the design-based approach and the model-based approach. The licensee may use modeling to assess and improve a design. In addition, it can use a risk-based approach to the site stability assessment; if modeling demonstrates that the risks are not acceptable, the licensee can modify the disposal site design to mitigate the risk.

The U.S. Nuclear Regulatory Commission (NRC) staff selected the West Valley Demonstration Project (WVDP) site as an example of a model-based approach[1] for this appendix because it is presently the only example of using landform evolution modeling to evaluate site stability in an NRC-regulated or NRC-reviewed waste disposal or site decommissioning application. Its use of the WVDP site example, however, does not indicate that the NRC is approving the approach. The U.S. Department of Energy (DOE) has not yet completed its work, nor has it selected its option for the final decommissioning of the WVDP site. However, the site stability assessment (modeling, data collection, development of analogs) for the WVDP site is reasonably representative of what may be expected for a low-level radioactive waste (LLW) disposal facility located in an unstable to moderately unstable site and, therefore, may be useful to some licensees under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 61, "Licensing Requirements for Land Disposal of Radioactive Waste." Chapter 5.0 provides more information on assessing shorter term stability issues involving subsidence and differential settlement and on longer term stability issues involving the potential effects of erosion, flooding, seismicity, and other disruptive processes and events.

This appendix also includes an example of a design-based approach from the Moab, Utah, uranium mill tailings disposal site. A licensee for an LLW disposal facility can use the selected examples to understand the type of information and actions involved in using the different approaches. A licensee should not use the information in these examples to justify site-specific stability assessments for LLW disposal.

[1] A model-based approach was presented in the DOE's final environmental impact statement (DOE, 2010) and decommissioning plan (DOE, 2009) and is discussed in this appendix, although more recent activities have emphasized use of aerial photography, light detection and ranging (LiDAR), and other data to estimate erosion rates (stream valley widening, gully propagation) with associated uncertainty estimates for use in performance assessment modeling.

Model-Based Approach: West Valley Erosion Modeling

Background

West Valley is a complex decommissioning site located in western New York State, about 50 kilometers (km) (30 miles) south of Buffalo (see Figure E-1). The New York State Energy Research and Development Authority (NYSERDA) holds the license and title to the 13.5 km^2 (3,300 acre) Western New York Nuclear Service Center (WNYNSC), originally developed as the first and only commercial spent fuel reprocessing plant to operate in the United States. The 1980 West Valley Demonstration Project Act gave the DOE exclusive possession of a 800,000 square meter (m^2) (200-acre) portion of the larger WNYNSC, which includes the former reprocessing facility, a land disposal facility, and high-level waste (HLW) tanks, to allow the DOE to carry out certain activities, most notably, the solidification of HLW that had been generated as a result of spent fuel reprocessing. This portion of the WNYNSC is referred to as the WVDP, or project premises.

In conjunction with NYSERDA, the DOE issued a draft environmental impact statement in 2008 evaluating various alternatives to decommissioning and long-term stewardship. The DOE and NYSERDA selected the phased decision-making alternative as the preferred alternative in the final environmental impact statement (FEIS) issued in 2010 (DOE, 2010). Under the phased decision-making alternative, decommissioning would proceed in two phases. Phase 1 involves near-term decommissioning work (e.g., removal of the main plant process building and contaminated subsurface soils) and studies that could facilitate future decision making for the remaining facilities and areas (see Figure E-1). A phased approach would allow additional time to address technical and programmatic uncertainties before making a final decision regarding site decommissioning. Consistent with the phased decision-making alternative, the DOE finalized the Phase 1 Decommissioning Plan for the West Valley Decommissioning Project in 2009 (DOE, 2009). The NRC reviewed the plan and found it to be acceptable (NRC, 2010). The DOE has been implementing the Phase 1 decommissioning plan and has decontaminated and decommissioned facilities at the site. A draft supplemental environmental impact statement is being prepared to support Phase 2 decision-making and is expected to be completed by late 2025. The DOE will also prepare a draft Phase 2 decommissioning plan for NRC review.

Decommissioning Site Characteristics

The WVDP is located on the west shoulder of a steep-sided, glacially scoured bedrock valley that is filled with a sequence of glacial sediments. These glacial deposits are comprised of clays and silts separated by coarser-grained layers created during periods of glacial retreat. The site is bordered by two streams, Franks Creek to the east and Quarry Creek to the north. The WVDP is bisected by Erdmann Brook, which divides the site into the North Plateau and the South Plateau. Franks Creek is a tributary of Buttermilk Creek. Figure E-2 shows site facilities and features, as well as source areas to be addressed in Phase 1 and 2 decommissioning. Figure E-3 shows site streams and a portion of the Buttermilk Creek watershed.

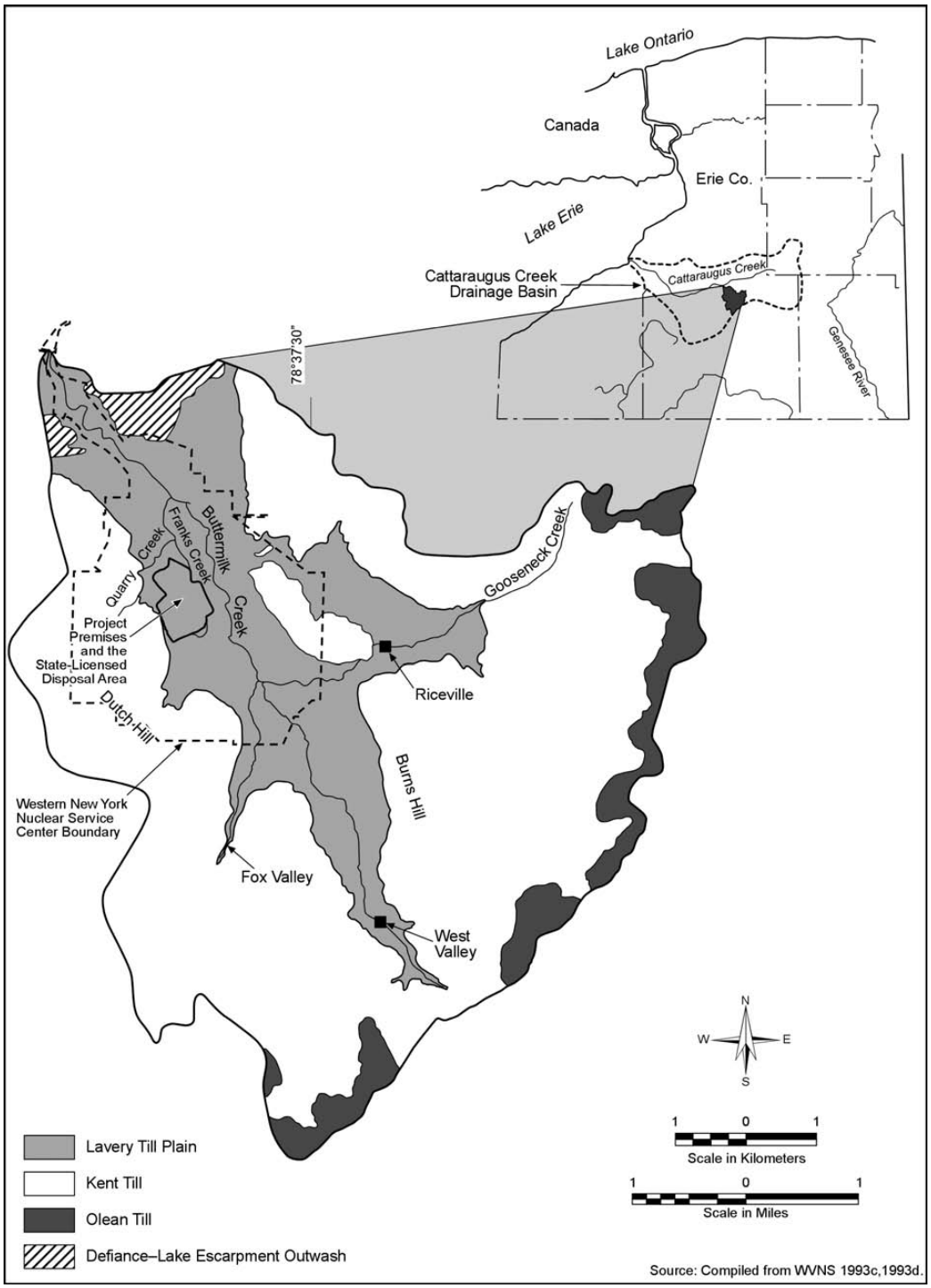


Figure E-1 Location of the West Valley Site
Image Credit: Figure F-2 (DOE, 2010)

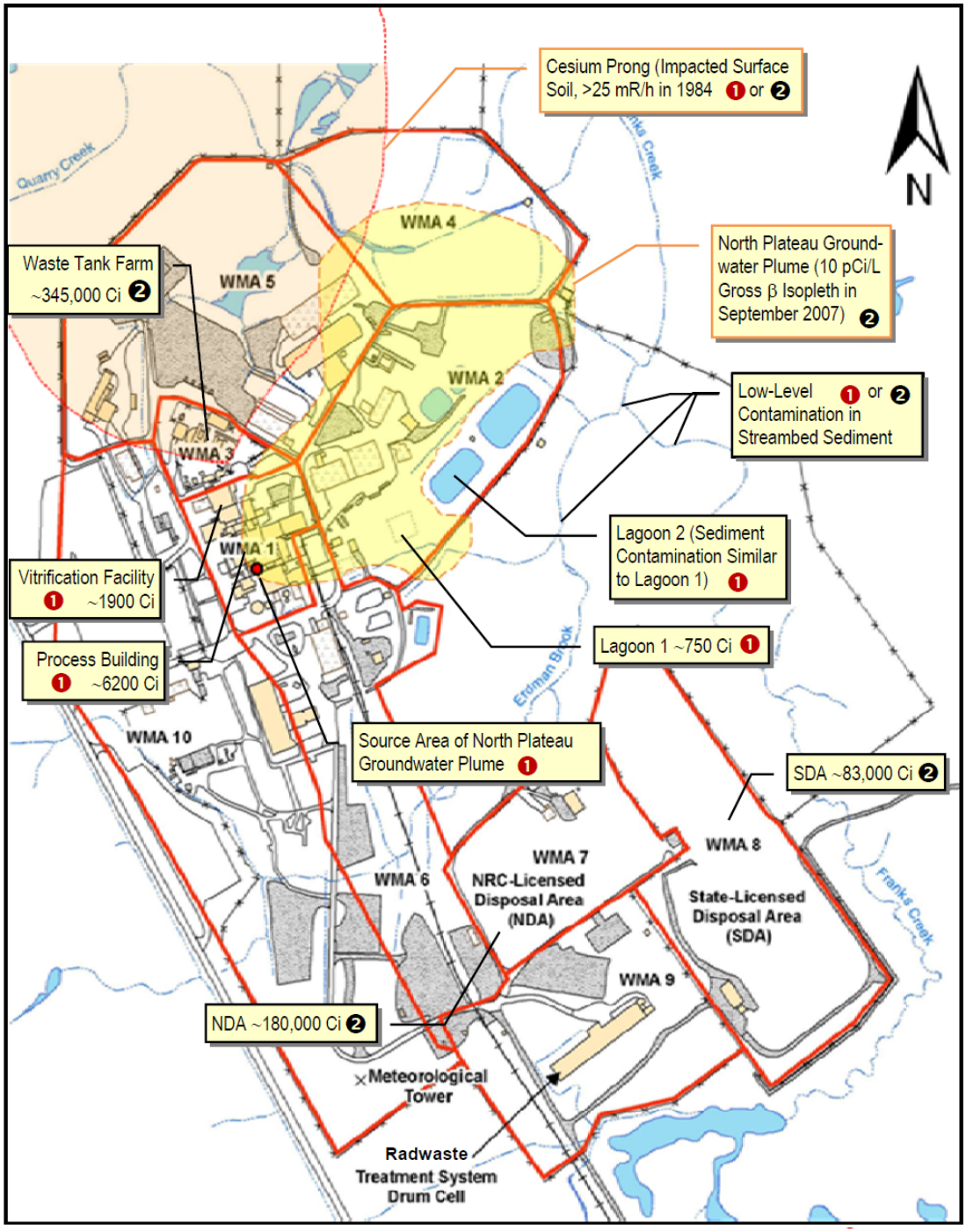
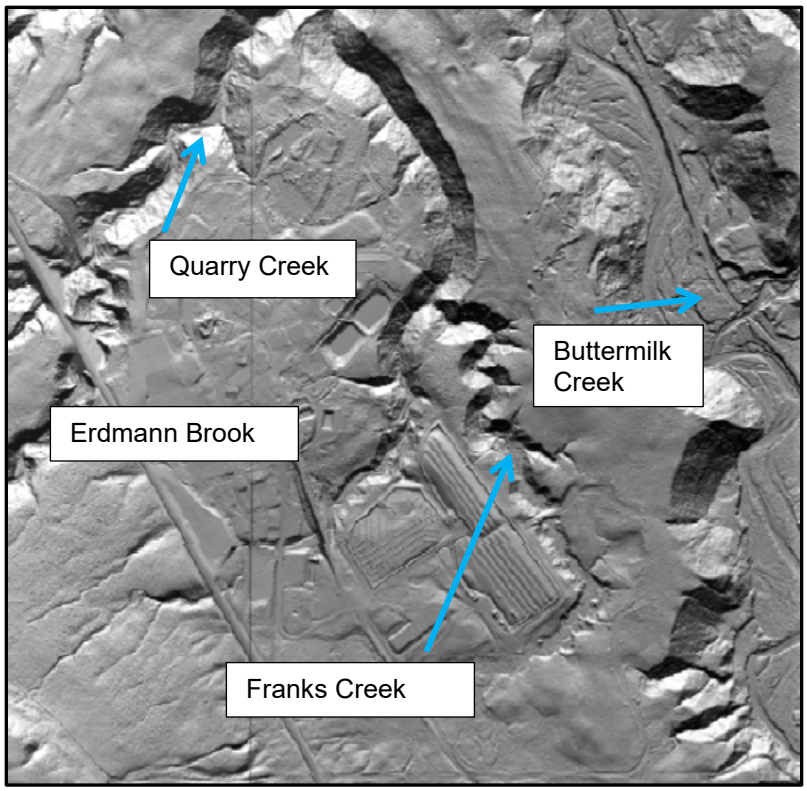


Figure E-2 Layout of the West Valley Site and Source Areas to Be Addressed in Phases 1 and 2

Areas to be addressed in Phase 1 are marked with a 1 symbol. Areas to be addressed in Phase 2 are marked with a 2 symbol.

Image Credit: Figure ES-5 (DOE, 2009)



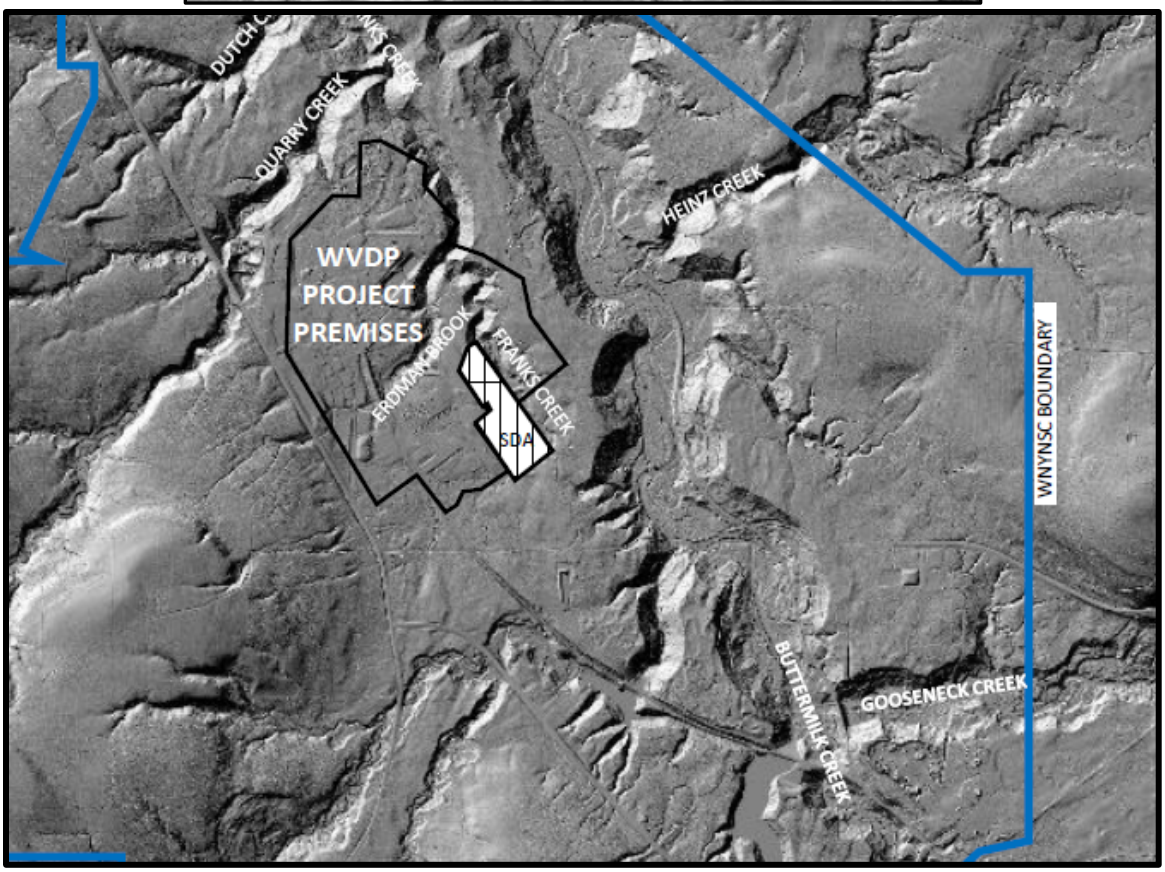


Figure E-3 West Valley Site Streams and Buttermilk Creek
Image Credit: Figure 4.2-2 (EWG, 2012a)

As a result of previous site operations, the WNYNSC and project premises soils, groundwater, and surface water/sediments are radiologically contaminated. Contamination includes what is referred to as the North Plateau Groundwater Plume that is characterized by high concentrations of relatively mobile and short-lived strontium (Sr)-90 (see Figure E-2). The North Plateau Groundwater Plume resulted from the accidental leak of radioactive nitric acid recovered from spent fuel reprocessing operations that traveled through a floor expansion joint into soils beneath the southwest corner of the Main Plant Process Building in 1968. The DOE installed a permeable treatment wall to remediate the North Plateau Groundwater Plume before risk-significant concentrations of Sr-90 could seep or discharge to surface water. In Phase 2, the DOE and NYSERDA will make decisions related to closure of four tanks used to store liquid HLW, two radioactive waste disposal facilities, and clean-up of other radiologically contaminated areas onsite (see Figure E-2).

Decisions regarding the amount of residual radioactivity that may safely remain in contaminated surface and subsurface soils, sediments, HLW tanks, treatment lagoons, and disposal areas are complicated by several technical and programmatic uncertainties that must be addressed before making a final decision regarding the disposition of the site. For example, one of the key technical uncertainties is the expected evolution of the landscape of the actively eroding site and the performance of engineered barriers used to minimize or mitigate the release of residual radioactivity to the environment. Phase 1 studies were proposed to address these technical uncertainties.

Key Site Stability Technical Issue: Erosion

As stated above, erosion is a key technical issue affecting West Valley site stability. Major erosion processes affecting the WNYNSC, including the WVDP, include stream channel downcutting, stream valley rim widening, gully advance, and, in disturbed areas, sheet and rill erosion. Figure E-4 shows site erosion features, while Figure E-5 shows Buttermilk Creek watershed erosion features. The FEIS modeling simulated development of the current topography and stream drainage patterns, beginning with the glaciation and retreat process that ended approximately 17,000 years ago. Erosion processes in the Buttermilk Creek watershed have affected site topography. Buttermilk Creek flows in a northwesterly direction along the central axis of the WNYNSC at an elevation of approximately 60 m (200 feet (ft)) below the plateau on which the WVDP is located (see Figures E-1 and E-3). At the WVDP, Franks Creek flows along the eastern boundary and drains to Buttermilk Creek. Franks Creek downcutting rates reflect base level lowering of Buttermilk Creek at the confluence of Franks Creek and Buttermilk Creek. Buttermilk Creek downcutting rates are, in turn, affected by base level lowering of Cattaraugus Creek at the confluence of Buttermilk and Cattaraugus Creeks.

The FEIS (DOE, 2010) included Channel Hillslope Integrated Landscape Development (CHILD) long-term erosion modeling predictions to evaluate the impact of erosion on site performance and the ability of various proposed alternative end states to meet decommissioning criteria. The FEIS also provided details regarding the results of numerous other studies (e.g., erosion frame measurements, age dating of terraces to estimate stream down-cutting and stream valley rim-widening rates, aerial photography comparisons to estimate gully migration, and various short-term modeling exercises) that were used to evaluate the reasonableness of CHILD modeling predictions.



Erdmann Brook Knickpoint



Franks Creek Knickpoint



WVDP Site Gully

Figure E-4 WVDP Erosion Features



Buttermilk Creek Landslide

Figure E-5 Buttermilk Creek Watershed Erosion Features

West Valley CHILD Model

The model domain for the CHILD model used in the FEIS consisted of the Buttermilk Creek watershed area (see Figure E-6). Prior to predictive modeling, the CHILD model was calibrated against site data. Calibration modeling simulations began approximately 17,000 years ago using remnant terrace elevations and valley slope data to estimate the post-glacial (pre-incised) Buttermilk Creek watershed surface. To minimize the "butterfly effect" in which small perturbations in the initial conditions lead to notable differences in simulated drainage patterns, the existing drainage network was etched on the initial topography. Boundary conditions included the elevation history of Cattaraugus Valley at the outlet of Buttermilk Creek. As a modeling simplification, and in the absence of more detailed information about climate variation over the past 17,000 years, a constant climate was assumed. Three primary material types were selected, including (1) Paleozoic bedrock, (2) thick but unlithified glacial sediments, and (3) shallow surface soils/sediments. Five discrete values that reflect the range of each parameter value were assigned based on site-specific measurements (e.g., water balance data), site-specific material correlations (e.g., parameters related to detachment capacity), values taken from the literature for similar or analog sites (e.g., creep coefficients), or commonly accepted values (e.g., critical slope parameter). Selected calibration metrics reflected characteristics of the present-day Buttermilk Creek watershed, including the (1) creek longitudinal profile, (2) hypsometric curve (area below a certain elevation), (3) slope-area diagram (gradient versus upstream contributing areas), (4) width function (frequency distribution of catchment flow-path length), (5) cumulative area distribution (rate of flow aggregation), and (6) strath terrace positions (pass/fail terrace elevation criterion considering measurement uncertainty).

Monte Carlo methods were used in the calibration process. One thousand sets of parameter values were generated by sampling the discrete parameter distributions. These 1,000 sets of parameter values were run through the Buttermilk Creek watershed model described above. The results of 1,000 simulations were evaluated against the calibration metrics. The scores for the first five calibration metrics were normalized to allow equal weighting of each calibration metric for subsequent averaging. Four final calibration criteria were established to select the most likely sets of parameters to be used in forward model projections: (1) the total average, normalized score must be greater than 0.5 (considering the first five calibration metrics listed above), (2) the longitudinal profile score, by itself, must be greater than 0.7, (3) the intermediate strath terrace elevation metrics should be met within a given tolerance and within a given time span, and (4) visual agreement between the model simulation results and the current topography should be achieved. Only five runs passed the four calibration criteria, with the five sets of parameter values subsequently used in forward modeling projections to predict future erosion at the WVDP.

The observed topography and "best-fit" model run topography are presented in Figure E-6. Sensitivity runs were conducted that considered (1) wetter climates and less permeable soils and (2) wetter climate parameters with fast creep (for the South Plateau only). The forward modeling projections used a digital elevation model of the current topography with a 10 m (33 ft) resolution. Additionally, a second set of model simulations were performed to estimate erosion for the Sitewide-Close-In-Place alternative, in which three burial mounds would remain on the North and South Plateaus. The modeling grid was refined in the North Plateau and South Plateau with a grid resolution of approximately 3 m (10 ft) to facilitate simulation of smaller scale erosion features such as gullies.

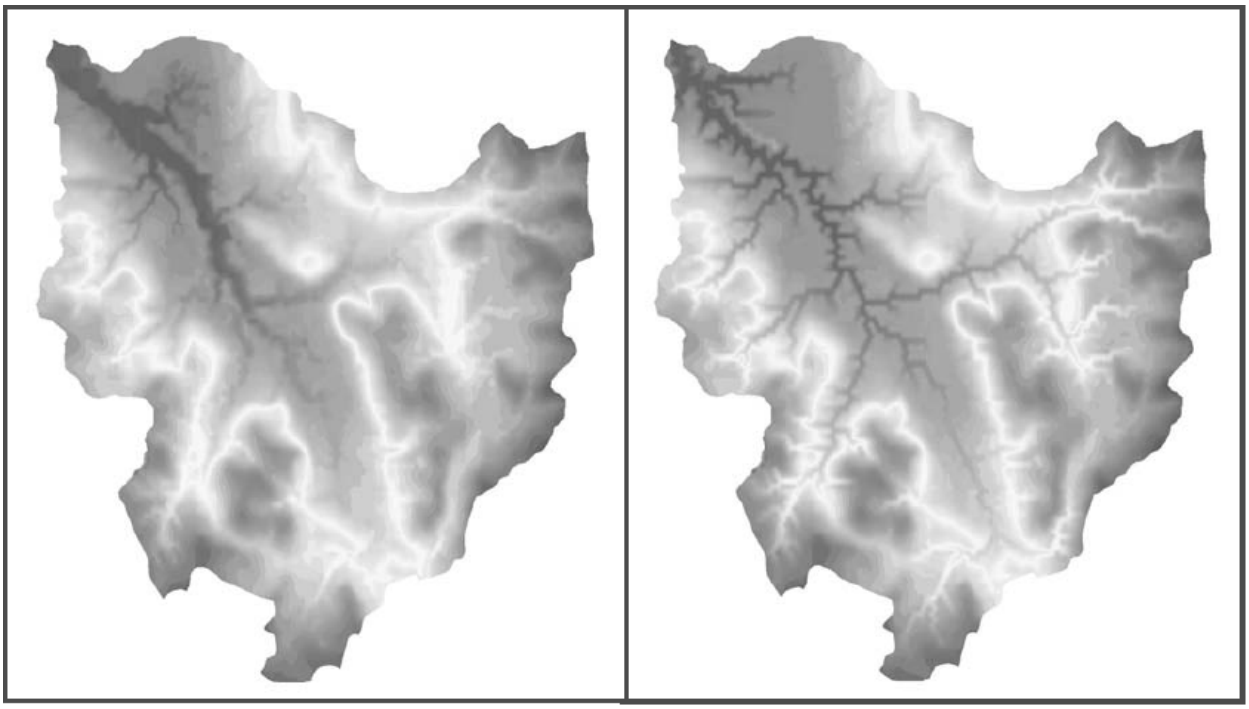


Figure E-6 Observed (left) and "Best-fit" (right) Model Run Topography
Image Credit: Figure F-8 (DOE, 2010)

Due to the computation demands of a finer grid resolution, only one of the plateaus at the finer grid resolution could be modeled at a time, leading to two separate models reflecting the mesh refinements on the North and South Plateaus. With respect to boundary conditions, the final base level lowering rate from the corresponding calibration run was applied at the outlet of Buttermilk Creek. The results of the 26 modeling simulations were presented and discussed in the FEIS. The FEIS modeling results provided a reasonable approach to evaluating erosion impacts at the WVDP. However, the DOE and NYSERDA acknowledged limitations of the modeling and recognized several areas where additional information could be collected to improve and refine erosion predictions for future Phase 2 decision-making.

Additional Studies

The DOE and NYSERDA commissioned an erosion working group (EWG) to develop recommendations for studies that could be conducted during Phase 1 to further reduce uncertainty associated with the impact of erosion on site performance (EWG, 2012a; EWG, 2012b). The purpose of the additional studies would be to (1) fill data gaps, (2) produce converging lines of evidence, (3) improve scientific defensibility, and (4) strengthen confidence in long-term erosion projections.

One of the Phase 1 studies performed in the 2015 to 2016 timeframe resulted in the EWG producing a Terrain Analysis, Age Dating, and Paleoclimate report (EWG, 2018a). The objectives of the study were to (1) establish a more precise timing of the last ice sheet recession, (2) establish the sequence of major or identifiable events in postglacial time, (3) better understand the past history of postglacial erosion and its relation to the prediction of future erosion, (4) investigate relations between paleoclimate and erosion rates, and (5) recommend how these factors should be incorporated into predictive erosion models. As part of this effort, the EWG reviewed information from the literature, reports and data; analyzed

maps, satellite imagery, aerial photography, and detailed LiDAR maps. They also conducted walkdowns and sampling for carbon (C)-14, performed optically-stimulated luminescence and terrestrial cosmogenic nuclide dating; and used ground penetrating radar surveys. Additionally, they performed geologic logging, sampling and photography of exploratory pits and trenches. As a result of the study, more definitive dates for the last glacial recession, sediment yields from the Buttermilk and Connoisarauley watersheds, and changes in and the nature of the Buttermilk Creek incision and its rates were estimated. DOE and NYSERDA gained a better understanding of changes in climate and erodibility of materials, as well as insights on historical and potential future erosion mechanisms from the EWG's activities.

The EWG evaluated uncertainty in erosion estimates and the impact of uncertainty on site and facility performance in EWG (2018b), incorporating new data and information collected since the FEIS was prepared. Additionally, the EWG analyzed 37 models for long-term erosion to incorporate model as well as parameter uncertainty in the evaluation. The EWG calibrated each model to identify an optimal match between observed and simulated modern topography. Based on the results of the calibration, the EWG selected nine models for future study. Sources of uncertainty include future climate, Buttermilk Creek valley downcutting, model structure, land use, model structure, and model parameters. The EWG provided results in the report, including a series of maps and digital data files that depict the projected erosion depth and associated uncertainty at different locations across the watershed. These results show that the greatest projected future erosion is associated with major valleys and side slopes and with gullies that drain into those valleys. The EWG projected the lower portion of Quarry Creek (downstream of Rock Springs Road), and the reach of Franks Creek downstream of its confluence with Erdman Brook to be the greatest area of erosion. They also projected the margins of the south plateau to be susceptible to erosion from valley widening along upper Franks Creek and Erdman Brook.

Recent erosion work has focused on review of orthorectified historical aerial photography (as far back as 1939) which is paired with LiDAR images to estimate the location and rates of gully migration and hillslope failure. Erosion rates and associated uncertainty estimates (e.g., uncertainty based on interpretation of historical imagery) are being used to support the probabilistic risk assessment being prepared to assist with Phase 2 decision-making.

How Does the West Valley Erosion Modeling Example Relate to 10 CFR Part 61 Guidance?

Although the West Valley erosion modeling example described above applies to a decommissioning site and has not been fully executed by the DOE and NYSERDA for NRC review, the example illustrates many of the steps listed in Section 5.2.2 of this document, and reproduced below for ease of reference, that may be taken to assess performance of a 10 CFR Part 61 facility.[2] Because future LLW disposal facilities will be sited according to the siting characteristics requirements found in 10 CFR 61.50, "Disposal site suitability requirements for land disposal," this example represents a likely endpoint, in terms of difficulty and complexity, for a model-based approach to site stability assessment.

[2] Inclusion of this example does not provide any tacit or implied approval of the erosion modeling as a basis for demonstrating compliance with the radiological criteria for license termination. Such determinations will be made at the appropriate time following review of the Phase 2 decommissioning plan. The example is provided because the current approach nicely illustrates the components of a model-based stability assessment.

These steps include the following:

(1) Define model objectives.
(2) Develop or select the model.
(3) Document and provide the basis for assumptions.
(4) Parameterize the model.
(5) Calibrate the model.
(6) Verify the model.
(7) Characterize uncertainty.
(8) Provide model support.
(9) Iterate, if necessary.

The DOE has performed data collection and modeling of erosion at West Valley in an iterative manner and has continued to improve erosion modeling over time as it has obtained additional information. The DOE's code selection process began with the initial evaluation of various shorter-term, smaller scale models that evaluated one or two key erosion processes operable at the WNYNSC and WVDP. These codes were limited in their ability to simulate multiple, coupled erosion processes over larger time and spatial scales. The DOE also used sophisticated landscape evolution modeling using codes such as SIBERIA in earlier environmental impact analyses. These modeling exercises led to the selection of the CHILD model that was considered by the DOE to be the state-of-the-art in landscape evolution modeling at the time of the FEIS. Due to the complexity of the site, landscape evolution modeling was considered necessary to provide technically defensible erosion modeling predictions to facilitate decision-making.

The DOE used site-specific data and measurements, as well as information from the literature on analog sites and generic information sources, if more relevant data were not available, to assign parameters for CHILD modeling. The DOE used calibration to identify the most likely set of parameter values to predict future erosion at the site. The DOE also used site data and other shorter-term modeling results to evaluate the reasonableness of the CHILD modeling results. The DOE performed Monte Carlo analysis to evaluate the impact of parameter uncertainty on the results of the analysis and attempted to reduce uncertainty in the erosion predictions through selection of "best-fit" parameter values in the calibration process that produced current-day topography. The DOE performed sensitivity analysis to evaluate "what-if" site scenarios such as a wetter climate or faster creep coefficients and documented the results of its erosion analyses in the FEIS including a discussion regarding potential model limitations. The DOE also provided information on areas for potential improvement, including the collection and use of additional data or additional analyses that might be conducted as code capabilities matured. The WVDP EWG recommended Phase 1 studies to further reduce uncertainty and provide additional support for erosion modeling predictions. Finally, the DOE collected additional data that are being used to support future performance assessment modeling.

As indicated in Section 5.2.2.2, either a model-based or a design-based approach may be used to evaluate site stability. Typically, model-based approaches are used for longer periods of performance, while a design-based approach may be acceptable for shorter periods of performance. A hybrid approach that combines both model- and design-based approaches may be used for assessments of site stability and could help provide multiple lines of evidence that the performance objectives in 10 CFR Part 61, Subpart C, "Performance Objectives," will be met. For example, a licensee may provide support for the performance of a design that mitigates the risk of relatively short-lived waste. Over longer time periods, the performance of the design may be less certain, and those design features relied on for short-term performance

may no longer mitigate impacts of erosion. Licensees could perform sensitivity analyses to analyze the impact of various levels of underperformance of the design over time in order to evaluate the acceptability of the design and the ability of the site to meet performance objectives. This would lend confidence to the compliance and performance period analyses.

Design-Based Approach: Moab, Utah, Example

Background

The Moab Mill was a uranium milling site that operated from 1956 until 1984 and contains mill tailings from uranium-ore processing. The DOE is currently remediating the Moab under Title I of the Uranium Mill Tailings Radiation Control Act of 1978 (UMTRCA). Moab is in Ground County, Utah, on the northwest shore of the Colorado River (see Figure E-7). The Uranium Reduction Company (URC) built and operated the Moab Mill beginning in October 1956. Atlas Corporation acquired the URC in 1962 and operated the mill until 1984, when it was placed in stand-by status. Atlas submitted decommissioning and reclamation plans for the site, which the NRC approved in 1988 and 1999, respectively. In 1998, however, Atlas filed a petition for relief under Chapter 11 of the US Bankruptcy Code. In October 2000, the Floyd D. Spence National Defense Authorization Act for Fiscal Year 2001 amended UMTRCA Title I, giving the DOE responsibility for remediation of the Moab site. The NRC license was terminated, and the title to the site was transferred to the DOE in October 2001.

The remedial action plan proposed by the DOE consisted of removal and subsequent relocation of the residual radioactive material and contaminated materials from the Moab site to a disposal cell at Crescent Junction, Utah (also shown on Figure E-7). The DOE proposed to construct an approximately 100-hectare (250-acre) engineered cell, partially below grade, to contain the material. An engineered cover will limit radon migration from the cell, protect the cell from erosion, and reduce infiltration.

Moab Site Characteristics

The Crescent Junction Site is in the north end of the Canyon Lands section of the Colorado Plateau physiographic province. The Canyon Lands section is characterized by deeply incised drainages, isolated mesas, gently dipping bedrock, and anticlines formed by salt intrusion that have been breached by erosion in places to form anticlinal valleys. North of the Canyon Lands section is the Uinta Basin section of the Colorado Plateau; the boundary between the two sections is the Book Cliffs, an erosional escarpment just north of the site. The Uinta Basin section is characterized by a rugged, intricately dissected plateau bounded on the south by sets of cliffs (one of which is the Book Cliffs) that are highly irregular, with many salients and canyons.

The Crescent Junction site is situated in the Mancos Shale Lowland at the foot of the Book Cliffs, a broad, gently sloping area that is underlain by several thousand feet of Mancos Shale (see Figure E-8). The thickness of the Mancos Shale is approximately 730 m (2,400 ft) based on a test hole drilled approximately 530 m (1,750 ft) south of the disposal cell location and on published geologic maps and literature of field and geologic mapping of the area. Ten cores were drilled to depths of approximately 90 m (300 ft) into the Mancos Shale. Core samples were logged using visual soil- and rock-classification procedures, and the cores were geophysically logged.

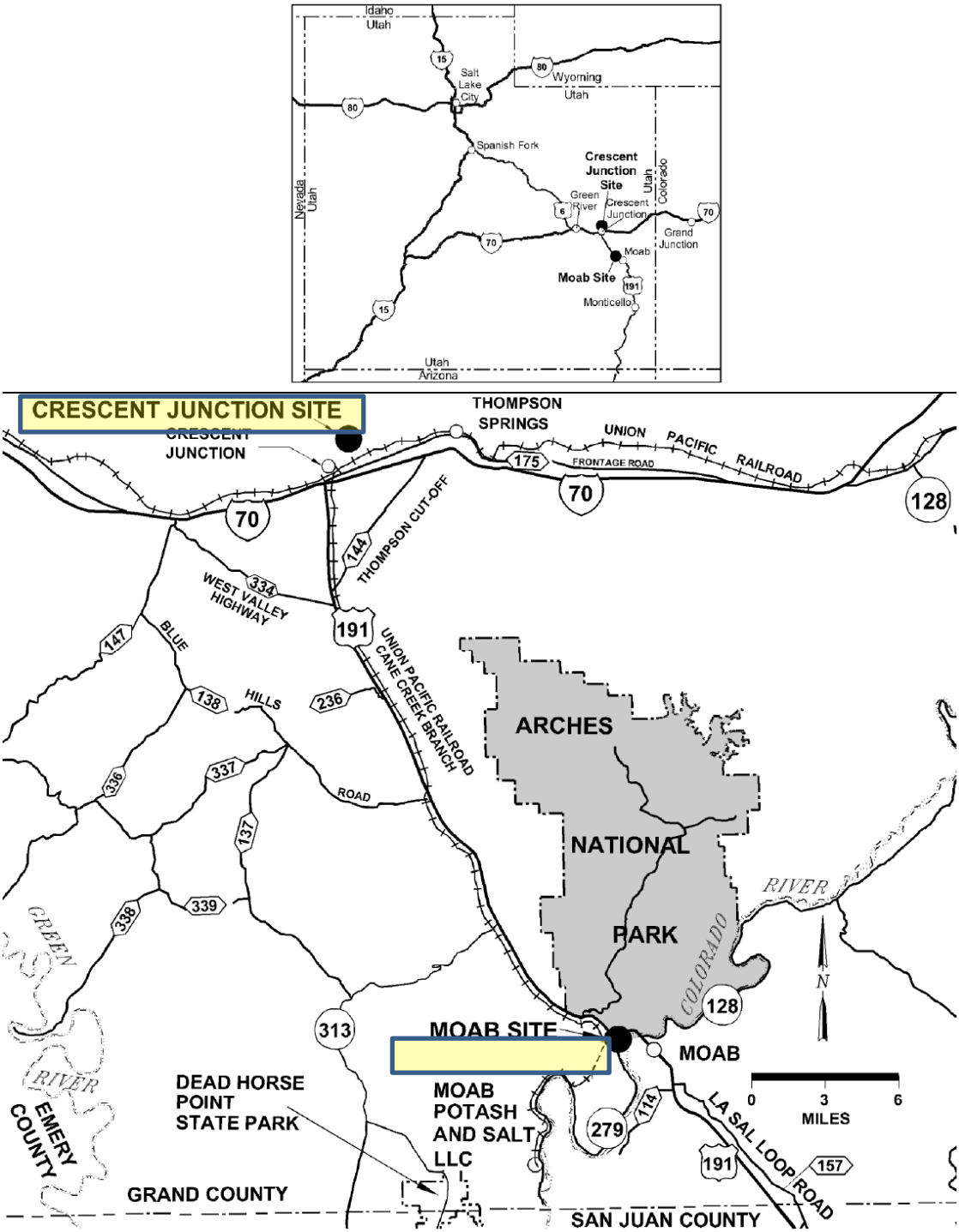


Figure E-7 Location of Moab and Crescent Junction Sites
Image Credit: Figure 1-1 (DOE, 2008)

The DOE differentiated the Mancos Shale into "weathered" and "unweathered" portions. The primary difference between the weathered and unweathered Mancos Shale is the spacing of natural bedding plane fractures and high-angle fractures, both of which decrease or are non-existent with depth in the unweathered Mancos Shale. This transition occurs at approximately 16 m (50 ft) below the top of the Mancos bedrock surface.

Overlying the Mancos Shale is approximately 6 to 9 m (20 to 30 ft) of alluvial and outwash material that has originated locally from the Book Cliffs. The DOE drilled 100 geotechnical boreholes to depths as much as 8 m (26 ft) through the surficial unconsolidated material into the shallow weathered Mancos Shale. Samples were logged in the field. Additionally, the DOE dug five test pits 5 to 7 m (15 to 23 ft) deep that assisted with the geologic characterization and allowed hydraulic and geotechnical testing of the weathered Mancos Shale.

Underlying the Mancos Shale is the Dakota Sandstone containing the uppermost aquifer. The Dakota Sandstone is considered the uppermost aquifer because the Mancos Shale has an extremely high-saline content and poor quality and quantity of water. Figure E-8 illustrates the hydrostratigraphic system at Crescent Junction.

Key Technical Review Area: Erosion Protection

The NRC staff reviewed the DOE's remedial action plan against the U.S. Environmental Protection Agency requirements presented in 40 CFR Part 192, "Health and Environmental Protection Standards for Uranium and Thorium Mill Tailings." To evaluate the DOE's remedial action plan, the NRC staff used Section 3.0 of the "Final Standard Review Plan for the Review and Remedial Action of Inactive Mill Tailings Sites Under Title I of the Uranium Mill Tailings Radiation Control Act," Revision 1, issued June 1993 (NRC, 1993). Covered review areas include estimates of flood magnitudes, water elevations and velocities, sizing of riprap to be used for erosion protection, long-term durability of erosion protection components, and testing and inspection procedures to be implemented during construction. A description of the DOE's design approach (DOE, 2008) and summary of the NRC's review (NRC, 2008) is provided below.

Hydrologic Description and Erosion Protection Design

Regulations in 40 CFR Part 192 require stability of the tailings for 1,000 years to the extent reasonably achievable and in any case for 200 years. In its remedial action plan, the DOE proposed to construct a disposal cell to protect the contaminated material from flooding and erosion. The probable maximum precipitation (PMP) and the probable maximum flood (PMF) events, both of which are considered to have very low probabilities of occurring during the 1,000-year stabilization period, were both considered as part of the design basis.

The top surface of the disposal cell was designed to drain in multiple directions at a slope of about two percent, and a vertical to horizontal slope of 1:5 (20 percent) was integrated into the design on the embankment side. Rock riprap would be used on the top and side slopes to protect against erosion. At the toes of the side slopes, rock riprap aprons would be constructed to protect against gully advancement toward the disposal cell. Several drainage channels would convey flood flows off the disposal cell and away from the disposal area. Figure E-9 illustrates the dimensions, slope, and flow directions associated with the disposal cell.

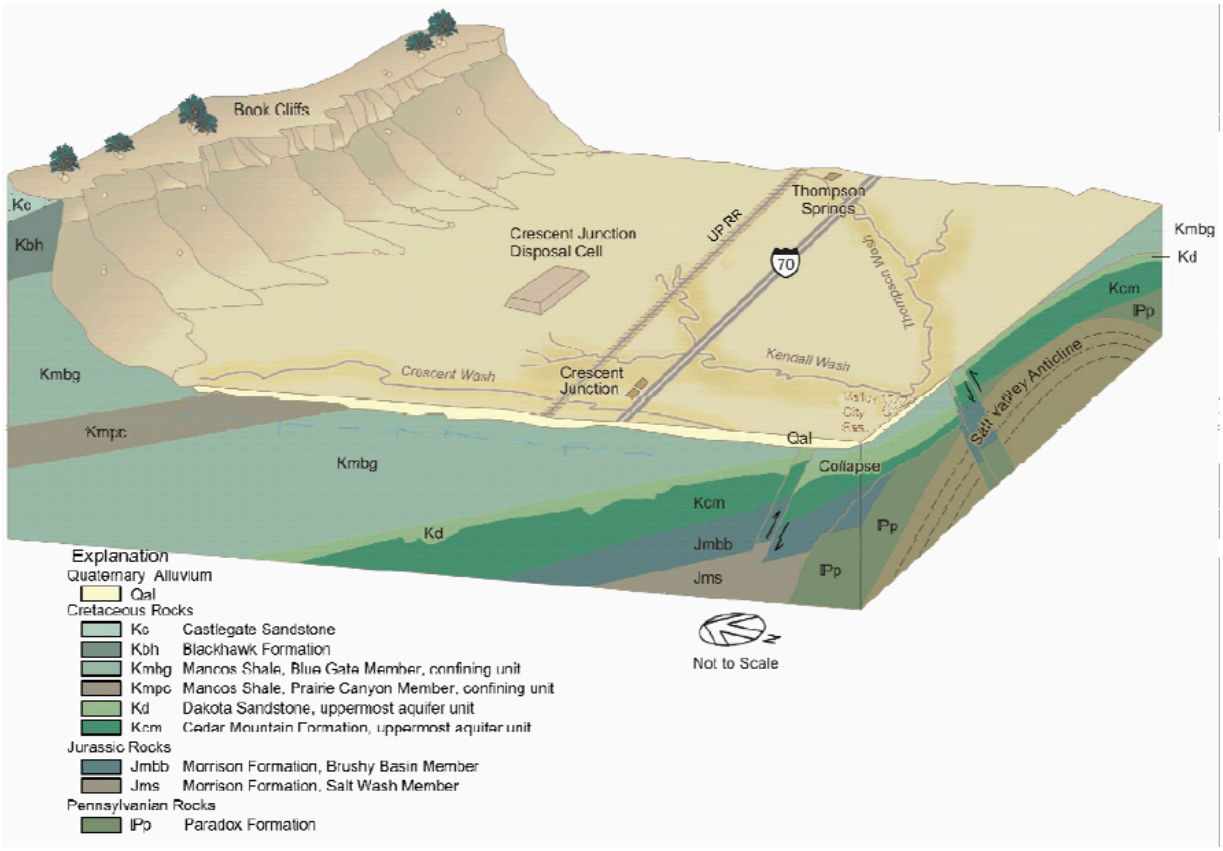


Figure E-8 Schematic of Hydrostratigraphic System and Topographic Features at the Crescent Junction Site
Image Credit: Figure 3-1 (DOE, 2008)

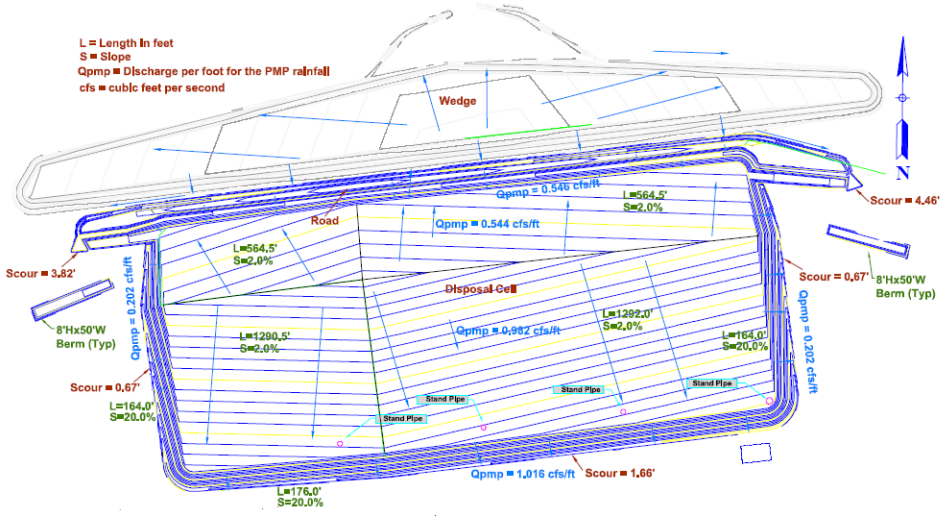


Figure E-9 Schematic of the Disposal Cell Dimensions, Slopes, Flows, and Scour Depths
Image Credit: Figure 6-3 (DOE, 2008)

The DOE computed peak flood discharges for various site design features in five steps:

(1) selection of a design rainfall event

(2) determination of infiltration losses

(3) determination of times of concentration

(4) determination of appropriate rainfall distributions and intensities, corresponding to the computed times of concentration

(5) calculation of flood discharge

Input parameters derived from each of these steps were then used to calculate the peak flood discharges to be used in the final determination of rock sizes for erosion protection.

One of the phenomena most likely to affect long-term stability of the disposal site is surface water erosion. To ensure adequate protection against surface water erosion, the flood protection design should be based on conservatively selected events. The DOE used a PMP event as the design basis. The PMP was computed using deterministic rather than statistical methods and was based on site-specific hydrometeorological characteristics. The PMP has been defined as the most severe reasonably possible rainfall event that could occur from a combination of the most severe meteorological conditions occurring over a watershed. No recurrence interval is assigned to the PMP; however, the probability of such an event being equaled or exceeded during the 1,000-year stability period is very low. Accordingly, the PMP is generally considered by the NRC staff to provide an acceptable design basis.

Hydrometeorological reports developed for specific regions can be used to estimate the PMP. This technique provides a straightforward procedure with minimal variability. The DOE estimated PMP values using Hydrometeorological Report No. 49, “Probable Maximum Precipitation Estimates, Colorado River and Great Basin Drainages,” issued 1977 (NOAA, 1977). The DOE used a 1-hour PMP of 21 centimeters (cm) (8.2 inches (in.)) as a basis for estimating PMFs for small areas at the site such as the top and side slopes. The NRC staff concluded that the sources used to estimate PMP values and the PMP values themselves were acceptable (NRC, 2008).

The peak runoff rate is dependent on the amount of precipitation that infiltrates into the ground during the PMP’s occurrence, which is dependent on vegetation and soil type. Typically, runoff models incorporate a variable runoff coefficient or variable runoff rates. Commonly used models such as the U.S. Bureau of Reclamation (USBR) Rational Formula (USBR, 1987) incorporate a runoff coefficient (*C*); a *C* value of 1 represents 100 percent runoff and no infiltration. Other models such as the U.S. Army Corps of Engineers (COE) Flood Hydrograph Package HEC-1 (COE, 1998) separately compute infiltration losses within a certain time to arrive at a runoff amount. In computing the peak flow rate for the small drainage areas at the site, the DOE used the Rational Formula and assumed a *C* value of 1 (i.e., the DOE assumed that no infiltration would occur). Based on the conservatism of this parameter selection, the NRC staff concluded that the assumption was acceptable (NRC, 2008).

Another key parameter is the time of concentration (t_c), or the amount of time required for runoff to reach the outlet of a drainage basin from the most remote point in that basin. The peak runoff for a given drainage basin is inversely proportional to the time of concentration. For example, if

the time of concentration is assumed to be smaller, the peak discharge will be larger. Times of concentration and lag times are typically computed using empirical relationships. Velocity-based approaches are sometimes used when accurate estimates are needed. Velocity-based approaches rely on estimates of actual flow velocities to determine the time of concentration of a drainage basin. The DOE estimated times of concentration for the riprap design using an average of several methods, including the Kirpich Method (USBR, 1987). These methods are generally accepted in engineering practice and were considered by the NRC staff to be appropriate for estimating times of concentration at the Moab site. Based on a review of the calculations provided, the NRC staff concluded that the t_c values used by the DOE were acceptably derived (NRC, 2008).

NUREG-1623, "Design of Erosion Protection for Long-Term Stabilization," issued September 2002 (NRC, 2002), recommends use of the Rational Method (Chow, 1959) for estimating flood discharges. In using the Rational Method to calculate PMF discharge rates for the top and side slopes, the DOE assumed a *C* value of 1 and a flow concentration factor of 3. For a maximum top slope length of about 400 m (1,300 ft) (with a slope of 0.02) and a side slope length of about 55 m (180 ft) (with a slope of 0.2), the DOE estimated the peak flow rates to be about 0.119 m^3 per second (s) per meter of width (1.28 ft^3/s per foot of width) for the top slope and 0.124 m^3/s per meter of width (1.33 ft^3/s per foot of width) for the side slope. The DOE estimated the PMF flow rates for the downstream aprons; they were comparable to the flow rates for the side slopes. The PMF flow rates for the channels calculated by the DOE represented an accumulation of flows down the side slopes and offsite runoff. For the various channels and drainage structures, the DOE used the Soil Conservation Service unit hydrograph method (USBR, 1987) to calculate peak PMF flows. Based on a review of the calculations, including the time of concentration, rainfall intensity, and runoff, the NRC staff concluded that the DOE's estimated flow rates were acceptable (NRC, 2008).

The ability of a riprap layer to resist the velocities and shear forces associated with surface flows over the layer is related to the size and weight of the stones that make up the layer. Typically, riprap layers consist of a mass of well-graded rocks that vary in size. Because of the variation in rock sizes, design criteria are generally expressed in terms of the median stone size, D_{50}. A rock layer with a minimum D_{50} of 10 cm (4 in) could contain rocks ranging in size from 1.9 cm (0.75 in) to 15 cm (6 in); however, at least 50 percent of the weight of the layer will be provided by rocks that are 10 cm (4 in) or larger. Depending on the rock source, variations occur in the sizes of rock available for production and placement, and it is therefore necessary to ensure that these variations in rock sizes are not extreme.

Design criteria for developing acceptable gradations are provided by various sources (e.g., Simons and Li, 1982), and examples of acceptable gradations may also be found in NUREG-1623. Riprap layers of various sizes and thicknesses were proposed for use at this site, and the design of each layer is dependent on its location and purpose. To reduce the number of gradations that need to be produced, the DOE will place larger rock than is required in some areas. For ease of construction and to minimize the number of gradations, the DOE has purposely over-designed several areas by providing larger rock than needed in areas of the slopes and channels. The portion of the top slope that drains to the south will be protected by a 15 cm (6 in) thick layer of rock with a minimum D_{50} of about 4.6 cm (1.8 in). The area of the top slope draining to the north will be protected by a 15 cm (6 in) layer of rock with a minimum D_{50} of 3 cm (1.2 in). Based on a review of the proposed gradation specifications, the minimum D_{50} that will be provided is about 5 cm (2 in), which is conservative. For the north side slope of the disposal cell, the DOE proposed to use a 20 cm (8 in) layer of rock with a minimum D_{50} of about 10 cm (4 in). The south side slope will be covered with a 30 cm (12 in) layer of rock with a D_{50}

of about 15 cm (6 in). The east and west side slopes will be protected by 15 cm (6 in) layers of rock with a minimum D_{50} of 5 cm (2 in). The DOE used methods suggested in NUREG-1623 to determine the required rock sizes.

To protect the toe of the disposal cell and to dissipate the energy as the side slopes transition to natural ground, the DOE planned to construct aprons along the toe of the side slopes. The DOE planned to protect the area along the base of the south side slope by a rock toe/apron with a minimum D_{50} of 30 cm (12 in), while the toe of the north side slope would be protected by rock with a minimum D_{50} of 20 cm (8 in). The volume of rock was computed using a minimum depth of 3 times the D_{50} size and an apron width of 15 times the D_{50} size, or 3 m (10 ft), whichever is greater. The DOE used the design criteria suggested in NUREG-1623 to determine rock sizes and rock volumes for the toe aprons. Based on the staff's review of the DOE's analyses, the NRC staff concluded that the proposed rock sizes for the top slopes, side slopes, and aprons were adequate (NRC, 2008).

The DOE proposed to construct diversion channels at various locations. The DOE developed peak PMF flows, rock sizes, and scour depths in accordance with methods recommended in NUREG-1623. Based on a check of the computations, the NRC staff concluded that the peak flows, rock sizes, and scour depths were acceptable (NRC, 2008). The diversion channels would extend several hundred feet past the edge of the disposal cell to prevent flows from directly impacting the cell side slopes. The channels would convey flows to the east and west sides of the cell and then would turn southward. At the end of the channels, the channels will be widened (termed flow "spreaders" by the DOE). At the downstream end of the flow spreaders, additional rock would be provided to prevent gully headcutting into the spreaders. To reduce rock sizes to manageable levels, the DOE intended to construct a pre-formed slope of 1 vertical on 10 horizontal (10 percent), and this slope will be extended to the expected scour depth. The NRC staff's review of the design of the riprap for the channel outlets indicated that the rock was large enough and extended to a sufficient depth to resist gully intrusion (NRC, 2008).

The north side of the disposal cell would normally receive runoff directly from the area between Book Cliffs and the cell. This area would be protected by constructing a barrier using a very large quantity of excess excavated material or "wedge," which would act as a diversion berm to re-direct runoff away from the disposal cell. An access road between the cell and the wedge would be left in place. Runoff from the south side of the wedge would flow to the east and west in a ditch along the north side of the road, and runoff from the disposal cell would flow east and west along the south side of the road. Figure E-10 provides a cross-section through the wedge, ditches, access road, and north slope of the disposal cell.

The wedge would protect the disposal cell by reducing the amount of runoff that is carried in the diversion channels to the north of the cell. Also, the wedge would reduce the amount of sediment entering the diversion channels. The DOE evaluated sediment accumulation in the ditches located adjacent to the access road and provided analyses to show that the riprap sizes are large enough to resist the increased velocities associated with a reduction in channel capacity.

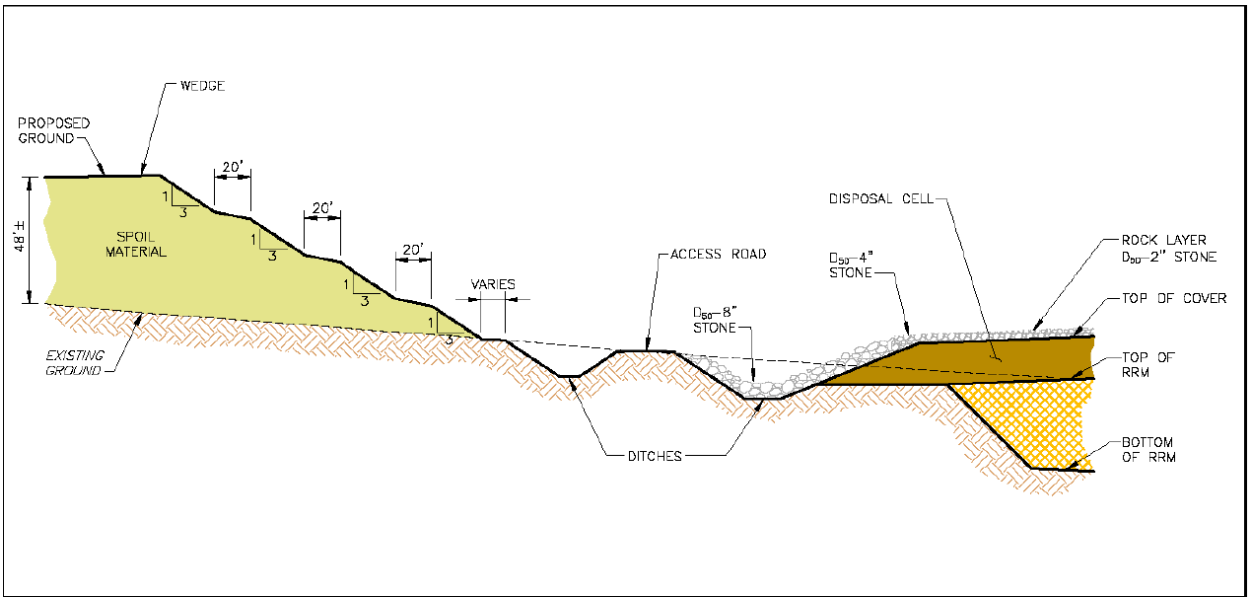


Figure E-10 Cross-Section Through Wedge, Ditches, Access Road, and North Slope of the Disposal Cell.
Image Credit: Figure 6-7 (DOE, 2008)

Rock durability, or resistance to weathering, is a key factor in evaluating the long-term stability of the rock cover. For rock to remain effective to control erosion over the time periods relied on for performance, the rock size selected should not be reduced by weathering processes. Therefore, if the rock size used for the cover does not diminish over the 1,000-year compliance period, its ability to control future erosion will be sustained. However, uncertainties exist with estimating future rock durability for 1,000 years. As a result, the NRC staff has identified in guidance three evaluations of rock durability to provide multiple and complementary lines of evidence and greater confidence in the sustained durability of the rock source selected. These evaluations are (1) rock durability testing and scoring, (2) absence of adverse minerals and heterogeneities, and (3) evidence of resistance to weathering. The paragraphs below describe the information provided by the DOE and the conclusions from the NRC staff's review.

A description of the rock types and the deposit that is proposed for the rock source is important to understand the variability of the deposit or formation containing the proposed rock source (e.g., percentage of each rock type) and the variability within the proposed rock source (e.g., different fabrics that could affect rock durability and resistance to weathering). Understanding the variability of the deposit/formation and rock types is important to obtaining representative samples for durability tests and developing rock production procedures that may be needed to mitigate the impacts from the use of adverse rock types in the deposit/formation.

The DOE selected basalt from a site approximately 6.44 km (4 miles) east of Fremont Junction, Utah, which is approximately 150 km (95 miles) west of the Crescent Junction site. The NRC approved the DOE's use of this rock source in 1988 for the erosion cover for the Green River UMTRA disposal cell in Green River, Utah (see Figure E-11 for a picture of the rock used at the Green River site). The Fremont Junction site consists of 160 hectares (400 acres) of property owned by the State of Utah School of Institutional Trust Lands Administration that has been permitted for the purpose of mining ordinary sand and gravel.



Figure E-11 Erosion Protection Cover at Green River Title I Site

The DOE based its selection of the Fremont Junction basalt on the combined results of the 1988 evaluations of the basalt for the Green River disposal cell and the recent studies in 2007 and 2008 for the Crescent Junction site. The 1988 evaluations consisted of field observations at two test pits, durability tests, petrographic analyses, and x-ray diffraction analyses. The 2007 and 2008 evaluations include field observations at eight test pits, durability tests, observations of the basalt on the Green River disposal cell, and natural analogue studies that provide evidence of long-term resistance to weathering. The basalt used at Green River was excavated from the same alluvial deposit about 1.6 km (1 mile) northeast of the areas that would be excavated for the Crescent Junction site. The 1988 petrographic analyses and x-ray diffraction analyses were relied on to support the DOE's evaluation of rock durability for the Crescent Junction site.

The Fremont Junction deposit includes an overburden layer that is approximately 2.4 m (8 ft) thick that consists of clayey sand and clayey silt with a small percentage of basalt clasts with caliche crusts, a reddish relic soil layer, and in places a white calcified zone. Beneath the overburden layer is the alluvial deposit that is at least 6 m (20 ft) thick and consists of 15–45 percent subrounded cobbles and boulders of basalt and other rock types such as tan sandstone, limestone, chert, and quartzite. Matrix material supports the cobbles and boulders and consists of sand and gravel up to 8 cm (3 in). The DOE's rock production procedures included screening to separate the matrix material from the cobbles and boulders of basalt and non-basalt that would then be crushed to the sizes specified for use as cover material. Based on the estimates of rock types and alluvial deposit thickness, the DOE estimated that the volume of useable rock should be at least twice the volume required by the design of the erosion cover.

The DOE estimated that the cobble and boulder portion of the alluvial deposit includes about 95 percent dark gray basalt and 2–3 percent red basalt. These two types of basalt were likely derived from two different sources that are 24 to 32 km (15 to 20 miles) southwest and

south-southwest of the site. The remaining non-basalt lithologies in the alluvial deposit make up about 2–3 percent. The estimates of the non-basalt lithologies and their respective rock durability scores are important to conclusions about how much of this material is acceptable and unacceptable for use. The DOE's rock production procedures indicated how the unacceptable material, such as the tan sandstone, would be removed from the deposit either by crushing, which would reduce its percentage further, or removal of the boulders before crushing.

The DOE evaluated rock durability using procedures outlined in NUREG-1623. The evaluation procedure provides a consistent and quantitative way to evaluate rock sources at NRC-regulated sites using American Society for Testing and Materials (ASTM) tests for parameters that are good indicators of rock durability (i.e., specific gravity, absorption, sodium sulfate soundness, and Los Angeles abrasion). Test data were provided for both gray and red basalt samples from the Fremont Junction area. These samples were collected in 1988 and 1989 for the Green River disposal cell and in 2007 and 2008 for the Crescent Junction site. The scores of the 1988 samples for the Green River disposal cell ranged from 66.7 to 79.4 percent. The sample that scored 66.7 percent was described in the 1988 report as severely weathered. The DOE indicated that these samples were likely the vesicular red basalt. The DOE's 1988-1989 quality control testing and scoring of four samples collected during the actual placement of the basalt cover at the Green River disposal cell resulted in additional scores for the Fremont Junction basalt that were higher than the initial test results. Scores for the Type A rip rap ranged from 78 to 90 percent with an average of 85 percent. Scores for the Type B rip rap ranged from 80 to 90 percent with an average of 83 percent.

The DOE's scores from the 2007 and 2008 samples also provided additional results. The gray basalt, which makes up approximately 95 percent of the alluvial deposit, had scores of 82.9 and 83.3 percent. These scores exceed the 80 percent score that indicates a high-quality rock that can be used for most applications according to the NRC's guidance. The red basalt that makes up approximately 2–3 percent of the basalt had a score of 63.7 percent, which was comparable to the 1988 initial score. Although the 63.7 percent score is lower than that for the gray basalt, it is in the range for rock that would be acceptable for use in non-critical areas. The DOE noted that the sample of the red basalt was softer than the gray basalt, possibly because it was vesicular. During the June 25, 2008, site visit, field observations of cobbles and boulders of dense non-vesicular red basalt appeared to both the DOE and NRC staffs to be more competent than the vesicular red basalt that had a low score (NRC, 2008).

The DOE used information from field observations and the 1988 petrographic and x-ray diffraction analyses to identify whether adverse heterogeneities or adverse minerals (e.g., olivine and clay) were present that could be vulnerable to weathering. Field observations were used to identify large-scale adverse heterogeneities such as the undesirable overburden layer and fine-grained matrix sediments supporting the cobbles and boulders in the deposit. The DOE proposed to remove the overburden layer before excavation of the basalt alluvial deposit. The DOE also proposed to screen out the finer matrix material from the basalt cobbles and bounders. The DOE's 1988 petrographic analyses concluded that the samples lacked significant amounts of adverse minerals such as calcite, clays, olivine, and feldsphathoids. X-ray diffraction analyses determined that the basalt samples contained only 1 percent olivine.

The DOE attempted to find evidence of resistance of the rock to weathering. This evidence can be both direct and indirect. The DOE's 2007 and 2008 field observations of the eight test pits did not indicate evidence of significant weathering of the basalt (e.g., weathering rinds). To confirm these observations and to resolve reports of an upper weathered zone and weathering rinds on the red basalt, the DOE also observed the crushed basalt used on the cover of the

Green River disposal cell and the subrounded basalt boulders in the Green River channel. This basalt from the same deposit at Fremont Junction provided a large "exposure" of the basalt that was clean and free of the fine material and dust that limited observations in the test pits at Freemont Junction. The DOE did not observe weathering rinds on either the dark gray or red basalt. The DOE concluded that the descriptions of weathering rinds from the 1988 investigation possibly were interpreted to be the thick caliche crusts on some basalt clasts. The only evidence of basalt weathering was the leaching of olivine crystals by chemical weathering on the surface of a sample observed in the 1988 petrographic analyses. This analysis noted that the olivine crystals observed in the interior of the sample had not been weathered. As mentioned above, the x-ray diffraction analysis indicated that olivine only made up 1 percent of the sample analyzed.

Because of the absence of quantitative weathering rate studies for basalt as well as other rock types, the NRC's guidance in NUREG-1757, "Consolidated Decommissioning Guidance," Volume 2, Revision 2 (NRC, 2022c) notes that indirect evidence of resistance to weathering can add confidence in the durability and slow weathering of rock types selected for long-term erosion protection. The DOE identified the following geologic analogues to support a determination that the Fremont Junction basalt has remained resistant to weathering for thousands of years, well beyond the 1,000-year compliance period:

- Basalt boulders may have resisted weathering for 500,000 years based on the estimated age of the alluvial deposit, using a published stream downcutting rate for this part of the Colorado Plateau.

- Basalt boulders have resisted weathering for possibly 8,000 to 10,000 years based on the estimated age of wind-fluted surfaces on exposed basalt boulders caused by wind-driven sand.

- Rock varnish on exposed basalt boulders may have been formed several thousand years ago.

- Lichen cover on exposed basalt boulders may have been in place for hundreds of years.

- Basalt boulders buried at depths of 1 to 2 m (3 to 6 ft) in the overburden commonly have white calcium carbonate crusts, which can take tens of thousands of years to form, indicating these boulders have been in place for many thousands of years without noticeable weathering effects.

The NRC staff concluded the following:

- Durability test results and scores demonstrate acceptable physical properties of the Fremont Junction basalt.

- Adverse minerals such as olivine and clay are present in very small amounts (1 percent), and adverse heterogeneities, such as friable sandstone and matrix sand and gravel, can be identified and avoided when rock is excavated or screened and crushed in processing.

- There is direct evidence from the Fremont Junction deposit and Green River disposal cell cover of the absence of weathering such as the lack of weathering rinds.

- Indirect evidence from natural basalt analogues adds confidence that basalt weathering rates are slow and the basalt has resisted weathering for thousands of years at the Fremont Junction area (NRC, 2008).

The NRC staff concluded that the Fremont Junction basalt is durable and should resist weathering and associated size reduction for at least the 1,000-year compliance period. Therefore, the NRC staff considered the Fremont Junction basalt acceptable for use in the erosion controls at the Crescent Junction site.

The DOE provided information regarding testing, inspection, and quality control procedures to be used for the erosion protection materials. Because there was indication that the rock in the proposed quarry could be variable in certain locations, the DOE provided information to document the quality assurance and quality control procedures that would be implemented during rock production to address variability and to ensure that rock of acceptable quality would be consistently produced.

DOE plans included stripping overburden from the alluvial deposits and separation of this material from the basalt cobbles and boulders. Rock materials would be excavated, crushed, and screened into stockpiles of various sizes. The rock would then be crushed and further screened, as necessary, to produce the required rock sizes. These primary and secondary sorting processes should ensure that rock would be relatively homogeneous and that visible portions of the stockpiles would be representative of the entire stockpile. Crushing and screening would also remove significant amounts of weak, friable materials (i.e., tan sandstone and limestone), resulting in a product that contains only limited amounts of poor-quality materials. In the unlikely event that a stockpile contains significant unacceptable rock, the lower quality material (i.e., tan friable sandstone and limestone) would be extracted to ensure that the final product contains no more than 10 percent by volume. Based on observations during a June 2008 site visit and information provided by the DOE, the NRC staff concluded that the proposed program for rock production was acceptable (NRC, 2008).

The DOE proposed that rock durability and gradation testing would be performed a minimum of four times or at a frequency of one test for every 7,600 m^3 (10,000 yd^3) (or both) of material produced, as recommended in NUREG-1623. Rock durability testing would include the following:

- bulk specific gravity—ASTM C127, “Standard Test Method for Relative Density (Specific Gravity) and Absorption of Coarse Aggregate”
- absorption—ASTM C127
- sodium sulfate soundness—ASTM C88, “Standard Test Method for Soundness of Aggregates by Use of Sodium Sulfate or Magnesium Sulfate”
- Los Angeles abrasion at 100 cycles—ASTM C131, “Standard Test Method for Resistance to Degradation of Small-Size Coarse Aggregate by Abrasion and Impact in the Los Angeles Machine,” or ASTM C535, “Standard Test Method for Resistance to Degradation of Large-Size Coarse Aggregate by Abrasion and Impact in the Los Angeles Machine”
- Schmidt rebound hardness—International Society for Rock Mechanics method

Based on its review of the information submitted by the DOE and independent NRC staff calculations, the NRC staff concluded that it had reasonable assurance that the erosion protection requirements in 40 CFR Part 192 would be met (NRC, 2008).

How Does the Moab Design-Based Example Relate to 10 CFR Part 61 Guidance?

The Moab example illustrates the design-based approach described in Section 5.2.2 of this document and reproduced below for ease of reference. This approach may be used to assess performance of a 10 CFR Part 61 facility:

(1) Define the design objectives.
(2) Develop or select the design.
(3) Document and provide the basis for assumptions.
(4) Characterize or parameterize the design.
(5) Assess the expected performance of the design.
(6) Provide support (technical basis) for the design.

The DOE clearly defined design objectives for the erosion protection system at the Crescent Junction site. The DOE provided sufficient detail on the selected design, as well as sufficient information to support its calculations and assessments. Supporting information included information from the literature, field observations, testing, and analog studies. In demonstrating compliance with the applicable regulatory criteria, the DOE followed guidance provided in NUREG-1623, which gives acceptable methods for designing erosion protection systems to help ensure site stability at NRC regulated facilities, greatly facilitating the NRC staff's review.

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APPENDIX F
10 CFR PART 61 DRAFT ENVIRONMENTAL IMPACT STATEMENT DEFAULT SCENARIOS

This section summarizes the scenarios and calculations used to determine the waste classification concentrations specified in Title 10 of the *Code of Federal Regulations* (10 CFR) 61.55, "Waste classification." The summary is intended to aid licensees and Agreement States in understanding how the waste classification concentration limits were developed. The U.S. Nuclear Regulatory Commission (NRC) staff developed the concentration limits for the initial promulgation of 10 CFR Part 61, "Licensing Requirements for Land Disposal of Radioactive Waste," as described in more detail in NUREG-0782, "Draft Environmental Impact Statement on 10 CFR Part 61 'Licensing Requirements for Land Disposal of Radioactive Waste,'" issued September 1981 (See Vol. 2, Chapter 7 and Vol. 4, Appendix G; NRC, 1981a), and NUREG-0945, "Final Environmental Impact Statement on 10 CFR Part 61 'Licensing Requirements for Land Disposal of Radioactive Waste,'" issued November 1982 (NRC, 1982b). The calculations discussed in NUREG-0782 and summarized in this appendix can be used to determine the waste classification limits as well as to assess both the potential impacts from the management of low-level radioactive waste (LLW) and potential controls to ensure protection of public health and safety and the environment. To develop the waste classification limits in 10 CFR 61.55, the NRC staff relied upon a subset of the calculations and assumptions that are summarized in this appendix. Although this appendix summarizes calculations beyond those used to develop the waste classification limits, the NRC staff has indicated herein those approaches and parameter values that were used specifically to develop the waste classification limits. The NRC staff also summarizes the scenarios evaluated, assumptions employed, calculations performed, and adjustments made to develop the concentration limits. Both accident scenarios and intruder scenarios were considered.

The NRC staff recently developed a user-friendly tool, known as TableCalculator (Ridge et al., 2019), with a graphical user interface to help stakeholders better understand the calculations performed to provide the basis for the waste classification limits (concentrations) found in 10 CFR 61.55. This tool has been made available through the NRC's Radiation Protection Computer Code Analysis and Maintenance Program (RAMP) (https://ramp.nrc-gateway.gov/codes/tablecalculator). The tool replicates the original calculations that were documented in a series of large NUREG reports and gives direct references to all the assumptions, and correction factors used in creating the tables. Users are able to change inputs and trace calculations.

After the initial promulgation of 10 CFR Part 61, the NRC updated the impact analysis methodology in NUREG/CR-4370, "Update of Part 61 Impacts Analysis Methodology," issued January 1986 (NRC, 1986a). The update included (1) an update of the low-level radioactive waste source term, (2) consideration of additional alternative disposal technologies, (3) expansion of the methodology used to calculate disposal costs, (4) consideration of an additional exposure pathway involving direct human contact with disposed waste due to a hypothetical drilling scenario, and (5) use of updated health physics analysis procedures (i.e., International Commission on Radiological Protection (ICRP) Publication 30, "Limits for Intakes of Radionuclides by Workers" (ICRP; 1979; ICRP, 1980; ICRP, 1982)). The purpose of the update was, in part, to potentially be used as a methodology to generically analyze disposal

of individual wastes that exceed Class C concentrations on a case-by-case basis. The update to the methodology did not alter the waste classification limits in 10 CFR 61.55 and is not summarized here.

The NRC staff evaluated both acute and chronic exposures for a reference disposal facility to develop the 10 CFR 61.55 concentration limits. The NRC staff evaluated acute exposures using the intruder-construction scenario and its variant, the intruder-discovery scenario.[1] The NRC staff evaluated chronic exposures using the intruder-agriculture scenario. Figures F-1[2] and F-2 depict the exposure pathways associated with the intruder-construction and intruder-agriculture scenarios, respectively. Both the acute and chronic intruder scenarios employed several common assumptions, including the following:

- temporary breakdown of institutional controls, allowing inadvertent intrusion to occur
- inadvertent intruder directly contacts the waste
- intrusion into last disposal cell constructed (i.e., radionuclide decay during operations neglected)
- waste remaining on the disposal site

To estimate exposures in each receptor scenario, the NRC staff evaluated expressions of dose equivalent, *H*, using the following equation:

$$H = \sum_{i,j} PDCF \times C_a \qquad \text{(F.1)}$$

Where:

PDCF pathway dose conversion factor [Sv per Bq/m^3] for radionuclide *i* for exposure pathway *j*; and

C_a concentration [Bq/m^3] of radionuclide *i* in environmental media (i.e., either onsite soil or air) at the inadvertent intruder access point for exposure pathway *j*.

The NRC staff reports the pathway dose conversion factors used in the calculations in Tables G.4 and G.7 of NUREG-0782 for the intruder-construction and intruder-discovery receptor scenarios. Pathway dose conversion factors used in the calculations for the intruder-agriculture receptor scenario are reported in Tables G.5 through G.7 of NUREG-0782. The pathway dose conversion factors were calculated in NUREG/CR-1759, "Data Base for Radioactive Waste Management," Volume 3, Appendix B (NRC, 1981b). The information used to calculate the pathway dose conversion factors includes human physiological parameters (e.g., breathing rates, nuclide metabolism), dietary intakes, and nuclide-specific food chain transfer rates. The dose for an uptake pathway (e.g., breathing contaminated air) can be determined with transfer factors (e.g., soil-to-air) and fundamental dose conversion factors. However, the NRC staff used pathway dose conversion factors that account for assumptions regarding environmental characteristics and human actions upon which the transfer factors

[1] For stable waste, the NRC staff assumed the intruder would recognize the waste and would limit exposures once the hazard was recognized. Thus, the NRC staff limited the number of hours of exposure for waste that is stable and segregated from unstable waste. This variant of the intruder-construction receptor scenario is known as the intruder-discovery receptor scenario.

[2] Figures and tables can be found after the text so the reader can better follow the calculations.

would depend to determine, on a generic basis for a reference disposal site, the total organ doses received from concentrations of radionuclides in environmental media (e.g., air, water, soil).

Inhalation dose conversion factors were derived from Bates, et al. (1966) and ICRP Publication 19, "The Metabolism of Compounds of Plutonium and Other Actinides," issued 1972 (ICRP, 1972). Ingestion dose conversion factors were derived from 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," Revision 1, issued October 1977 (NRC, 1977a), and NUREG-0172, "Age-Specific Radiation Dose Commitment Factors for a One-Year Chronic Intake," issued November 1977 (NRC, 1977b). Direct gamma dose conversion factors for exposure to volumetric sources were calculated from equations in Health and Safety Laboratory Report No. HASL-195, "The Radiation Field in Air Due to Distributed Gamma Ray Sources in the Ground," issued 1968 (AEC, 1968), with emitted gamma energy characteristics in *Table of Isotopes,* Sixth Edition, issued 1967 (Lederer et al., 1967). Direct gamma dose conversion factors for submersion in air were developed from NUREG-0456, "A Classification System for Radioactive Waste Disposal—What Waste Goes Where?" issued June 1978 (NRC, 1978).

The NRC staff estimated the concentration of radionuclides at the inadvertent intruder access point using a series of four factors to account for control mechanisms. The control mechanisms were required by 10 CFR Part 61 and would result in differences between the concentration in the waste and the concentration to which the inadvertent intruder is exposed using the following equation:

$$C_a = C_w \times f_o \times f_d \times f_w \times f_s \qquad \text{(F.2)}$$

Where:

C_w concentration (Bq/m^3) of radionuclide *i* in the waste

f_o time-delay factor (unitless (-)), which accounts for all the control mechanisms that increase the time period between the cessation of operation of the reference disposal facility and the end of the active institutional control period for radionuclide *i*

f_d site design and operation factor (-), which accounts for effects of any engineered barriers designed into the waste disposal facility in addition to any site operational practices that may reduce transport

f_w wasteform and package factor (-), which accounts for the physical and chemical characteristics of the waste that may inhibit release and transport of contaminants

f_s site selection factor (-), which accounts for the dependence of the radionuclide concentration at the access point on either or both the environmental characteristics of the disposal site or exposure durations associated with the various receptor scenarios

The following paragraphs give an additional description of the control factors and how the NRC staff estimated their values.

The NRC staff performed a screening analysis to determine which waste streams required special consideration of institutional controls, wasteform, and natural or engineered barriers. The process relied upon five distinct classification tests: regular standard, regular modified, layered standard, layered modified, and hot waste facility. Figure F-3 summarizes the classification test process. For the regular-standard classification test, no additional control factors were applied to intruder-construction or intruder-agriculture scenarios. For the regular-modified classification test, the NRC staff assumed that the waste is stable and segregated from unstable waste; thus, only the intruder-discovery receptor scenario was applicable.

For the layered-standard classification test, the NRC staff assumed that waste was likely to be disposed beneath a minimum of 2 meters (7 feet) of cover and 4 to 5 meters (13 to 16 feet) of other regular wastes; therefore, the intruder-agriculture receptor scenario was not applicable. Additional control factors applied to the intruder-construction scenario for air uptake and direct gamma exposures. For the layered-modified classification test, the NRC staff assumed waste is stable and segregated from unstable waste; thus, only the intruder-discovery scenario was applicable. The NRC staff also assumed no credit for layering of waste after 500 years in either the layered-standard or layered-modified classification tests.

For the hot waste facility classification test, the NRC staff assumed further controls for air uptake and direct gamma exposures in the intruder-construction scenario. The NRC staff assumed the intruder-agriculture scenario did not apply because of the design of the facility. The NRC staff also assumed no credit for controls associated with the hot waste facility after 1,000 years. Because the hot waste facility did not affect the concentration limits in 10 CFR 61.55, the control factors are not discussed further in this appendix.

In Chapter 7 of NUREG-0782, the NRC staff documented the calculation to derive concentration limits using a 500-millirem annual inadvertent intruder exposure. Table 7.1 of NUREG-0782 provides the concentration values calculated from the intruder-construction, intruder-discovery, and intruder-agriculture receptor scenarios. The NRC staff calculated the values in Table 7.1 assuming the most conservative assumption regarding the wasteform (i.e., the wasteform is dispersible as ordinary soil, and no credit was taken for improved wasteforms). The values in Table 7.1 were derived using the regular-standard and regular-modified classification tests described above at 100 years (for Class A and B limits) and the regular-standard classification test at 500 years (for Class C limits). The intruder-discovery receptor scenario at the end of the active institutional control period (i.e., 100 years) was originally considered as the basis for the Class B limit because the NRC staff assumed that an intruder would recognize stabilized waste and exposures would be limited once the hazard was recognized. However, for several radionuclides, the intruder-discovery scenario at the end of the active institutional control period was less limiting than the proposed Class C limits, which were based on intruder-construction or intruder-agriculture scenarios at 500 years. Thus, the NRC decided not to establish a Class B limit for long-lived radionuclides. For short-lived radionuclides, the NRC staff selected the more limiting of either the Class C limit or the concentration established by the intruder-discovery scenario as the Class B limit.

After the scenario selection adjustment for Class B limits, two additional adjustments were made to the values listed in Table 7.1 of NUREG-0782. The first adjustment involved cesium (Cs)-137. Section 7.2.5 of NUREG-0782 explains that waste contaminated with Cs-137 was likely to be disposed with waste with very low levels of Cs-137. Therefore, the NRC staff increased the limits for Cs-137 by a factor of 20 for Class A waste and a factor of 10 for Class B

and C waste. Table 7.2 of NUREG-0782 lists these adjusted values, and these Class A and B values were incorporated into Table 1 of 10 CFR 61.55. The second adjustment was the result of public comments on the initial proposed rule. The NRC staff received comments that further adjustments should be made to the Class C limits to account for the difficulty of an inadvertent intruder contacting Class C waste. The NRC staff agreed and increased all Class C values by a factor of 10, except for Cs-137, which had already been adjusted in NUREG-0782.

Time-Delay Factor

The NRC staff represented radioactive decay expected to occur over the time between the cessation of operations at the reference disposal facility and the end of the active institutional control period. The NRC staff calculated a time delay factor, f_o, as follows:

$$f_o = e^{-\lambda t} \qquad \text{(F.3)}$$

Where:

λ radionuclide decay constant (year^{-1}) of radionuclide *i* in the waste

t time (year) from cessation of operations at the reference disposal facility to the time that the specific scenario is initiated

The time-delay factor is a property of the scenario and the disposal technology being considered. For the intruder scenarios, the time-delay factor accounts for the time to the end of the active institutional control period (i.e., 100 years) for Class A and B limits, or as long as the intruder barrier is assumed effective (i.e., 500 years) for the Class C limit. The assumed time is equivalent to the assumption that the intrusion scenario involves the last disposal cell constructed at the site and neglects the possibility that the intrusion scenario may involve one of the earlier disposal cells.

Site Design and Operation Factor

The NRC staff estimated a site design and operation factor, f_d, to represent the impact of waste emplacement methods on the accessibility of waste to the inadvertent intruder. The site design and operation factor accounted for the hazard of the waste, emplacement techniques, and exposure pathway. The site design and operation factor is dependent upon the efficiency of the disposal design and the intruder scenario. To develop values for the site design and operation factor, the NRC staff evaluated regular waste, which is emplaced without consideration of protecting an inadvertent intruder, and layered waste, which is emplaced at the bottom of the disposal cell, so that at least 5 m (16 feet) of earth or other (lower activity) waste streams cover the layered waste. Within each waste type (i.e., regular and layered), the NRC staff assigned values denoting the dilution of the waste due to disposal practices for waste emplacement. The assumed values of the site design and operation factor for regular and layered wastes waste are listed in Table F-1.

Wasteform and Package Factor

The NRC staff estimated a wasteform and package factor to represent the resistance of the waste to mobilization from actions of the inadvertent intruder. For example, this factor would be considerably less than unity for waste streams solidified in a matrix or packaged in containers

that are likely to retain their integrity at the time of the inadvertent intrusion. The wasteform and package factor is dependent upon the property of the waste stream as it is being disposed. Therefore, the NRC staff assigned differing values for the various exposure pathways to account for the impact of the waste stream properties on mobilization in the exposure pathways.

To develop the values for the wasteform and package factor, the NRC staff considered the accessibility of the contamination to mobilization, which is dependent on the wasteform properties (e.g., surface-contaminated objects versus activated metals) together with either the dispersibility of the contamination for airborne pathways or whether the waste was solidified (e.g., in cement) for direct exposure pathways. Thus, the NRC staff estimated the wasteform and package factor, f_w, for airborne and direct exposure pathways according to the following equations:

$$f_w = f_{accessibility} \times f_{dispersibility} \quad \text{(Airborne Pathways)} \quad \text{(F.4a)}$$

$$f_w = f_{accessibility} \times f_{solidification} \quad \text{(Direct Pathways)} \quad \text{(F.4b)}$$

Where:

$f_{accessibility}$ accessibility multiplier (unitless (-)) of each wasteform that accounts for the location of contamination in or on the waste

$f_{dispersibility}$ dispersibility multiplier (-) for each wasteform that accounts for resistance to mobilization through air pathways

$f_{solidication}$ solidification multiplier (-) of each wasteform that accounts for self-shielding of direct radiation as a result of using a solidification agent

Table F-2 lists the values of the accessibility, dispersibility, and solidification multipliers for the airborne and direct exposure pathways used to estimate the wasteform and package factor. For the purposes of developing waste classification limits, the NRC staff assumed no wasteform credit for the dispersibility multiplier and no solidification credit for the solidification multiplier to derive the classification limits. Therefore, the remaining values listed in Table F-2 were not used to develop the waste classification limits.

For food pathways evaluated in the intruder-agriculture receptor scenario, the NRC staff accounted for wasteforms that would reduce leaching of radionuclides from disposed waste and subsequent uptake by plant roots. The NRC staff estimated the wasteform and package factor, f_w, using the following equation for the food pathways:

$$f_w = M_o \times t_c \times f_{accessibility} \times f_{leaching} \quad \text{(Food Pathways)} \quad \text{(F.4c)}$$

Where:

M_o partition ratio (unitless (-)) of individual radionuclides between leachate and unsolidified wasteforms

t_c fraction of each year (-) that the waste is in contact with irrigation or rainwater

$f_{accessibility}$ accessibility multiplier (-) of each wasteform that accounts for the location of contamination in or on the waste

$f_{leaching}$ leaching multiplier (-) of each wasteform that accounts for reduction of leaching due to solidification and the presence or absence of chelating chemicals

The values of the radionuclide-specific partition ratios are listed in Table G.14 of NUREG-0782, Vol. 4, Appendix G. The irrigation contact time was assumed to be unity. The values for the accessibility multiplier values are identical to the values for airborne exposure pathways listed in Table F-2. The values for the leaching multiplier are a function of (1) a wasteform's resistance to leaching as a result of the solidification technique, (2) the presence of chelating agents or organic chemicals that may increase the mobility of radionuclides during or after leaching, or both, and (3) whether the waste containing chelating agents is mixed with other wastes or segregated when disposed. The values for the leaching multiplier for the food pathways used to estimate the wasteform and package factor are listed in Table F-3. For the purposes of developing waste classification limits, the NRC staff assumed that waste is unsolidified. Therefore, the remaining values listed in Table F-3 were not used to develop the waste classification limits.

Site Selection Factor

The NRC staff estimated a site selection factor to represent the environmental characteristics of the site and the behavior of the inadvertent intruder. For direct exposure pathways, the site selection factor is dependent upon the exposure duration and the areal extent of the direct gamma source (and its attenuation) that the inadvertent intruder is exposed to for the inadvertent intruder scenario. For airborne exposure pathways, the site selection factor is dependent upon the transfer of contaminants in the soil to the air, which is a function of environmental characteristics of the region in which the disposal facility is located, and the exposure duration for the inadvertent intruder scenario. For food exposure pathways, the site selection factor is dependent upon the amount of food consumed by the inadvertent intruder that is grown onsite. The NRC staff assumed that half the food consumed by the inadvertent intruder in the intruder-agriculture scenario was grown onsite.

The site selection factor, f_s, for direct pathways was estimated by the following equation for the intruder-construction and intruder-agriculture scenarios:

$$f_s = \sum_k f_{attenuation} \times t_k \qquad \text{(F.5)}$$

Where:

$f_{attenuation}$ attenuation factor (unitless (-)) for activity *k*

t_k fraction (-) of the year that the inadvertent intruder engages in activity *k*

The NRC staff estimated the value of the site selection factors for the direct pathways using the values listed for the parameters in Table F-4. The following equations estimate the site selection factor, f_s, for airborne pathways for the intruder-construction scenario and the intruder-agriculture scenario, respectively:

$$f_s = T \times t_k \qquad \text{(Intruder-Construction)} \quad \text{(F.6a)}$$

$$f_s = T \times \sum_k \left[\frac{m_k}{m_g} \times t_k\right] \qquad \text{(Intruder-Agriculture)} \quad \text{(F.6b)}$$

Where:

T transfer factor (cubic meters (m^3)(air)/m^3(soil)) of contaminants from soil to air

m_k mass loading (micrograms (µg)/m^3) of particulates in air during activity k

m_g mass loading (µg/m^3) of particulates in air while gardening

t_k fraction (unitless (-)) of the year that the inadvertent intruder engages in activity k

The NRC staff estimated the value of the site selection factors for the airborne pathways using the values listed for the parameters in Table F-4.

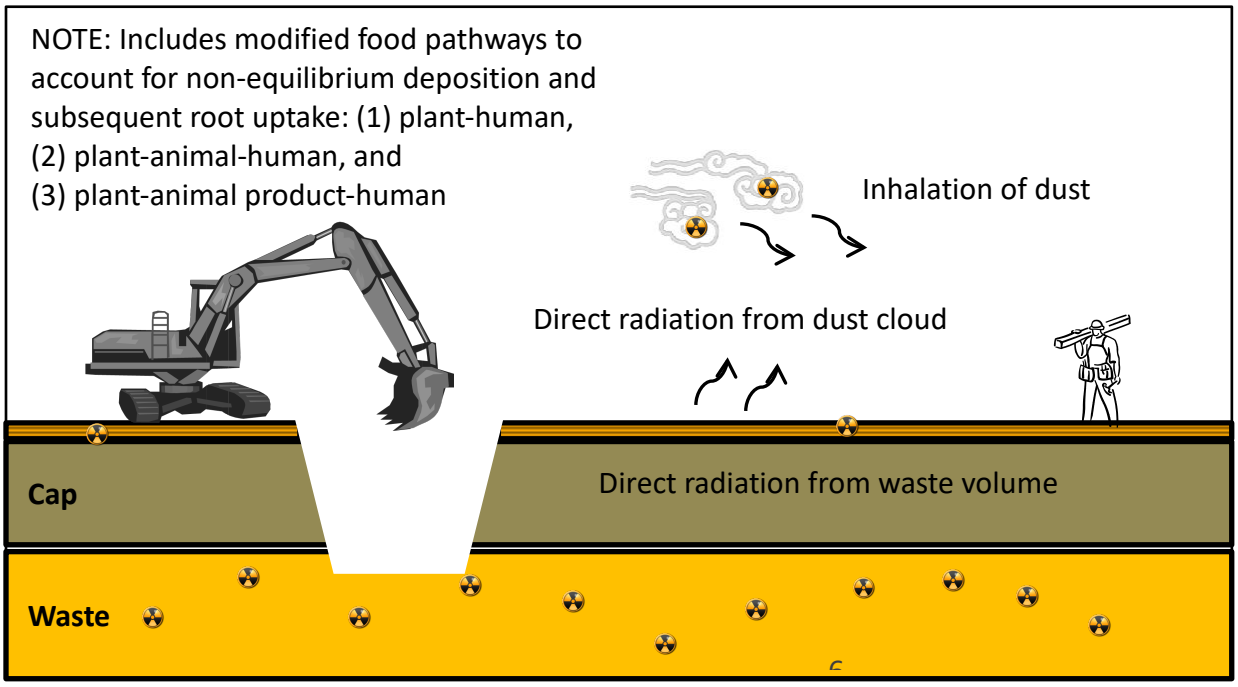


Figure F-1 Intruder-construction receptor scenario exposure pathways

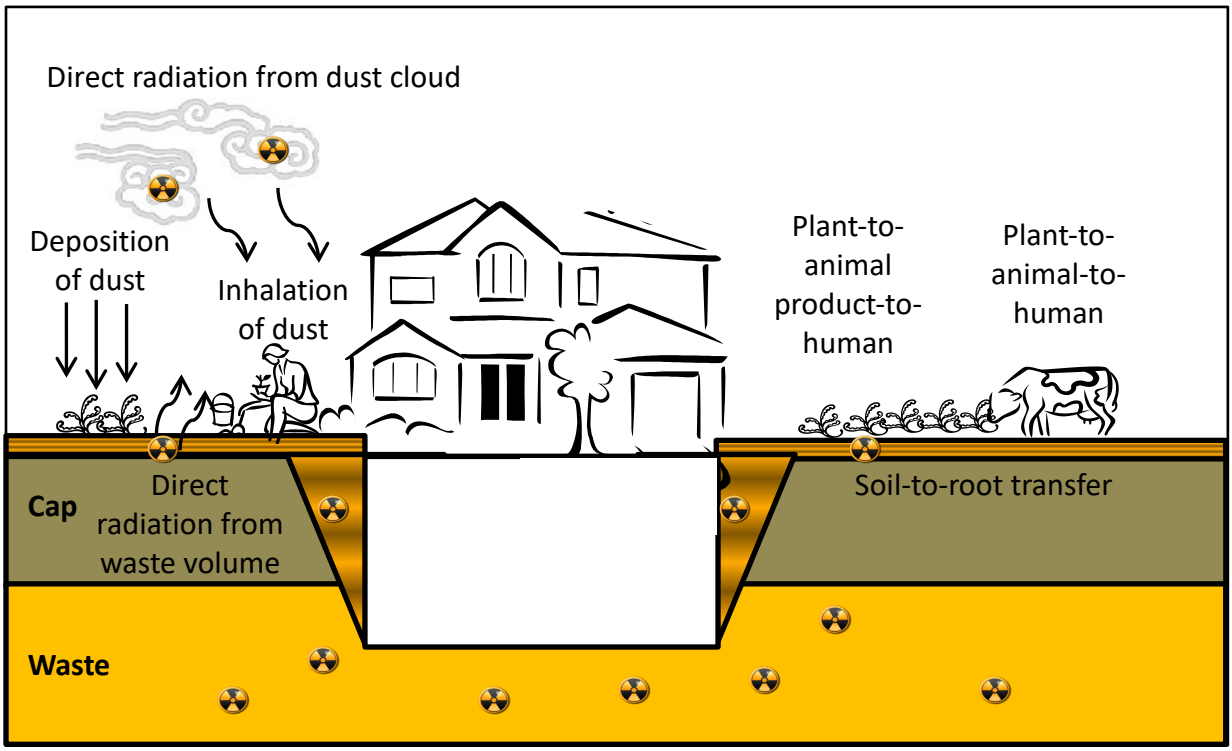


Figure F-2 Intruder-agriculture receptor scenario exposure pathways

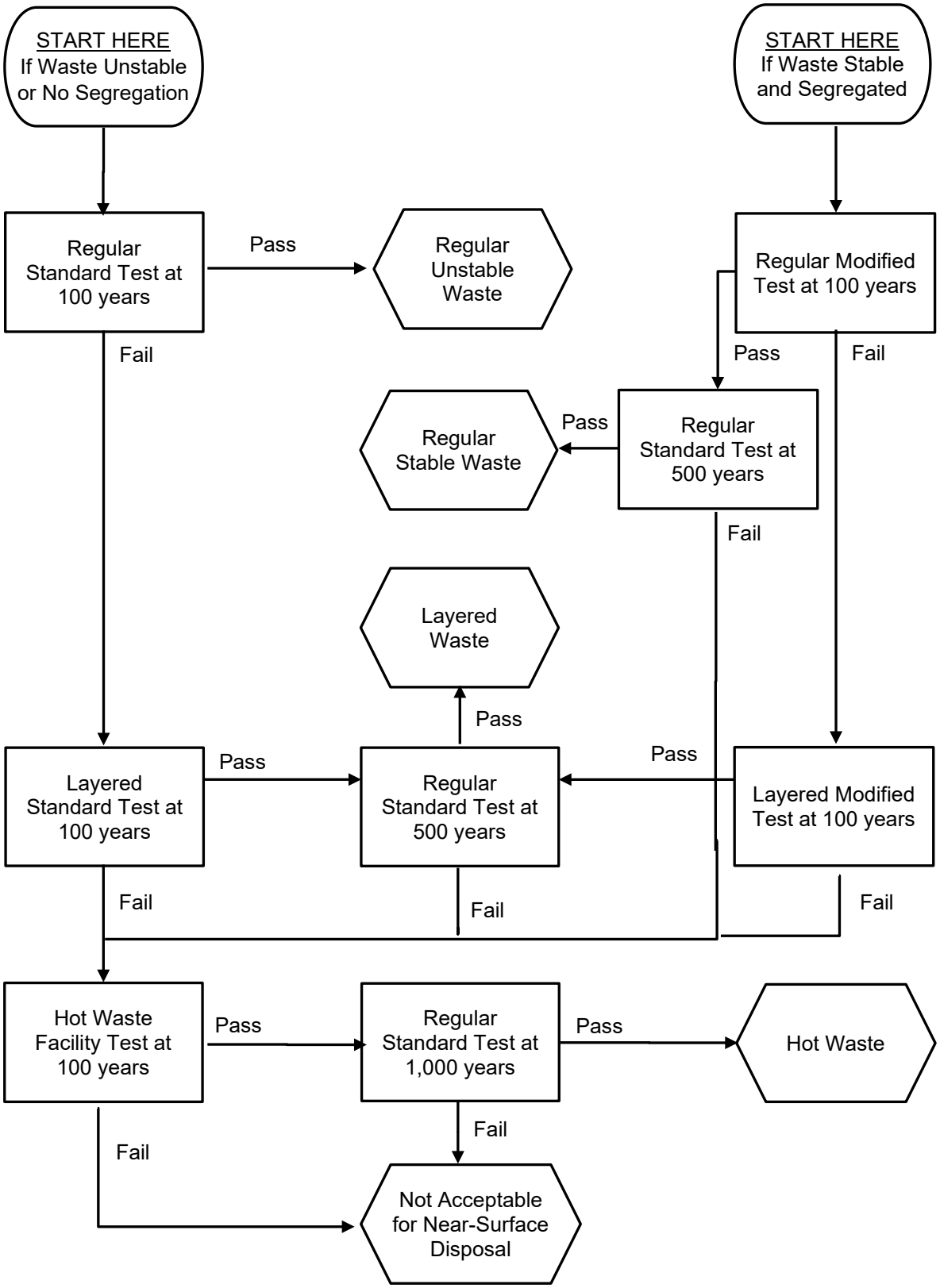


Figure F-3 Waste Classification Test Procedure (Adapted from NUREG-0782, Vol. 4, Appendix H)

Table F-1 Site Design and Operation Factors for Regular and Layered Waste

Emplacement Method†	Intruder-Construction Receptor Scenario			Intruder-Agriculture Receptor Scenario		
	Regular	Layered‡		Regular	Layered‡	
		Airborne Pathways	Direct Pathways§		Airborne Pathways	Direct Pathways§
Random Containers	0.5	0.1 × Regular Waste Value	(1/1200) × Regular Waste Value	0.5 × f_{mixing}§§	0.1 × Regular Waste Value	(1/1200) × Regular Waste Value
Stacked Containers	0.75			0.75 × f_{mixing}§§		
Decontainerized	0.5			0.5 × f_{mixing}§§		

Notes:

† To derive the classification limits, the NRC staff only used a value for this factor that is consistent with random containers or decontainerized waste.

‡ The NRC staff used regular-standard and regular-modified classification tests to derive the classification limits. Therefore, the values for the layered classification tests reported in this table were not used to derive the classification limits.

§ Attenuation of radiation through a layer equivalent to 1 meter (3 feet) of soil.

§§ $f_{mixing} = \frac{Volume\ of\ Waste\ Excavated}{Total\ Volume\ of\ Excavation} = \frac{232\ m^3}{(232\ m^3 + 680\ m^3)} = 0.25$

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Table F-2 Accessibility, Dispersibility, and Solidification Multipliers for Airborne and Direct Exposure Pathways

	Airborne Exposure Pathways	Direct Exposure Pathways
Accessibility Multiplier		
Surface-contaminated waste and waste containing radionuclides in soluble form	1	1
Waste with significant metallic content, with both activated and surface crud contamination	0.1	0.1
Activated metals	0.01	0.1
Dispersibility Multiplier		
No wasteform credit (i.e., soil)	1	N/A
Waste that crumbles or fractures extensively and decomposes quickly	0.1	N/A
Mix of above and below wastes	0.01	N/A
Waste likely to resist biological and chemical attack with significant compressive strengths (e.g., good synthetic polymer)	0.001	N/A
Solidification Multiplier		
Wastes solidified using significant cement for solidification	N/A	0.8
Wastes without significant solidification	N/A	1

Note:
The NRC staff assumed no wasteform credit for the dispersibility multiplier and no solidification credit for the solidification multiplier to derive the classification limits.

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Table F-3 Leaching Multiplier Values for Food Exposure Pathways

	Waste contains or is disposed mixed with chelating agents or organic chemicals that may increase the mobility of radionuclides during and/or after leaching	Waste does not contain and is not disposed mixed with chelating agents or organic chemicals that may increase the mobility of radionuclides during and/or after leaching
Unsolidified waste	1	1
Waste solidified assuming: • half of waste is solidified using cement • half is solidified using urea-formaldehyde	1	1/4
Waste solidified assuming: • half of waste is solidified using cement • half is solidified using synthetic organic polymers	1/4	1/16
Waste solidified assuming all waste is solidified using synthetic organic polymers	1/16	1/64
Note: The NRC staff assumed waste is unsolidified to derive the classification limits.		

Table F-4 Soil-to-Air Transfer Factors, Annual Activity Fractions, and Attenuation Factors for the Direct and Airborne Exposure Pathways

	Intruder-Construction Scenario	Intruder-Agriculture Scenario		
Soil-to-Air Transfer Factor	3.53×10^{-10} m^3/m^3†	2.53×10^{-10} m^3/m^3		
Activity	**Time**	**Mass Loading**	**Time**	**Attenuation Factor§**
Construction	$\frac{500\ hrs}{8760\ hrs}$	N/A		1
Discovery‡	$\frac{6\ hrs}{8760\ hrs}$	N/A		1
Gardening	N/A	565 $\mu g/m^3$	$\frac{100\ hrs}{8760\ hrs}$	0.74
Outdoor	N/A	100 $\mu g/m^3$	$\frac{1700\ hrs}{8760\ hrs}$	0.74
Indoor	N/A	50 $\mu g/m^3$	$\frac{4380\ hrs}{8760\ hrs}$	0.24
Offsite	N/A	0 $\mu g/m^3$	$\frac{2580\ hrs}{8760\ hrs}$	0

Notes:

† The NRC staff estimated the soil-to-air transfer factor for the intruder-construction scenario by accounting for environmental characteristics of the site using the equation, $T = T_o \times \left(\frac{10}{v}\right) \times \left(\frac{s}{30}\right) \times \left(\frac{50}{PE}\right)^2$, where T_o is equal to 2.53×10^{-10} [see NUREG-1759, Vol. 3 (NRC, 1981b)], v is the average wind speed at the site in m/s (v = 3.61 m/s), s is the silt content of the site soils in percent (s = 50%), and PE is the Thornthwaite precipitation-evaporation index of the site vicinity indicative of the antecedent moisture conditions (PE = 91).

‡ For stable waste, the NRC staff assumed the intruder would recognize the waste and would limit exposures once the hazard was recognized. Thus, the NRC staff limited the number of hours of exposure for waste that is stable and segregated from unstable waste. This variant of the intruder-construction scenario is known as the intruder-discovery scenario.

§ The NRC staff only considered attenuation of direct radiation in the intruder-agriculture scenario, thus the values for the intruder-construction scenarios are presented as unity (i.e., no attenuation) in this table. The NRC staff estimated the attenuation of direct radiation in the intruder-agriculture scenario assuming the source's areal extent is a circular disc using the equation, $f_{attenuation} = \frac{E(\mu \times r_1) - E(\mu \times r_2)}{E(\mu \times r_0)}$, where E(μ×r) is the first-order exponential integral; μ is the linear attenuation coefficient (μ = 0.0097 m^{-1}); and r_0, r_1, and r_2 are the radii of the circular disc geometry shown in NUREG-0782, Figure G.5. For the time spent outdoors, r_1 in the above equation equates to r_0 in NUREG-0782, Figure G.5, (NRC, 1981a) because no attenuation by structures is assumed. For time spent indoors, r_1 accounts for the attenuation afforded by the house.

To convert m^3 to ft^3 multiply by 35.3. To convert g to lb multiply by 0.0022.

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APPENDIX G
GREATER-THAN-CLASS C WASTE DISPOSAL

G.1 General Considerations

The disposal of Greater-Than-Class C (GTCC) waste in a land disposal facility is more challenging than the disposal of Class A, B, and C low-level radioactive waste (LLW) because of the higher concentrations of both short- and long-lived radionuclides. The U.S. Nuclear Regulatory Commission (NRC) designed the requirements in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 61, "Licensing Requirements for Land Disposal of Radioactive Waste," to ensure that the public is protected from the disposal of all types of LLW. The guidance in this appendix addresses technical issues relevant to the disposal of GTCC waste that are generally not applicable to the disposal of Class A, B, and C waste. GTCC waste disposal will be in a disposal facility licensed by the NRC.

A facility that has been licensed for the disposal of Class A, B, and C waste may not be suitable for the disposal of GTCC waste. Additional technical analyses are necessary to demonstrate that the performance objectives of 10 CFR Part 61 will be met for the disposal of GTCC waste.

Not all GTCC waste is the same from a health and safety analysis standpoint. Some GTCC waste will be comprised of primarily short-lived isotopes (e.g., activated metals), whereas others may be comprised of primarily long-lived isotopes. The technical analyses performed, and the regulatory review of those analyses, should be based on the GTCC waste characteristics that are important to safety for the specific waste stream.

Additionally, some GTCC waste streams may be precluded from disposal in a near-surface disposal facility but may be disposed in a specialized land disposal facility. This appendix indicates specific guidance for a specialized land disposal facility, where appropriate, otherwise the guidance in this appendix is considered applicable to land disposal of GTCC waste (i.e., both in a near-surface and a specialized facility).

G.2 Timeframes for Analyses of GTCC Waste Disposal

The analysis timeframe to use when evaluating the disposal of GTCC waste will be dependent on the radiological characteristics of the waste. Most GTCC waste will have concentrations and quantities of long-lived radionuclides that would be considered "significant" following the guidance provided in Appendix H to this document. Therefore, the compliance period to use for GTCC waste containing significant quantities of long-lived radionuclides would be 10,000 years, followed by a performance period.

Some types of GTCC waste may have concentrations of short-lived radionuclides (e.g., cesium (Cs)-137, strontium (Sr)-90) that are the main contributors to the waste classification. In these cases, the predominant risk from the materials is to an inadvertent intruder. Because the short-lived radionuclides may experience many half-lives of decay over a 500-year period, the materials may represent a substantial risk at 500 years and an insignificant risk at 1,000 years after disposal. For these types of GTCC wastes, a 1,000-year compliance period may be sufficient. The licensee should ensure that these wastes do not have concentrations of

long-lived mobile radionuclides that would impact the assessment of compliance with 10 CFR 61.41, “Protection of the general population from releases of radioactivity.”

G.3 GTCC Waste Classification

GTCC waste is LLW that exceeds the concentration limits provided in Table 1 and Table 2 of 10 CFR 61.55 “Waste classification.” If the concentration of a radionuclide listed in Table 1 of 10 CFR 61.55 exceeds the value in Table 1 or the value provided in column 3 of Table 2, the waste is GTCC. Waste exceeding the concentration limits in Table 3 of 10 CFR 61.55 are not classified as GTCC waste and are generally not suitable for a land disposal facility unless proposals for disposal of such waste in a disposal site licensed pursuant to this part are approved by the Commission. GTCC waste must meet the requirements for Class C waste and the additional requirements set forth in 10 CFR 61.56(c). The additional requirements in 10 CFR 61.56(c) are:

> (1) Waste must not generate significant heat in amounts that impact the release of radioactive material from the disposal site or the long-term stability of the disposal site.
>
> (2) Waste must not contain quantities and concentrations of fissionable radionuclides that could result in criticality.
>
> (3) Waste must not emit radiation at levels that could lead to degradation of the disposal environment conditions, such as through radiolysis.
>
> (4) Waste must not be dispersible.

The licensee is to determine the concentrations of radioactivity acceptable for disposal as GTCC using technical analyses that demonstrate that the requirements in the performance objectives can be met.

Calculation of the concentration of alpha-emitting transuranic radionuclides with half-lives greater than 5 years should follow the guidance and procedures with respect to concentration averaging found in the branch technical position (BTP), “Concentration Averaging and Encapsulation Branch Technical Position, Revision 1,” issued February 2015 (NRC, 2015c). For stabilization (waste uniformly mixed with stabilizing materials), the stabilizing agent needs to have a purpose other than reducing the waste class by dilution. Generally, wasteforms can achieve a 10 percent waste loading, which is equivalent to a factor of 10 reduction in concentrations for classification purposes. For encapsulation of waste, the guidelines are slightly more complicated, and the licensee should consult the BTP for guidance. Some averaging is still allowable, but it depends on the waste loading.

For waste that has been determined to exceed the Class C limits under § 61.55(a), the NRC has identified two waste concentration categories for land disposal of GTCC LLW. One category identifies concentrations of GTCC waste radionuclides that can be considered for near-surface disposal and the other category is for GTCC waste disposal in a specialized land disposal facility (e.g., borehole disposal that is not in the near-surface). The final ILLRWD rule provides the following threshold requirements in 10 CFR 61.55(b) (NRC, 2026a) for determining what waste could be considered for disposal in a near-surface facility and in a specialized land disposal facility:

(i) GTCC waste containing alpha-emitting transuranic radionuclides with half-lives greater than 5 years and a concentration of no more than 10,000 nanocuries per gram can be disposed at a near-surface disposal facility that meets the requirements for such a facility in 10 CFR Part 61.

(ii) GTCC Waste with concentrations of groupings of radionuclides with half-lives greater than 50 years as described in § 61.55(b)(2) can be disposed in a specialized land disposal facility. For example, alpha-emitting transuranic radionuclides with concentrations of no more than 500 microcuries per gram can be disposed in a specialized land disposal facility that meets the requirements for such a facility in 10 CFR Part 61.

(iii) GTCC waste exceeding the Table 3 values at 61.55(b) is not generally acceptable for a land disposal facility unless proposals for disposal of such waste in a disposal site licensed pursuant to this part are approved by the Commission.

A proposal to dispose of GTCC waste should do the following:

- Describe the characteristics of the waste (e.g., form, volume, radionuclides, concentrations).
- Describe the design and approaches for ensuring safety during facility operations (e.g., receiving, handling, and emplacing waste).
- Describe the disposal technology that will be used to limit the release of radioactivity to the environment.
- Provide the basis for the performance of the disposal technology over timeframes consistent with the hazards posed by the radionuclides contained in the waste.
- Demonstrate how the disposal system will provide protection to offsite members of the public and provide for intruder protection.
- Demonstrate how the disposal site and near-surface environment, as appropriate, will ensure that proper stability is achieved for the appropriate timeframe.

The list above is not comprehensive but rather is intended to provide the key considerations to disposing of concentrated alpha-emitting TRU. Licensees are encouraged to seek early interaction with the NRC staff to discuss a potential proposal and achieve agreement on the technical information that they would need to develop and provide for review.

G.4 GTCC Waste Characteristics

The following sections describe various characteristics of GTCC waste that licensees and reviewers may need to consider and evaluate prior to disposal. Because of higher concentrations of short-lived radionuclides, GTCC waste may generate sufficient heat that must be considered in the design of a GTCC waste disposal facility. Thermal considerations are discussed in Section G.4.1. Due to the potentially higher radiation levels from GTCC waste compared to typical LLW, licensees also need to consider potential impacts to storage and disposal from radiolysis and radiation effects. Radiolysis and radiation effects are discussed in

Section G.4.2. Section G.4.3 provides guidance on the safety and security of fissile materials. Section G.4.4 discusses other properties of GTCC wastes to consider, such as the form and dispersibility of the waste.

G.4.1 Thermal Effects

Waste heat may elevate temperatures to levels that, if not planned for, could degrade barrier components or substantially modify the hydraulic behavior inside the facility (e.g., large-scale redistribution of moisture due to boiling). This section discusses heat generation and its impacts, including the proper design of the disposal site to manage decay heat. Attachment G-1 contains a detailed evaluation of different disposal designs, including trench, vault, and large borehole disposal, which have different implications with respect to thermal dissipation. Attachment G-1 provides example calculations with tools, equations, and results to enable a licensee to evaluate potential thermal effects and properly design the engineered system. Section G.4.1.3 provides guidance to licensees and reviewers on managing heat produced by GTCC waste emplaced in land disposal facilities.

G.4.1.1 Heat Generation

Interaction of radiation with media generates heat. Class A, B, and C wastes generate a negligible amount of heat, because the waste has low concentrations of decaying radionuclides. However, GTCC waste, such as the activated metals generated during normal reactor operation, (e.g., reactor components and internals composed of Inconel alloys, zircaloy, or stainless steel), have large concentrations of decaying radionuclides that generate substantial decay heat. Sealed sources, such as waste from Cs-137 irradiators, also generate heat.

Heat generation due to radioactive decay, for most GTCC waste streams, is principally due to the decay of (1) nickel (Ni)-63 and cobalt (Co)-60 for activated metals from commercial reactors and (2) Cs-137 for sealed source waste. Table G-1 provides the initial power density for these radionuclides using inventories generated by operating and closed commercial reactors, estimated by the NRC (NRC, 2019, Tables A-2 and A-3) from U.S. Department of Energy (DOE) DOE/EIS-0375, “Final Environmental Impact Statement for the Disposal of Greater-Than-Class C (GTCC) Low-Level Radioactive Waste and GTCC-Like Waste,” issued January 2016 (DOE, 2016a, 2016b) (GTCC FEIS). The initial thermal output for activated metals is dominated by Co-60, but within 40 years, Ni-63 becomes the dominant heat source (Figure G-1) as the Co-60 decays. Thermal output from activated metals is larger initially than that from sealed sources, but the output from sealed sources is larger from years 45 through 100 because of the different radionuclide decay rates. In practice, it is expected that activated metal wastes will not be immediately emplaced after removal from a reactor; thus, the waste will undergo aging to some extent. Estimation of heat generation rates should account for the age of the waste at the time of disposal.

The NRC staff also evaluated the thermal characteristics of GTCC wastes that could potentially arise from the reprocessing of spent nuclear fuel in its Hazard Analysis supplement (NRC, 2026b). Reprocessing activities could result in GTCC waste with concentrations of heat generating radionuclides (e.g., Co-60, Sr-90, and Cs-137) that are comparable to and most likely have larger thermal impact than that of reactor activated metals and sealed sources. Reprocessing wastes could involve larger amounts of Sr-90 (29-year half-life) and Cs-137 (30-year half-life) that would increase the longevity of the thermal impact compared to wastes with

large amounts of Co-60 (5.3-year half-life). A specialized land disposal facility (e.g., borehole disposal) may provide more options for addressing potential thermal concerns.

G.4.1.2 Potential Consequences of Heat Generation

Once waste is emplaced in a land disposal facility and closed, waste heat is dissipated through a combination of conduction and convection from the waste to the surrounding environment. Conduction would be expected to be the dominant heat dissipation mechanism because a disposal facility normally is designed to minimize the exchange of water and air between a disposal cell and the surrounding environment; however, at high temperatures, vapor movement (convection) within the disposal cell may enhance heat dissipation. In a conduction-dominated system, the waste heat moves approximately radially down a temperature gradient from the waste canisters into the surrounding media. With a constant heat source, the interior of the cell warms and the resulting thermal plume spreads until an equilibrium is reached, with energy gain from the source balanced by energy loss from the system. With a decaying heat source, the initial thermal history is similar, but the rate of temperature increase slows as the source decays until, eventually, dissipation of heat overcomes the decreasing supply of heat, and the interior of the cell begins to cool.

Table G-18 Estimated Initial Waste Heat Generated by Radioactive Decay for DOE's GTCC FEIS Waste Streams (i.e., reactor activated metals and sealed sources)

Radionuclide	Half life (years)	Energy release (MeV/ disintegration)	Energy release W/MCi	Initial activity density (Ci/m^3)	Initial power density (W/m^3)
Cobalt-60	5.26	2.6	15,400	57,000	875
Nickel-63	100.1	0.017	101	20,500	2.1
Cesium-137	30	1.17	695	1,700	11.8

To convert Ci to Bq multiply by 3.7 x 10^{10}. To convert m^3 to ft^3 multiply by 35.3.

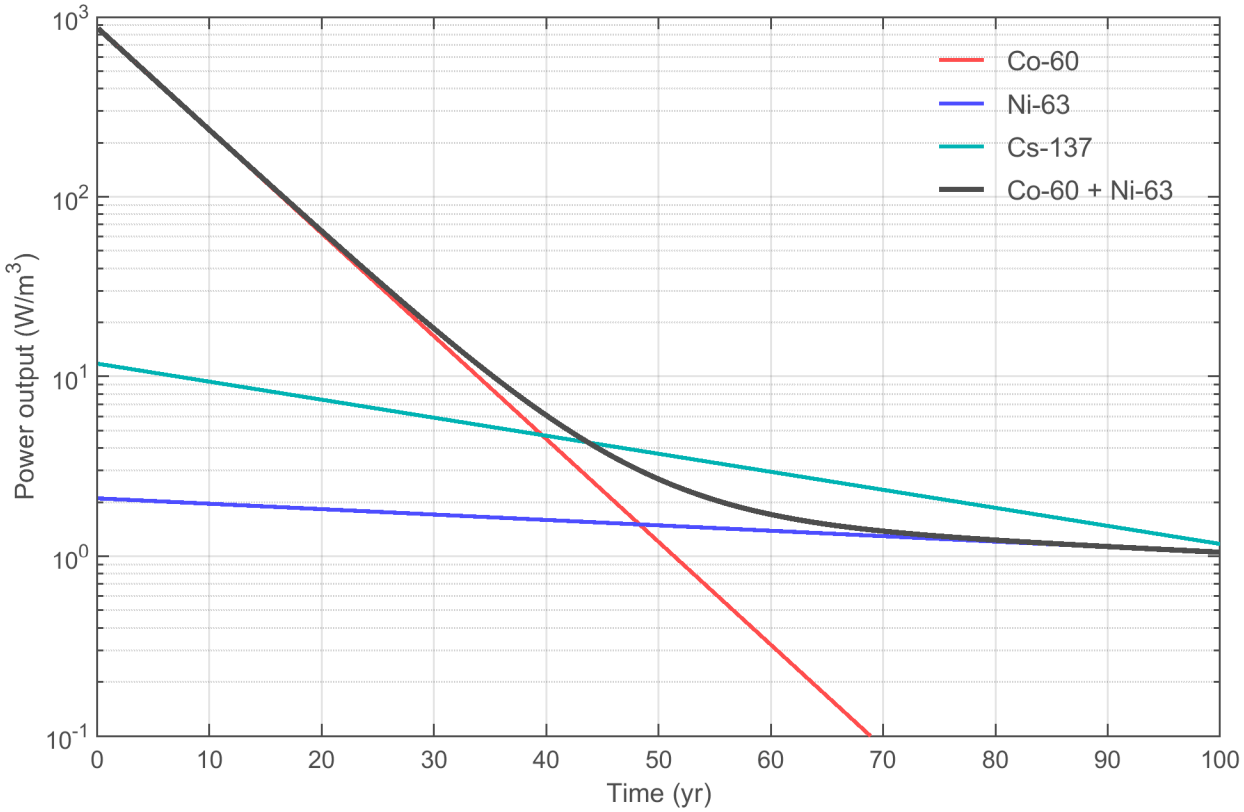


Figure G-13 Representative Power Output from Reactor Activated Metal Due to Radioactive Decay of Co-60 and Ni-63 after Cessation of Irradiation

A thermal pulse may be consequential when elevated temperatures or steep temperature gradients impact disposal system barriers, thereby increasing the probability of enhanced release of radioactivity when compared to non-thermally modified conditions. Elevated temperatures may increase the rate of release and transport of gaseous radionuclides because of convection of gases. Enhanced pathways may consist of increased material permeability, connected cracks or gaps, or openings in barriers above the canisters or within the disposal cell. Potential degradation mechanisms include the following:

- loss of free water and dehydration in cement paste, causing degradation of the pore structure and enhancing phase transformation processes

- cracking of concrete or degradation of the cement pore structure from (1) differential thermal expansion of cement and aggregates, (2) differential expansion or contraction along thermal gradients, differential heating, or differential cooling, or (3) increased pore pressures

- differential expansion or contraction stresses between concrete and other materials

- enhanced degradation of geomembrane polymers that weakens stress resistance

If thermal effects are possible, a licensee should assess the features, events, and processes (FEPs) that are important to safety to determine potential changes in response to elevated temperatures. Not all changes will necessarily be negative. For instance, drying of surrounding geologic materials could reduce the rate of aqueous diffusion of radionuclides.

G.4.1.3 Thermal Management Guidance

The design of the disposal facility in concert with the characteristics of the waste are primary considerations when evaluating decay heat management. A licensee may want to dispose of different volumes or types of GTCC waste. Thermal management is more difficult with large volumes and certain types of waste. A specialized land disposal facility (e.g., borehole disposal) may be more appropriate for addressing potential thermal concerns.

The age of the waste at the time of disposal, as well as the duration of disposal facility operations are important considerations. The isotope Co-60 is a key radionuclide with respect to thermal management. LLW generated (e.g., activated metals) from nuclear reactor operation is not immediately sent to a disposal facility. Because the half-life of Co-60 is approximately 5 years, the decay heat output of activated metal waste decreases about an order of magnitude for every 16 years of storage. Concentrations of heat generating isotopes in reprocessing wastes will also be impacted by the age of the spent fuel when reprocessed and time periods the reprocessing wastes are to be stored prior to shipment to a disposal facility that would reduce the thermal output of the waste at the time of disposal.

Ni-63 and Co-60 in activated metals and Cs-137 in sealed sources are the primary radionuclides in DOE's GTCC FEIS that a licensee would need to evaluate when considering the effects of heat generation. For waste inventory that does not include these radionuclides, an evaluation of thermal effects is likely not needed. Future nuclear development, such as advanced reactor designs and reprocessing of spent nuclear fuel could generate wastes with different thermal profiles. Licensees should consider the thermal profiles of new waste streams rather than basing their decisions on the presence or absence of particular radionuclides (DOE, 2020).

For waste that does generate heat, the licensee would need to determine whether that heat could cause, or lead to, degradation of significant disposal system components. As demonstrated in Attachment G-1, an analysis needs to account for the thermal properties of the materials used to determine the peak temperature. For example, the thermal properties of the wasteform would need to be determined for a borehole disposal system in addition to the thermal properties of the cementitious or granular fill material and the side walls of the borehole. To determine the temperature field from a planned disposal facility design, calculations to determine thermal properties of materials used such as concrete cylinders, remote-handled

waste packages, backfill, and specific earthen materials or soil may also be needed. An evaluation of disposal vaults would need to include the concrete walls, floor, and ceiling.

If potential temperatures are such that deleterious effects are possible, two basic strategies can reduce temperatures within a disposal cell: (1) reducing the overall thermal density of heat-generating waste (e.g., reduce the amount of waste per canister, spread the waste canisters within a larger volume, allow the waste to age) and (2) enhancing dissipation of the generated heat to the environment. Consideration should be given to allowing the waste to age (i.e., be kept in protective storage) before final disposal. Because more heat tends to be generated by short-lived radionuclides, aging is a dominant factor, although it may not be under the control of the disposal facility operator. The operator may specify limiting heat generation rates (e.g., kilowatt (kw)/m^3) as part of the waste acceptance criteria.

If aging the waste before disposal does not reduce the waste's heat generation sufficiently, the licensee should consider managing the heat load within the disposal cells. The range of options explored in the example calculations provided in Attachment G-1 suggests that geometric factors related to packing strategies may affect peak temperatures by up to a factor of 4, all other factors being equal. Table G-2 summarizes emplacement power density within a representative waste container that limits the maximum temperature increase to 40 degrees Celsius (°C) (104 degrees Fahrenheit (°F)) (i.e., roughly 60°C (140°F) accounting for the ambient temperature in a warm climate). Limiting the maximum temperature increases to less than 60°C (140°F) should ensure that the complex processes associated with boiling and redistribution of moisture will not occur and do not need to be further evaluated. Although Table G-2 was based on waste streams in DOE's GTCC FEIS and specific facility designs, Attachment G-1 provides sufficient detail that a licensee may use the described approaches to scale the results in Table G-2 to their disposal system, including the specific waste to be disposed.

Table G-2 Summary of Emplacement Power Density (W/m^3) Creating a Maximum Rise of 40 °C for Representative Disposal Configurations

	Effective Thermal Conductivity (W/m-K)						
Peak ϕ (m^2)	0.2	0.25	0.3	0.35	0.4	0.45	0.5
0.5	16.0	20.0	24.0	28.0	32.0	36.0	40.0
1	8.0	10.0	12.0	14.0	16.0	18.0	20.0
1.5	5.3	6.7	8.0	9.3	10.7	12.0	13.3
2	4.0	5.0	6.0	7.0	8.0	9.0	10.0
2.5	3.2	4.0	4.8	5.6	6.4	7.2	8.0
3	2.7	3.3	4.0	4.7	5.3	6.0	6.7
3.5	2.3	2.9	3.4	4.0	4.6	5.1	5.7

To convert m to ft multiply by 3.2805. To convert a unit of K to F multiply by 1.8

The initial heat load generated by wastes containing significant amounts of Co-60, Sr-90, or Cs-137 (e.g., activated metal and reprocessing wastes) is large enough that thermal issues are likely to occur without proper design and management. Much of the existing GTCC waste is already aged; after four or five decades of aging, some of the strategies examined will accommodate the generated heat at well below boiling temperatures. Some care may be required to manage the Cs-137 sealed source waste heat output, because the initial power

density of the source is equivalent to 25-year activated metal and newly generated reprocessing wastes with large amounts of Sr-90 and Cs-137.

G.4.2 Radiolysis

G.4.2.1 Introduction

Due to the potentially higher radiation levels from GTCC waste compared to typical LLW, licensees need to consider potential impacts from radiolysis and radiation effects (here collectively termed "radiolysis") on the storage, transportation, and disposal of GTCC wastes. In discussing source term, Section 3.2 of the main report notes that radiolysis may be a factor in estimating the degradation time of the waste container. Section 3.2 also discusses how radiolysis may generate gases that can affect LLW facility performance. Radiolytic gas generation was identified as a potential hazard resulting from GTCC disposal in "NRC Staff Analyses Identifying Potential Hazards Associated with the Disposal of Greater-Than-Class C and Transuranic Waste," issued February 2018 (NRC, 2018a). The DOE addresses the operational safety implications of radiolytic gas generation for LLW facilities in the criteria and review and approach document for safe disposal of LLW (DOE, 2019a) and the DOE radioactive waste management manual (DOE, 2021). Radiolytic gas generation is within the scope of 10 CFR 61.56(a)(5). In addition, radiolysis effects are included in the FEPs discussion in Appendix C to this document.

A general definition of radiolysis is "chemical decomposition by the action of radiation" (NRC, 2015a, page 12-12). The effects of radiolysis are dependent on not only the radiation level but also the radiation type (alpha, beta, gamma, or neutron). The general areas of potentially negative impacts on safety and containment are (1) gas generation, (2) changes to water chemistry, and (3) radiation damage in solids. These may be more likely for GTCC wastes than for Class A, B, or C LLW.

Radiolysis of water and other hydrogenous materials (e.g., cementitious materials and organic solids and liquids) can lead to the generation of hydrogen gas (H_2). NRC (2019) specifically cites radiolytic gas generation among the phenomena that could be more significant for GTCC wastes. Radiolytic gas generation can exceed gas generation from other processes such as corrosion in more highly radioactive wastes (Rodwell et al., 2003). If H_2 can accumulate to above around 5 percent of a mixture, it can be flammable if oxygen is also present. Ignition could cause an explosion or fire that could endanger workers or damage the waste, containers, or other structures in a facility. In addition, if the H_2 accumulates in a confined space, such as a tightly sealed waste container, the wasteform, the binders, or the waste containment may mechanically fail from overpressurization even in the absence of ignition.

Radiolysis of water and other substances could lead to a more corrosive aqueous environment. Irradiation can alter the pH, redox, and aqueous concentrations of potentially corrosive radicals and species such as hydrogen peroxide (H_2O_2) and nitric acid (HNO_3) or lead to the formation of organic acids. These changes could make the corrosion of cementitious binders or metallic waste containers more likely or could modify the aqueous speciation of radionuclides, potentially enhancing radionuclide release from GTCC wastes.

Irradiation may more directly affect the integrity of solids such as plastics, cements, geomembranes, or ion-exchange media. These changes (e.g., cracking and disintegration)

could affect estimates of performance if these solids are relied on to contain radionuclides. Irradiation effects on metals are expected to be minor for GTCC and other LLW.

The following discussion is organized around the two stages of GTCC waste management once the wastes are received by the disposal facility operator: onsite storage and disposal. Transportation is not addressed because the timeframe for transportation is relatively short compared to the other stages, and transportation of nuclear materials is regulated under other requirements.

Important assumptions in this analysis are:

- Freestanding water in packages will be removed to below 1 percent of the waste volume (and therefore less than or equal to 1 percent of the package volume) in compliance with 10 CFR 61.56(a)(3).

- Packages will be stabilized by filling the void spaces with cement-based grout (DOE, 2016b, Appendices B and E) or other appropriate waste forms (e.g., grouted or glass wasteforms for reprocessing wastes).

- All packaging will be in sealed metal containers.

Waste acceptance criteria may be used to address radiolysis in GTCC waste storage and disposal. Chapter 8 of this document describes the three components of waste acceptance under 10 CFR Part 61: waste acceptance criteria, waste characterization, and waste certification. The implementation guide for use with the DOE radioactive waste management manual (DOE, 1999) provides an appropriate discussion of waste acceptance criteria that is also applicable to this NRC guidance. Waste acceptance criteria establish the conditions or parameters that the waste and waste containers must meet for a specific facility to ensure that the waste can be safety managed. As the designer and operator of the waste facility, the licensee is likely to be in the best position to establish the waste acceptance requirements or criteria specific to that facility, with oversight from its regulator. Some of these criteria, like freestanding water content identified in the preceding list, serve as inputs when assessing potential impacts of radiolysis and can possibly obviate the need for additional analyses.

G.4.2.2 Radiolysis During GTCC Waste Storage

It is anticipated that, in some cases, storage of packaged GTCC wastes for extended time periods—up to years—may be necessary. Adverse effects of radiolysis, therefore, will (1) have more time to develop compared to transportation timeframes and (2) could affect not only short-term operational safety but also the long-term integrity of the wastes and their packaging. This subsection emphasizes the potentially adverse effects that would be manifest during storage. As noted in Section G.4.2.1, it is possible that waste recipients will impose waste acceptance criteria that are based, in part, on limiting the potential for adverse impacts of radiolysis during storage. These criteria should be relied on only if they are supported by a sound technical basis specific to radiolysis.

A useful discussion of areas of concern during long-term (for more than a few months) storage of LLW can be found in NUREG/CR-4062, ""Extended Storage of Low-Level Radioactive Waste: Potential Problem Areas," issued in 1985 (NRC, 1985). Given the higher radioactivity of GTCC waste compared to the wastes generally considered in that report, its findings may be applicable

to even short-term storage of GTCC waste. The impact from radiolysis is a product of time and radiation levels. The main areas of concern from radiolysis, identified in that report and corroborated in the subsequent literature, are as follows:

- flammability from the generation of H_2, including potential explosion, which has implications for operational safety
- compromise of container integrity from the generation of H_2, including potential explosion or rupture, which has implications for operational safety
- container corrosion from radiolytic chemical effects, with possible operational safety implications for long storage times

The NRC standard for preventing radiolytic flammability is that H_2 comprises less than 5 percent by volume of the gas inside the package. For preventing mechanical package damage, the potential pressure increase in a tightly sealed package should be compared to the pressure rating of the package.

The potential for H_2 gas generation can be estimated most readily by calculations using established $G(H_2)$ values, such as those outlined in NUREG/CR-6673, "Hydrogen Generation in TRU Waste Transportation Packages," issued May 2000 (NRC, 2000a). (See also, for example, Flaherty et al., 1986; NRC, 1985; Nuclear Waste Partnership, 2021.) These calculations for a given container require values for masses of water and other hydrogenous materials and radiation levels. The most efficient approach may be to perform calculations only for bounding packages that have radiation levels and hydrogenous material masses most likely to lead to significant H_2 generation, to determine whether further analysis is necessary. Aside from water, the NRC staff expects that hydrogenous materials that should be included in H_2 calculations are cementitious materials such as void-filling grout, ion-exchange resins, plastics, and other organic substances. It is noted that H_2 generation rates calculated from $G(H_2)$ values based on laboratory measurements of water and pure materials are likely to be conservative relative to actual generation rates (EPRI, 1988; Devarakonda, 2006).

Packages for GTCC wastes can contain a wide array of materials, including plastics, ion-exchange resins, fabrics, paper, and other organic substances, and they will also contain substantial masses of void-filling grout. These packages, therefore, present the greatest potential for H_2 generation during storage. Calculations of H_2 generation should be performed for bounding packages to determine whether further analysis is necessary. Freestanding water in these packages may be minor after grouting, but gas generation analyses need to consider unbound water associated with the grouts themselves (Dole and Friedman, 1986). As noted previously, the calculational approach to gas generation in NUREG/CR-6673 using supportable $G(H_2)$ values for package contents is used in certain cases by the NRC staff for confirmatory calculations and is cited in the context of GTCC in NRC (2018a). Numerous compilations or studies of G values (e.g., Dole and Friedman, 1986) and other quantitative approaches (e.g., Esnouf et al., 2022) have been developed.

In the case of waste packages containing activated metals, wasteforms resulting from reprocessing of spent nuclear fuel, and sealed sources; solid hydrogenous materials are expected to be absent or present only in small masses. Therefore, the only significant H_2 generation concern in these package types is from freestanding water. The 1 percent limitation on freestanding water in 10 CFR 61.56(a)(3) was developed with consideration of release and

transport of radioactivity. The 1 percent free liquid requirement may be protective for H_2 generation, but this should be confirmed by calculations for the waste packages containing the highest radioactivity levels. A simplified calculation for the maximum potential gas pressure from water radiolysis in a waste package in the absence of other radiolysis sources was performed for an FEPs analysis for high-level waste disposal at the proposed Yucca Mountain repository (SNL, 2008, page 6-435). Based on an assumed mass of residual water, a temperature, a void space volume, and a maximum water vapor pressure was calculated using the ideal gas law. If the water vapor were radiolytically converted to H_2 and oxygen (O_2), the pressure would increase by 50 percent, because each mole of water (H_2O) vapor is converted to one mole of H_2 and 0.5 mole of O_2. The maximum radiolytic H_2 and H_2+O_2 pressures could be considered in the context of other gases in the package. This maximum gas generation approach may be useful in cases where small water volumes can be assumed.

A licensee can determine whether hydrogen generation is a concern by estimating the potential for H_2 gas generation, as described in the preceding paragraphs, and comparing the estimates to appropriate thresholds. In addition, a licensee can measure the hydrogen levels within the waste containers during storage to determine whether hydrogen generation is a concern. If it is a concern, several approaches can prevent or mitigate potential hydrogen production during storage, including demonstrating removal of water, ventilating to avoid container overpressurization and buildup of explosive gases, and monitoring of hydrogen and explosive gases. Ventilation of waste packages can be achieved by several methods, including pressure relief devices such as vent clips and filtered vents designed to control the buildup of hydrogen (DOE, 1999; NRC, 1985). Venting the waste container mitigates overpressurization and buildup of explosive gases, but it also results in the potential release of radiological airborne effluents. Therefore, venting must include filtering and monitoring for radiological constituents in addition to monitoring for hydrogen and explosive gases (DOE, 1999; NRC, 1985). As described in NUREG/CR-4062, radiolytic gas generation is likely to be a problem only for the highest activity LLW, and especially for those wastes containing organic material or water.

Monitoring for flammable or explosive gas mixtures would be advisable, especially during initial storage (e.g., the first 15 years). For example, at the Three Mile Island Unit 2 (TMI-2) independent spent fuel storage installation (ISFSI), the core debris inside the TMI-2 canisters was originally treated as having the potential for H_2 generation due to radiolysis (Areva Federal Services and Spectra Tech, Inc., 2018). Based on this consideration, the dry shielded canister was modified to include venting through high-efficiency particulate air (HEPA) grade filters during storage. This venting system allows for the release of the H_2 and for monitoring, purging, or both of the system during operation. This facility established a threshold (i.e., limiting condition of operation) that the H_2 concentration inside each canister shall not exceed 10 percent of the explosive limit of 5 percent (i.e., 0.5 percent or 5,000 parts per million). Canisters that reach this action threshold are vented and purged to lower the hydrogen content. A useful discussion of the hydrogen monitoring of the TMI-2 ISFSI can be found in the license renewal application (Areva Federal Services and Spectra Tech, Inc., 2018). From 2004 to 2014, the annual monitoring of the hydrogen levels within the dry shielded canisters indicated a steady-state production of approximately 0.04 percent hydrogen, which is about 10 percent of the facility action threshold and 1 percent of the hydrogen explosive limit (Areva Federal Services and Spectra Tech, Inc., 2018).

In addition to ventilating individual waste containers, ventilation of the storage area can address potential hydrogen accumulation within the area (DOE, 1999; NRC, 1985). If gas generation is only a concern for some of the waste containers at a facility, these containers could be

segregated, and the ventilation could be localized. As described in NUREG/CR-4062, should a ventilation system be warranted, factors to consider in the system design include the potential for the waste to release gaseous radiological effluents, the need to control ambient conditions (e.g., humidity and temperature) for maintaining waste package integrity, and the potential for localized accumulation of flammable gases. Venting for storage areas includes filtering and monitoring for radiological constituents as well as monitoring for hydrogen and explosive gases (DOE, 1999). Regardless of the ventilation, surveillance plans can include visual checks for container overpressurization (e.g., bulging).

Another method that may be considered to mitigate radiolytic hydrogen generation during storage is the use of passive autocatalytic recombiners. The purpose of the catalysts, typically based on platinum or palladium, is to recombine radiolytic hydrogen and oxygen to prevent the buildup of potentially flammable H_2 gas and attendant pressure increase. While catalytic recombiners are commonly deployed in nuclear power plant containment for H_2 control during accidents (NEA, 2014), they have also been used to prevent H_2 buildup during storage or transportation of radioactive wastes from the TMI-2 (Henrie and Appel, 1985; Reno and Schmitt, 1988; Areva Federal Services and Spectra Tech, Inc., 2018) and Fukushima (Reinecke et al., 2021) accidents.

Radiolysis-induced corrosion can impact storage if liquid water is present, from either inorganic acid, H_2O_2, and O_2 generation from water radiolysis or organic acid generation from radiolysis of organic substances such as cellulosic materials. The latter process may be mitigated by filling packages with grout. In addition to potential corrosion of metal components and containers, as stated in NUREG/CR-4062, “Corrosion of binder materials and, in particular, of cement by radiolysis products is another possible area of concern. If significant degradation should occur during storage, the transportation and disposal properties of the wasteforms could be compromised.” To evaluate the potential for radiolysis-induced corrosion of materials during storage, calculations of radiolysis product generation in water can be performed similarly to those for H_2 generation. Appendix C to NUREG/CR-4062 provides example calculations, but the G values used should be based on the most up-to-date data. A key factor in these calculations will be the volume of freestanding water available in the package to produce corrosive liquids, as limited by 10 CFR 61.56(a)(3). If these analyses do not rule out the potential for radiolysis-induced corrosion during storage, monitoring is recommended. Because direct observation of radiolytic corrosion during storage will be impractical, a possible approach to evaluating its potential can be seen in the monitoring program for the TMI-2 ISFSI. As noted on page 3-10 of Areva Federal Services and Spectra Tech, Inc. (2018), the “presence of any significant hydrogen” inside a waste package was viewed by the NRC staff as indicative of corrosion. Therefore, a hydrogen-monitoring program is used to monitor for corrosion inside packages.

Direct irradiation effects (e.g., oxidation and cracking) on waste solids such as plastics, ion-exchange media, or other organic materials are not expected to have significant adverse impacts on the safety of GTCC storage. Because non-metallic “other” wastes are assumed to be encased in grout inside waste packages, degradation of material properties of solid package contents other than the grout will not compromise storage safety. Section 3.2.1 of NUREG/CR-4062 states, in the context of extended storage of LLW, “Cement as a binder material is expected to show good radiation resistance.” This statement is supported by references to high-dose experiments, so it would also apply to GTCC waste disposal. Nevertheless, the potential for irradiation damage to package-filling grout (e.g., cracking) should be considered using data, particularly considering that elevated temperatures could also affect

cementitious material integrity. Considerations should include whether grout cracking would present a safety issue for storage. The literature on radiation-cementitious materials interactions is extensive. The information in NUREG/CR-7280, "Review of Radiation-Induced Concrete Degradation and Potential Implications for Structures Exposed to High Long-Term Radiation Levels in Nuclear Power Plants," issued July 2021 (NRC, 2021), though focused on nuclear power plant containment, may also be informative with respect to the lower radiation fields attendant on GTCC waste storage and disposal.

If in-package irradiation effects that can adversely affect storage cannot be ruled out by calculation or citation of applicable experimental data, mitigation and direct monitoring may not be practical. Hydrogen monitoring could, as for corrosion, provide a proxy for radiolytic activity. The most robust approach to mitigate irradiation effects and radiolysis-induced corrosion of in-package materials during storage, if needed, is to use calculations as part of waste acceptance criteria to limit package radiation levels.

G.4.2.3 Radiolysis During GTCC Waste Disposal

Table C-9 of this report—a generic list of FEPs for modeling the performance of near-surface radioactive waste disposal facilities—lists three example FEPs related to irradiation and radiolysis:

- Radiation Effects on the Waste Closure System (SRR, 2012, FEP 4.5.02)
- Radiolysis Effects (SRR, 2012, FEP 4.5.07)
- Radiological Effects (Neptune, 2011, FEP 55)

As applied to land disposal of GTCC waste, these FEPs involve degradation of cementitious materials, wasteform, and containment; gas generation; and water chemistry changes, including attendant impacts on radionuclide mobility.

Although developed for a performance assessment for deep disposal of high-level waste, the radiolysis related FEPs for the Yucca Mountain license application (SNL, 2008) are generally applicable to GTCC waste disposal. The main FEP for radiolysis considered an array of potentially adverse impacts, including package pressurization, fire, enhanced corrosion of metals, enhanced wasteform degradation, and enhanced radionuclide mobility. Radiolysis was eliminated from detailed consideration in the performance assessment, but the arguments for elimination of this FEP were site and design specific and cannot be applied directly to GTCC waste disposal. Many of the Yucca Mountain screening arguments, however, relied on the low water quantity in the packages—which may also apply to many GTCC wastes.

Taken together, these FEP analyses and the discussion of radiolysis during storage in Section G.4.2.2 (see Section 3.2 of the main report; NRC, 2018a; NRC, 2019) suggest that the potentially adverse impacts to disposal safety from radiolysis that need to be considered in a performance assessment can be summarized as follows:

- container structural degradation from fire, explosion, or overpressure
- disposal site stability degradation from fire or explosion
- container degradation from corrosion
- wasteform degradation
- radionuclide mobility enhancement

Structural degradation of waste containers from radiolytic gas generation during disposal can be evaluated in the same ways as outlined in Section G.2.4.2 for storage. The key differences are that (1) much longer time frames must be considered, (2) the radiation field will decline during disposal, (3) active mitigation is not possible, and (4) impacts will differ. In calculations of radiolytic gas generation, a licensee must consider the duration of disposal addressed in the performance assessment, but this potential for greater gas generation will be partially mitigated by the decreasing radiation levels over time. The inability to actively mitigate (e.g., by venting) means that gas generation effects, if credible, must be analyzed with respect to probability, consequences, or both. It is also important to consider whether breaching of the waste containers has a negative impact in the context of long-term performance assessment. A licensee may elect to not take credit for package containment in the performance assessment. In that case, non-explosive package rupture is a radiolytic effect that does not need to be evaluated.

A potentially more consequential outcome from gas generation would be explosive container failure that structurally degrades the surrounding engineered system components. For example, impacts to the backfill and cover at a near-surface disposal facility could lead to markedly increased water flow to the container. Though potentially more consequential, the risk is likely mitigated because the probability of occurrence is low, but if such rupture occurred it would need to be evaluated. Because H_2 generation in a buried, sealed waste container would be a gradual process, container failure is much more likely to take the form of non-explosive cracking. Likewise, ignition sources will be absent inside the package such that an H_2 fire is not likely, and H_2 released from a package will be mixed with ambient air or pore gases (See SNL, 2008, page 6-774). Degradation to site stability from radiolysis is unlikely.

Container degradation from corrosion during disposal can also be evaluated using the same tools discussed for storage. Longer timeframes for generation of corrosive species and for exposure of metal surfaces to those species mean that corrosion is more credible for disposal than for storage. If containment is not credited in the performance assessment, container degradation from corrosion will not be relevant to performance, and thus not need to be evaluated. If evaluation of containment is needed, the calculation of corrosive species generation can use G values and assumptions of masses of water and other hydrogenous substances, as noted in Section G.2.4.2, and use defensible corrosion rate data. For certain containers (e,g., activated metals and sealed sources) and certain reprocessing waste forms (e.g., grout and borosilicate glass), simplified calculations (SNL, 2008, page 6-435) may suffice to show that corrosive species generation will be negligible or insufficient to affect package integrity. At high radiation levels, it is possible that the dose on the exterior of a package could be sufficient for radiolytic corrosion effects if water is present.

Depending on the radiation fields and disposal facility design, radiation effects could extend to other engineered barriers. For example, geosynthetics may be susceptible to radiation damage if exposed to sufficient levels of radiation. Testing to demonstrate performance has generally not evaluated radiation effects. Depending on the proximity of other engineered barriers to the waste testing may be necessary to demonstrate performance. Embrittlement and cracking of different materials have been observed in the nuclear field. Studies have examined the radiation damage of geosynthetics when they are used as a liner and contacted by leachate with low levels of radioactivity (Tian et al., 2017). Those studies estimated that radiation damage would not be limiting to the service life. Licensees should apply caution when applying the results of those studies to their waste and facility designs. The total amount of absorbed energy from radiation should be considered. For example, a geosynthetic liner placed above a

large column of depleted uranium can act as a barrier to the upward migration of radon gas. The radon gas will then decay at the barrier and the daughter products in the decay chain may deposit their decay energy in the geosynthetic liner.

Radiolytic degradation of wasteforms during disposal can result from (1) cracking or other physical disruption from gas generation, (2) aqueous corrosion, or (3) irradiation damage to solids. The initial consideration is whether wasteform performance is credited in the performance assessment. If not, these effects will not be relevant and do not need to be evaluated in the performance assessment. If wasteform performance is credited, the same tools as used for storage evaluation can be applied. A key difference is that, should a waste package be breached, substantially more water could be available as a source of radiolysis products that could promote physical and chemical degradation of wasteforms. In other words, non-linear effects may occur.

Should a GTCC waste package be breached, and its interior exposed to water, radionuclides could be mobilized from the wasteform. Radiolysis of the water interacting with waste and transporting radionuclides could affect how readily the radionuclides are mobilized (e.g., SRR, 2012, FEPs 4.5.02 and 4.5.07; SNL, 2008, pages 6-786 to 6-788). Most notably, radiolysis can increase the oxidation potential of the water (because the main reducing radiolytic species, H_2, has low solubility) by generating strongly oxidizing H_2O_2. In addition, pH could be lowered, and dissolved oxygen increased. Many radioelements with isotopes that are potentially important to dose (e.g., technetium and actinides) are more soluble and less sorptive under more oxidizing conditions. For performance assessment analyses, these effects would be manifested by higher release rates, increased solubility limits, and lower sorption coefficients.

Calculations with an assumed radiation field in the presence of water at the modeled time of package breach can estimate the potential for altering the mobilization parameters within the in-package geochemical environment. If the calculations indicate significant impacts, the nominal parameters for release can be adjusted to account for radiolysis. Alternatively, it could be argued that, for disposal in the near surface, conditions are already strongly oxidizing, and radiolysis will not appreciably increase radionuclide mobility (analogous to arguments in SNL, 2008, page 6-787).

G.4.2.4 Summary

The elevated radiation levels of GTCC waste compared to Class A, B, and C waste increase the likelihood of radiolytic effects on the waste facility, containers, and wasteforms during storage and disposal. In the context of storage safety and disposal site performance, these effects could potentially generate dangerous or destructive gases, degrade the container, degrade the wasteform and other engineered barriers, or increase radionuclide mobility. Given a specific or bounding waste package, licensees can use calculations to help determine the probability of these processes and, if further analysis is needed, identify mitigation strategies that can be adopted. If necessary, the effects can be considered in safety and performance assessments. Alternatively, waste acceptance criteria can be used to avoid conditions conducive to adverse radiolysis impacts.

G.4.3 Fissile Materials

Appendix G, “Requirements for Transfers of Low-Level Radioactive Waste Intended for Disposal at Licensed Land Disposal Facilities and Manifests,” to 10 CFR Part 20, “Standards for

Protection Against Radiation," requires that a waste generator, collector, or processor that transports LLW or offers LLW for transportation must prepare a manifest reflecting information relevant to the amount of fissile material, among other properties. In particular, NRC Form 541, "Uniform Low-Level Radioactive Waste Manifest (Container and Waste Description)," requires the following information:

- For the total shipment:
  Item 1: special nuclear material (SNM) (in grams) according to U-233, U-235, and Pu

- On a container basis:
  Item 9: waste weight in kilograms
  Item 13: approximate waste volume in container
  Item 16: radiological description—individual radionuclides and activity (megabequerel (MBq)) and container total; or container total activity and radionuclide percent

The container information in item 16 is provided in the units of activity and may be in units of Ci or Bq. Table G-3 provides conversion values to allow for determination of concentrations and mass for fissile radionuclides for either set of units. As discussed in Section G.6.3, it will be useful to determine mass concentration of fissile material in the non-fissile waste material.

G.4.4 Other Characteristics

Other characteristics of GTCC waste that may need to be considered include its physical and chemical properties, especially as they pertain to dispersibility. GTCC waste must meet the waste characteristics requirements found in 10 CFR 61.56, "Waste characteristics." High specific surface area (e.g., m^2/g) results in higher reaction rates between the waste and environmental media, especially water. The NRC staff anticipates that the primary method to reduce the dispersibility of the waste will be use of stabilizing materials. Engineered barriers such as containers and vaults may also be used.

Table G-3 Specific Activity Values of Relevant Isotopes

Isotope	Specific Activity (Ci/g)	Specific Activity (MBq/g)
U-233	9.7×10^{-3}	$3.6 \times 10^{+2}$
U-235	2.2×10^{-6}	8.0×10^{-2}
U-238	3.4×10^{-7}	1.2×10^{-2}
Pu-238	$1.7 \times 10^{+1}$	$6.3 \times 10^{+5}$
Pu-239	6.2×10^{-2}	$2.3 \times 10^{+3}$
Pu-240	2.3×10^{-1}	$8.4 \times 10^{+3}$
Pu-241	$1.0 \times 10^{+2}$	$3.8 \times 10^{+6}$

The benefits of limiting the dispersibility of the waste include the following:

- reducing release rates of radioactivity to water and air

- making the waste more recognizable as not being natural materials in an inadvertent intruder scenario (i.e., reducing exposure times and exposure pathways)

- reducing the potential impacts from accidents that may occur during operations

The NRC staff did not provide numerical requirements in 10 CFR Part 61 for dispersibility (e.g., specific surface area, particle sizes) because the necessary requirement is site-specific, and there are numerous appropriate methods to mitigate the impacts from dispersible waste. Licensees can use technical analyses to determine whether conditioning the waste is needed and what type of conditioning may be appropriate. Conditioning is the preparation of waste for disposal such as reducing the reactivity and conversion into a solid physical form. Some types of GTCC waste, such as activated metals, are already in a form that has low dispersibility. Other types of GTCC may need conditioning to reduce potential impacts or to provide defense-in-depth to mitigate large uncertainties. Cementitious wasteforms are commonly used in LLW disposal because the materials are relatively inexpensive and there is a large experience base in their use. However, other wasteforms may be appropriate. Other wasteforms used internationally for LLW disposal include glass, ceramic, Synroc, and bitumen. Encapsulation is an appropriate method to reduce dispersibility. Encapsulation is typically used for discrete items that cannot be well mixed with a stabilization medium. A licensee should consider disruptive events and accident scenarios that may result in failure of the encapsulation media and in release of the dispersible waste within. Section 3.2.4 of the main report provides additional guidance on the development and use of wasteforms.

Use of robust wasteforms can mitigate the impacts on all the performance objectives, whereas other structures, systems, and controls may only be effective for one performance objective. For example, use of a deflector plate to protect high-activity GTCC waste from being contacted in an intruder drilling scenario would not provide tangible benefits with respect to operational safety. The NRC staff should review the licensee's technical basis for waste dispersibility, including proposed barriers, actions, and controls to mitigate the impacts from dispersible waste. The regulator should ensure that the form of the waste assumed in the technical analyses for the performance assessment, intruder assessment, and operational safety assessment is consistent with the form of the waste proposed for disposal.

G.5 Intruder Barriers for GTCC Waste

Intruder barriers are intended to deter, discourage, or inhibit a human or group of humans from continuing to engage in actions such as drilling, trenching, or excavating that would eventually bring them into contact with radioactive waste. Simply put, an intruder barrier is intended to convince the inadvertent intruders to cease their efforts. An intruder barrier is not intended to be impossible to penetrate, break, or remove using normal construction equipment, such as backhoes, front end loaders, compressed air demolition hammers, bulldozers, or domestic water well drilling rigs. To achieve deterrence, a barrier need only be able to convince the intruders that the material they are trying to drill though, trench, or remove is unexpectedly resistant to their efforts to the degree that they decide their best course of action is to stop and investigate or relocate their activity. In many cases, convincing intruders to cease construction activities may be a simple matter of forcing them to significantly exceed the time allocated, their budget, or a well driller's willingness to allow expensive equipment to be heavily damaged.

The requirements for intruder protection are specific to the facility type (i.e., near-surface versus specialized disposal facilities). The guidance in Sections G.5.1 through G.5.4 primarily applies

to near-surface disposal facilities and could be relevant, in part, to specialized land disposal facilities. Additional guidance for specialized land disposal facilities is provided in Sections G.5.5 and G.7.

This section provides guidance to licensees (or applicants) and reviewers on choosing and testing physical barriers for deterring domestic water well drilling at near surface disposal facilities for GTCC waste. Section G.7 also provides additional considerations for deterring inadvertent intrusion into a disposal unit, such as use of advanced wasteforms, use of disposal units with a small footprint, disposing of waste at greater depth than near-surface disposal, and siting facilities away from natural resources that could motivate an intruder to drill near a disposal site. The main natural resource considered in this section is drilling to install a water well. In petroleum-producing regions, intrusion due to oil and gas drilling may need to be considered. However, setting up an oil well drilling rig is very expensive, and most companies conduct extensive geophysical surveys to locate traps and productive units before drilling starts. Similar considerations would also apply to mining for mineral or rare metal resources. In addition, passive institutional controls such as deed restrictions will be in place and are expected to inhibit unwanted future activities at the disposal site. Geophysical surveys would likely reveal the presence of disposal units, so this guidance focuses on inadvertent intrusion due to domestic water well drilling for which geophysical surveys are not commonly conducted in advance. Despite these anticipated controls, as a defense-in-depth measure, licensees should site disposal facilities away from known natural resources to reduce the chance of an inadvertent intruder disrupting waste while attempting to access resources.

Existing regulations for near-surface disposal of Class C waste require the construction of an intruder barrier if the final engineered surface cover is less than 5 m (16 ft) thick. For near-surface disposal, the revised 10 CFR Part 61 requires a disposal depth of at least 5 m (16 ft) (measured from the land surface) for GTCC waste and implementation of an intruder barrier that prevents contact with the waste for at least 500 years. For a specialized land disposal facility, the licensee must demonstrate that the engineered intruder barriers, as part of a multi-barrier system including natural features, will prevent an intruder from disrupting or contacting waste during any part of the compliance period in which the waste remains a radiological hazard. For both applications, an applicant must demonstrate that the intruder barrier will be designed to deter domestic water well drilling or deeper excavations.

Attachment G-2 describes four commercial near-surface LLW disposal facilities in the United States and identifies whether and what type of human intrusion barriers have been used at these facilities. Because experience constructing human intrusion barriers for shallow land disposal of LLW is limited, Section G.5.1 includes the types of human intrusion barriers, and specifically the types of domestic water well intruder barriers, discussed in the radioactive waste disposal literature. Hypothetical intruder barriers that were more frequently discussed are described in more detail. Section G.5.2 briefly discusses common domestic water well drilling methods used in the United States. In Section G.5.3, information on the design of domestic water wells is provided to help licensees and reviewers determine what types of domestic water well intruder barriers should be considered for a specific disposal site or region of the United States. In addition, guidance is presented on conducting field tests and engineering analyses to provide assurance that the selected intruder barrier will perform as expected. Section G.5.4 reminds licensees and reviewers that they should consider the added effects of the domestic water well intruder barrier on the geotechnical stability of the disposal facility (Appendix E), including impacts on the capacity of the engineered surface cover to divert water.

Finally, Section G.5.5 provides additional guidance on engineered intrusion barriers relevant to specialized land disposal facilities.

The discussion in this section is based, in part, on the waste stream inventory provided in the DOE's GTCC FEIS. The guidance is appropriate for any waste streams containing concentrations and quantities of radioactive material needing human intrusion protection; however, the NRC staff recognize that specific intruder barrier designs will depend on the longevity of the hazard as well as the quantity of material to be disposed.

Licensees should conduct drilling tests or technical analyses to support their choice of intruder barrier for a specific site and should not limit their assessment to the six intruder barriers or to the four deterrent effects identified in this guidance. These examples, which are more succinctly listed in Table G-11, are presented to assist licensees develop their barrier design, deterrence analysis, and drilling tests. The inferred deterrent capacity of a specific barrier to a specific drilling method in Table G-11 is based on physical principles; however, in some cases, the inferred deterrent capacity is not supported by actual test results or detailed engineering analysis. The NRC staff has recently tested some barriers and drilling methods and published the results (Wittmeyer, 2026). However, the NRC results do not represent an endorsement or approval of a specific approach. Licensees are responsible for presenting and supporting the results of independent tests and analyses they conduct to guide the final design of domestic water well intruder barriers.

G.5.1 Potential Barriers that May Deter Domestic Water Well Drilling

Potential barriers to deter drilling have been investigated; however, that research is ongoing and not yet definitive. Licensees may need to complete additional research to justify that the intruder barriers employed in their design will be capable of meeting the requirements. Several technical reports on the disposal of radioactive waste describe physical barriers that may be capable of deterring water well drillers from inadvertently drilling into LLW disposal facilities. Most of these reports support performance assessments for (1) the mined geologic repository for TRU at the Waste Isolation Pilot Plant (WIPP) site near Carlsbad, New Mexico, (2) the near-surface disposal facilities for DOE LLW in Areas 3 and 5 at the Nevada National Security Site, which was formerly known as the Nevada Test Site, and (3) LLW disposal at six DOE-sponsored national laboratories.

G.5.1.1 Examples of Potential Intruder Barriers

The following are examples of potential intruder barriers that may deter domestic water well drilling:

(1) a 1 m (3.3 ft) to greater than 10 m (32.8 ft) thick layer of used rubber tires (Tolan, 1993, page 39; EPA, 1996, page 12-67; Black et al., 2001, pages 74 and D-70; National Security Technologies, 2007, pages 2, 5, and 7)

(2) a 10 m (32.8 ft) thick layer of galvanized baling wire or fencing (Tolan, 1993, pages 39–40; EPA, 1996, page 12-67; Black et al., 2001, pages 74 and D-70)

(3) a steeply angled flat shield with a very hard surface that causes deflection of the drill string (Black et al., 2001, pages 74 and D-70; von Winterfeldt, 1994, pdf pages 25–26; IAEA, 2011)

(4) a thick layer of large (boulder-sized), very hard (e.g., basalt), spherical-shaped rocks, angular riprap, and coarse gravel that will deflect the drill string (Tolan, 1993, pages 28–30; Wood et al., 1994, page 10)

(5) a layer of plate steel, approximately 2.5 centimeters (cm) (1 inch (in.)) (National Security Technologies, 2007, pages 4, 5, and 7)

(6) a 0.5 m (1.6 ft) to 1.5 m (5 ft) thick layer of heavily reinforced concrete (Black et al., 2001, pages 74 and D-70; National Security Technologies, 2007, pages 4 and 7)

G.5.1.2 Layer of Rubber Tires

Tolan (1993) postulates that drillers will encounter problems when drilling into a thick deposit of used rubber tires from (1) a loss of drilling fluid circulation, (2) the inability of a rotary bit to cut through the rubber, and (3) the components of the rubber tires wrapping around the drill bit and drill string, making it difficult to remove from the borehole. Tolan (1993) explains that the pliability of the rubber makes it difficult for a small, truck-mounted rotary rig to cut through this barrier. The U.S. Environmental Protection Agency (EPA) (EPA, 1996) cites but does not provide a reference to a report by "Ptyalin" as the original source for the information supporting Tolan's statement that "…operators have encountered problems in drilling into old landfills containing rubber tires (at least 10 meters thick), layers of steel fencing, and baling wire." Black et al. (2001) also discuss the ability of a layer of rubber tires to impede drilling progress, but they cite the discussions of their subject matter experts (SMEs) for the expert elicitation and input from DOE/Nevada as the sources for their 9.1 m (30 ft) thick rubber tire barrier concept. National Security Technologies (2007) cites a 1996 personal communication with Pitcher Drilling for the 1 m (3.3 ft) thick rubber tire intrusion barrier identified in its report. National Security Technologies (2007) suggests that the effectiveness of the rubber tire layer may be reduced once the rubber degrades, adding that it is difficult to determine the longevity of the rubber tire barrier. Finally, Swift (2013) seems to identify Tolan (1993) as the source of his statement that "…thick deposits of pliable rubber tires in landfills have been shown to pose a significant barrier to shallow drilling operations."

Adams and Wittmeyer (2020) interviewed water well drillers who serve the four counties surrounding the DOE Idaho National Laboratory (INL) and drillers who serve the three counties surrounding the DOE Savannah River Site (SRS) near Aiken, South Carolina, about the drilling methods they use and whether they have encountered reinforced concrete, steel or other hard metallic objects, or any other unexpected material while drilling. Drillers working in the vicinity of INL typically use air rotary rigs with roller-cone bits for overburden and down-the-hole (DTH) hammers for the basalt bedrock. Drillers working in the vicinity of the SRS commonly use direct mud rotary rigs with drag bits for the overburden and roller-cone bits for the limestone and shale bedrock. One SRS-area driller reported that he had once drilled into an old landfill that the landowner did not know about. He drilled through old tires and steel cables. When the driller removed the drill string from the borehole, he could see cable wrapped around the drill pipe. The driller did not report what, if any, effect the old tires had on drilling. He subsequently abandoned the borehole and filled it with concrete.[1] In contrast, Wittmeyer (2026) found that air and mud rotary drilling both penetrated a barrier made of four layers of tires backfilled with a low-strength flowable material.

[1] July 2, 2019, telephone interview with M. Swearingen of AAA Well Drilling, Lexington, South Carolina.

G.5.1.3 Layer of Baling Wire or Fencing

Tolan (1993) states that drillers will encounter problems when drilling into a thick deposit of baling wire or fencing from (1) a loss of drilling fluid circulation, (2) the inability of a rotary bit to cut through the baling wire or fencing, and (3) the wire wrapping around the drill bit and drill string, making it difficult to remove the drill string from the borehole. As for the rubber tire barrier, EPA (1996) refers to a report by "Ptyalin" as the source for the concept of using a thick deposit of baling wire as an intruder barrier. Black et al. (2001) discuss the ability of a 9.1 m (30 ft) thick layer of galvanized baling wire to impede drilling progress, but they again cite discussions by their SMEs for the expert elicitation and input from DOE/Nevada as the sources for this concept. Black et al. (2001) state "[t]he tires and bailing (sic) wire would not physically deter drilling but may cause a driller to question why those foreign materials are there, and hence move drilling operations to another location."

National Security Technologies (2007) states "[a] thin layer of metal, such as the metal wire recommended in Tolan (1993), might temporarily impede a rotary drill, but would prove to be only a minor impediment to more sophisticated drilling technologies." National Security Technologies (2007) does not identify the more sophisticated drilling technologies that could penetrate the wire barrier; however, either a dual air rotary rig with a DTH hammer and a casing shoe studded with carbide or polycrystalline diamond compact (PDC), or a sonic drill rig with a core-barrel bit or a studded casing shoe (see Section G.5.2 for details on drilling methods), may be capable of penetrating a barrier composed of baling wire. Wittmeyer (2026) found that a 2.0 m (6.7 ft) thick barrier made of rolls of chain link fencing deterred a driller using an air rotary drill but did not deter a driller using mud rotary drilling.

EPA (1991) states "[s]ampling landfill contents can also be difficult, as garbage bags, baling wire, etc., cling to the augers," which adds credence to the third drilling problem (wire wrapping around drill string) identified by Tolan (1993). Anecdotal evidence of drillers encountering metal wire or cable in old landfills is described in Section G.5.1.2.

G.5.1.4 Steeply Angled Shield with a Hard Surface

Black et al. (2001) state that the "SMEs suggested placing an angled shield that would deflect a drill bit over the waste area would be almost certain to ensure that no wells are drilled into the waste area." To deflect a drill bit, Black et al. (2001) state that the shield would need to be made of a hard metal, such as titanium, which for a large area, such as a repository or near-surface disposal facility, could be cost prohibitive. The DOE's SME panel did not consider this barrier any further in their study since the barrier at the near-surface disposal facilities in Areas 3 and 5 at the Nevada Test Site would cover at least 2 acres and would be too expensive to install. Although von Winterfeldt (1994) does not explicitly identify a steeply angled shield with a hard surface as a protective barrier, he suggests that consideration be given to designing "plates" that resist or delay traditional drilling methods. Steeply angled hard surfaces may not be cost prohibitive if used for small areas, such as for small near-surface facilities (small volumes of waste) or the area around an individual borehole in a specialized land disposal facility (IAEA, 2021).

G.5.1.5 Layer Composed of Boulders, Riprap, and Coarse Gravel

Tolan (1993) discusses using boulders, riprap, and coarse gravel layers to construct a 5 m (16 ft) high barrier mound on top of a disposal facility to reduce the risk of inadvertent human

intrusion by drilling. Mounds are assumed to reduce the likelihood of drilling into the disposal facility primarily because it is easier to drill at the base of the mound than to maneuver a drill rig up the steep sides to the top of the 5 m (16 ft) high mound. However, even if a driller manages to drive a rig to the top of the mound, Tolan (1993) states that "…the riprap/gravel layers would be expected to present varying degrees of drilling problems, depending on the type of drilling method and the technology used, for the intruder."

Although Tolan (1993) does not describe the types of drilling problems that the 5 m (16 ft) mound of boulders, riprap, and coarse gravel would cause, several other sources on well drilling methods describe how boulders affect drilling performance and explain what to do when boulders are encountered. Ruda and Bosscher (2005, Section 4.9) indicate that borehole deflection can be a problem when the drill string or driven casing encounters a boulder. They describe procedures for correcting boulder-induced borehole deflection encountered using cable tool rigs, direct rotary rigs, reverse rotary rigs, and dual air rotary rigs with casing drivers. Driscoll (1986) states that "[i]n many formations boulders or large cobbles can slow or even stop drilling progress regardless of the drilling method being used." Driscoll (1986) provides technical advice on how to crush, blast, fish, or cement in place a single boulder; however, when drilling into a unit that is primarily composed of cobbles and boulders or other highly stratified alluvial deposits, he states that a dual air rotary rig with a DTH hammer bit and a casing driver may be required. With this drilling method, the casing is typically advanced a few inches behind or nearly flush with the drill bit to prevent borehole collapse. A DTH hammer is run into the casing on the drill string, where it crushes the cobbles and boulders, leaving cuttings that are easily lifted in the annulus between the casing and the drill string by an air-water or air-foam drilling fluid.

A second potential problem when drilling into a homogeneous layer of similar-sized boulders is loss of drilling fluid, which may cause a drill string to become stuck. Relatively homogenous boulder deposits have been formed by extreme flooding from glacier ice dam failures and large landslide-induced tsunamis, as well as boulder fields and scree slopes that result from in-place weathering; however, no information could be found about water wells being drilled into these types of deposits. An engineered, carefully emplaced homogeneous layer of uniform-sized boulders (250 millimeters (mm) (10 in.) or greater in diameter) will have large pores and be highly conductive to fluid flow. The high fluid conductivity of a deposit of uniform-sized boulders will result in a rapid loss of drilling fluid, which may force the crew to suspend drilling until the weight and viscosity of the mud can be increased or additional drilling fluid can be brought to the site. Dual air rotary drilling rigs should be relatively unaffected by lost circulation while drilling a boulder deposit. Either the casing shoe will penetrate a boulder and prevent compressed air from leaking into the formation, or the drilling rate can be reduced until the air compressor is able to overcome the rate of air loss. Consistent with that expectation, Wittmeyer (2026) found that a 1.8 m (5.8 ft) thick layer of boulders did not deter a driller using an air rotary drill but did deter a driller using mud rotary drilling.

G.5.1.6 Layer of Plate Steel

National Security Technologies (2007) identifies "[a] layer of plate steel, approximately 2.5 cm thick" as a potentially effective and economically viable underground barrier. However, National Security Technologies (2007) also notes that this barrier could be "penetrated by an aggressive driller" and does not include it as one of the three barriers recommended for use at the Nevada Nuclear Security Site.

Drillers operating near SRS and INL were interviewed by Adams and Wittmeyer (2020) to determine the types of drilling methods they use and whether they have encountered rebar or other steel objects while drilling. One INL-area driller reported accidentally drilling through existing 6.4 mm (0.25 in) thick steel well casing using a roller-cone bit, noting that the bit was damaged by the encounter. A second INL area driller described drilling into an old foundation with a roller-cone bit and hitting a piece of small-diameter rebar. Two SRS-area well drillers reported that they have drilled into reinforced concrete. One of these drillers fabricated his own "coring bit" to cut through reinforced concrete. The second driller has access to an air rotary rig with a DTH hammer that can drill through reinforced concrete and steel. If the second driller's air rotary rig is a dual air rotary rig, then it is possible that the casing shoe is "sawing" through the steel since the percussive action of a DTH hammer is unlikely to induce brittle fracture in steel.

A third drilling technology that has been demonstrated to cut through reinforcing and thick steel is a sonic drill rig with a studded coring bit or a studded casing shoe. Slides from a presentation[2] by sonic drill rig and equipment manufacturer Boart Longyear™ includes photographs of a reinforced concrete core sample as well as what appear to be core samples of steel rail.

G.5.1.7 Reinforced Concrete

SMEs consulted by Black et al. (2001) suggested a human intrusion barrier consisting of a 1.5 m (5 ft) thick layer of reinforced concrete with 2.5 cm (1 in) rebar set at 15 cm (6 in) intervals (lateral spacing) to ensure that a typical domestic water well drill bit cannot pass between the bars. To have a finite probability of not contacting rebar and potentially passing through the concrete portion of the barrier, a drill bit would have to have a diameter less than 13 cm (5 in). A simple geometric model can be constructed to show that a 10 cm (4 in) drill bit has a probability of 0.03 of not contacting 2.5 cm (1 in) two-way rebar mats set at 15 cm (6 in) intervals. Mat sets with thicker rebar increase the probability of drill contact but also increase cost. National Security Technologies (2007) states that Pitcher Drilling indicated in a personal communication from 1996 "…that steel-reinforced concrete would provide an effective barrier to rotary-wash drilling." However, because the cost of emplacing reinforced concrete is high and the barrier's longevity is uncertain, National Security Technologies (2007) states that a reinforced concrete barrier is not appropriate for LLW disposal at Nevada Test Site Areas 3 and 5.

The second paragraph of Section G.5.1.6 describes actual experiences of water well drillers encountering and being able to drill through reinforced concrete using air rotary rigs equipped with DTH hammers and mud rotary rigs equipped with drag and roller cone bits. The third

[2] PDF downloaded from https://www.boartlongyear.com/insite/a-sound-solution-for-frustrating-formations/ accessed 1/27/2020. See page 6.

paragraph of Section G.5.1.6 describes the ability of a properly equipped sonic drill rig to drill through reinforced concrete, although it is uncertain how many drillers use sonic rigs for drilling water wells.

The expert elicitation conducted by Black et al. (2001) estimates that the probability is 0.50 that a 1.5 m (5 ft) layer of reinforced concrete placed 6.1 to 9.1 m (20 to 30 ft) underground will deter the completion of a water well. The stated rationale for this deterrence probability is that "[t]he reinforced concrete was expected to ruin a drill bit every 6 in (15 cm), thus leading to a level of frustration over many feet that may cause a driller to be deterred from continuing at that location." The wording of the rationale indicates that the 1.5 m (5 ft) thick concrete layer has a horizonal layer of rebar placed every 15 cm (6 in) vertically. This amount of reinforcing probably exceeds that needed to resist tensional stresses in the concrete caused by a variation in the overburden load or in the location of barrier support, suggesting that it is primarily intended to be an obstacle to drilling. Consistent with those expectations, Wittmeyer (2026) found that a 1.5 m (5.0 ft) barrier of concrete reinforced with 1.6 cm (5/8 inch) diameter rebar with a 15-cm (6.0-inch) spacing deterred drillers using both air rotary and mud rotary drilling.

G.5.2 Overview of Domestic Water Well Drilling Methods

A water well drill rig must be able to create an open hole from the ground surface to water-bearing formations capable of meeting the long-term water supply requirements of the user. The water well borehole must have a diameter large enough to accommodate conductor or surface casing, well casing, and well screens, as well as grout placed for well-head protection, packers and concrete used to isolate formations, gravel packs surrounding well screens or slotted casing, and a submersible pump or the bowls of a shaft-driven pump. The mechanical systems of the rig must be able to penetrate unconsolidated and consolidated geologic formations to reach the desired water-bearing unit or aquifer, remove geologic material as the borehole advances, and raise and lower drilling tools and well components, such as new sections of drill pipe and well casing.

There are four methods commonly used in the water well industry to penetrate geologic materials and advance the borehole, as follows:

(1) rotary drilling, in which a rotating table fixed to the drilling platform (table drive) or a hydraulic motor that can move up and down the mast (top head drive) rotates the drill string and drill bit

(2) percussion drilling, in which a low-frequency (0.33 to 0.66 hertz (Hz) (20 to 40 beats per minute (bpm))) reciprocating cable tool string, or a high frequency (13 to 38 Hz (800 to 2,300 bpm)) pneumatic hammer, repeatedly lifts and drops/drives a drill bit to fracture and crush geologic material at the bottom of the borehole

(3) sonic drilling, in which a mechanical oscillator excites high-frequency (50 to 160 Hz [3,000 to 9,600 bpm]) vertical sympathetic harmonic resonance in the drill string that either induces liquefaction (reduces effective stress to zero) in unconsolidated sediments or delivers high-frequency, fracture-inducing blows to the bottom of a borehole in cohesive soil or consolidated material, or rock

(4) combined rotary and percussion drilling, in which a DTH or top drive pneumatic hammer applies percussive action at the bottom of the borehole to fracture the material while the drill string is rotated to both advance the bit teeth into fresh rock and grind rock chips

There are three primary methods residential water well drillers use for removing drill cuttings from the borehole, as follows:

(1) flushing with fluids, including air, air with a water spray, air with water and a foaming agent, pure water, and water with various additives, such as bentonite clay, barite, and polymers, that change the resulting mud's density and viscosity or modify the mud's rheology to make it shear resistant (thixotropic)

(2) bailers, which are long tubes with gravity-actuated, one-way valves at their down-hole ends that entrain and hoist water and cuttings from the bottom of the borehole

(3) coring bits and core barrels where special drill bits cut ring-shaped holes and then break off and catch the column of geologic material either inside the hollow drill rod or in a separate thin-walled core barrel run-in on wire line through the drill string

Not all combinations of borehole drilling methods and cuttings removal methods are used in modern water well drilling. Specific combinations are preferred for drilling certain rock types or regularly encountered sequences of rock formations (e.g., a sandstone facies, underlain by a shale facies, underlain by a marine limestone facies). For example, a rotary rig with a DTH hammer will typically be used for competent consolidated formations where the likelihood of borehole formation collapse is minimal. Fine cuttings created by the rotating DTH hammer are removed using air with water spray, and unlike mud, the air-water mixture will not clog the water-bearing fracture zones. For unconsolidated alluvial deposits and soft-to-hard sedimentary rocks, a rotary rig with drilling mud will be used to lift cuttings, lubricate and cool the drill bit, and prevent borehole collapse. If the likelihood of borehole collapse is high and water is readily available at the site, a driller may use a rotary rig with reverse mud circulation where the drilling fluid is injected into the annulus between the drill string and the borehole and returned up through the center of the hollow drill string to a pit large enough to provide sufficient drilling fluid to accommodate fluid losses to the formation.

Cable tool rigs, also known as pounder, churn, or spudder rigs, almost always use bailers to remove cuttings from the borehole. The cable tool rig operator usually pours water into the borehole to make the cuttings flow more rapidly into the bailer. Sonic rigs always retrieve drill cuttings by removing soil and rock cores. Adams and Wittmeyer (2020) provide more detailed information on sonic drilling. Table G-4 lists most of the typical combinations of drilling and cuttings removal methods used in modern residential water well drilling. The second column in Table G-4 identifies the general types of bits that are used to drill residential water wells for the combination of drilling method and cuttings removal method and the types of geologic formations being drilled.

Table G-4 Drilling Methods and Drill Bits Used for Constructing Domestic Water Wells

Drilling Method and Cuttings Removal Method	Type of Formation and Typical Drill Bits	Comments
Rotary Drilling Methods		
Rotary with Water, Direct and Reverse Circulation	Consolidated and unconsolidated alluvial deposits Drag bits, milled tooth roller-cone bits for "soft" formations and button bit roller cone bits for harder formations	Use if water is readily available and lower section of well will be screened and the cost of well development to remove mud that has invaded the formation is expensive.
Rotary with Mud, Direct Circulation	Consolidated and unconsolidated deposits and soft to medium hard sedimentary rock Drag bits or milled tooth roller-cone bits for "soft" formations, button bit roller cone bits for harder formations, and PDC bits for medium hard formations	Use where water is not readily available and likelihood of borehole formation collapse is minimal to moderate
Rotary with Mud, Reverse Circulation	Unconsolidated alluvial deposits and soft to medium sedimentary rock Drag bits or milled tooth roller-cone bits for "soft" formations, button bit roller cone bits for harder formations, and PDC bits for medium hard formations	Water is readily available and likelihood of borehole formation collapse is significant. This method requires a large mud pit and is usually reserved for large municipal water wells.
Percussion Drilling Methods		
Cable Tool Rig with Sand Bailer	All formations Star bits and Hubbard bits of various lengths and weights	Cable tool rigs cost significantly less than modern truck-mounted rotary rigs with or without pneumatic hammers and also less than sonic rigs. Rate of penetration can be very slow, but crews are smaller and require less training than rotary rig crews.
DTH or Top Drive Hammer, Rotary with Air, Air with Water Spray, Air-Water-Foam, Direct Circulation	Harder rocks such as granite, basalt, and quartzite Tungsten-carbide buttons set in cobalt binder brazed to the face of the hammer	Water is not readily available. Hammer drilling is typically much faster than pure rotary, even with a rotary bit that is well matched to the lithology being drilled
Sonic Drilling Methods		
Sonic Drilling	All formations, but often used when complete cores of soft clay, silt, and sand formations are needed. Capable of coring steel reinforced concrete and thick steel Diamond set shoe or diamond impregnated coring bit	Drilling rates with sonic rigs can be very rapid. Drilling depth is limited by the increased energy required to vibrate long drill strings at their natural harmonic frequency. Typically, it is used to obtain full core samples for geotechnical and near-surface mineralogical analyses.
Combined Rotary and Percussion Drilling Methods		
DTH Hammer, Rotary with Air, Air with Water Spray, Air-Water-Foam. Direct Circulation	Hard rocks such as granite, basalt, and quartzite Tungsten-carbide buttons set in cobalt binder brazed to the face of the hammer	Water is not readily available. DTH hammer with rotary drilling is typically faster than straight DTH

Drilling Method and Cuttings Removal Method	Type of Formation and Typical Drill Bits	Comments
DTH Hammer, Dual Rotary with Air, Air with Water Spray, Air-Water-Foam. Reverse Circulation Through Annulus of Simultaneously Driven Casing and Drill String	Formations that easily collapse and trap the drill string or could result in lost circulation of drilling fluid. Caving sands and alluvial fans with large boulders. Tungsten-carbide buttons set in cobalt brazed to the face of the hammer. Shoe on driven casing fitted with carbide studs.	Dual rotary rigs typically use a top drive system to rotate the drill string and a table drive to rotate and drive the casing, which is affixed with a carbide-studded casing shoe to drill and ream the borehole wall.

G.5.3 Selecting Intruder Barriers and Evaluating their Penetration Resistance

Figure G-2 shows the general steps that a licensee can follow to assist in selecting an appropriate site-specific intruder barrier. Detailed information on selecting intruder barriers is provided in the sections that follow.

Developing an understanding of the general design of the domestic water well that could be constructed near a disposal facility should allow the licensee to determine the approximate size of the drill bit being used when the intruder barrier is encountered. Then, a survey of local domestic water well drilling practices or information about the properties of the subsurface formations that must be drilled will help the licensee to bound the type of drilling and cuttings removal methods used to drill a domestic water well in the vicinity of the site. Knowing (1) the diameter of the drill bit when it contacts the barrier, (2) the type of drill bit expected to be used, and (3) the drilling and cuttings removal methods expected to be used in the vicinity of the site should allow the licensee to select one or more intruder barriers for further evaluation.

The process of designing a domestic water well requires gathering and analyzing information and measured data about (1) the hydrogeology of the proposed disposal site, including a stratigraphic column of the geology showing aquifers, confining units, water surface or piezometric elevations, and the target aquifer's hydraulic properties, (2) a survey of domestic water well drilling practices used in the vicinity of the proposed disposal site, and (3) estimates of the volume of water used by homes in the vicinity of the disposal site, including typical values for the acreage of irrigated turf and the number of trees used for landscaping and windbreaks. With this, the licensee may determine the design and discharge capacity of a water well that a typical residence in the vicinity of the disposal site would need. Alternatively, the licensee may consult one or more local drillers to obtain design drawings and construction specifications for a typical domestic water well.

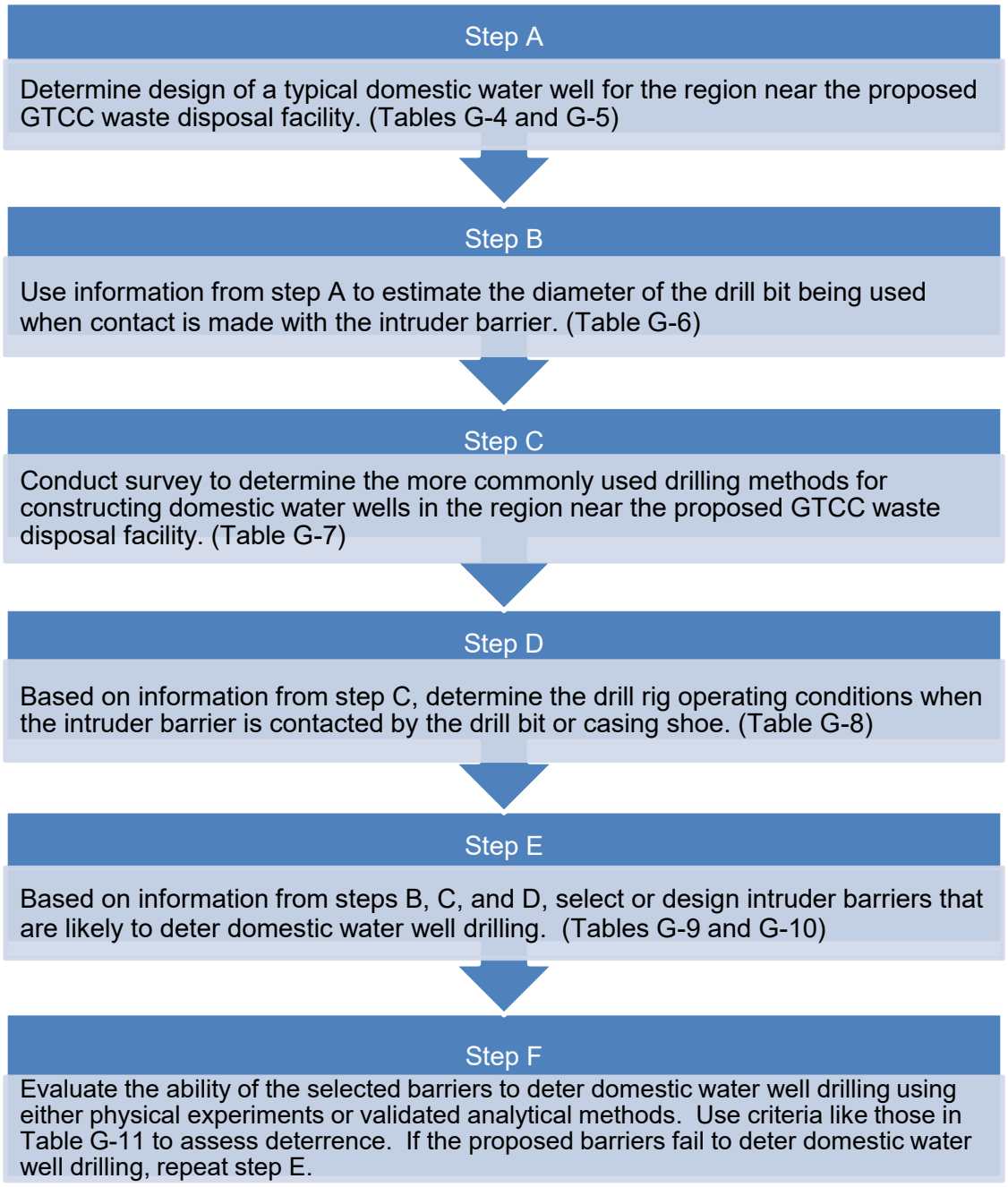


Figure G-2 General Steps in Selecting an Appropriate Site-Specific Intruder Barrier

G.5.3.1 Water Well Design and Construction

Glotfelty (2019) recommends following an "inside-out" approach to well design. For determining casing and borehole dimensions for a residential water well, Glotfelty (2019) suggests a design sequence that starts at the centerline of the well and moves radially outward:

(1) Size the pump and related equipment.
(2) Determine casing and well screen diameters.
(3) Determine needs for tubing strings in the annulus.
(4) Determine borehole size (diameter of the drill bit).
(5) Determine surface casing size.
(6) Determine surface borehole diameter.

The following example is provided as guidance to a licensee that has elected to determine the design and capacity of a water well for a typical residence in the vicinity of the disposal site. This example is based on one of the potential GTCC waste near-surface disposal sites identified in the DOE GTCC FEIS. Table G-5 lists hydrogeologic attributes of the Hanford Formation and the Ringold Unit E, which would be the likely targets of a driller inadvertently attempting to construct a residential water well in the vicinity of a hypothetical near-surface GTCC waste disposal facility at the Hanford Reservation Integrated Disposal Facility (IDF). If the driller follows the six-step design sequence suggested by Glotfelty (2019), the well design should be similar to that shown in Table G-6. The bases listed in the third column of Table G-6 are from standard references on domestic and commercial water well drilling (e.g., USBR,1985; Driscoll, 1986; Ruda and Bosscher, 2005). The licensee should provide sufficient detail about domestic water well design to support its determination of the types of drilling and cuttings removal methods, drill bit types, and drill bit diameters that may contact the intruder barrier.

Table G-5 Physical Attributes of the Hanford Formation and Ringold Unit E (all data are from DOE (2016a,b))

Disposal Site	Aquifer and Composition	Aquifer Conditions	Storage Property	Depth to Water, m (ft)	Hydraulic Conductivity, m/d (ft/d)	Aquifer Thickness, m (ft)
Hanford Reservation IDF	Hanford Formation (sands and gravels < 65 m thick) and Ringold Unit E (gravel)	Unconfined	Specific yield S_y=0.25 (effective porosity)	88 (290)	35 (115)	45 (148)

Table G-6 Inside-Out Design of a Residential Water Well that is Constructed at a Near-Surface Disposal Facility for GTCC Waste Located at Hanford

Well Component	Specifications	Basis
Pump	57 L/min [15 gpm] under a Total Dynamic Head (TDH) = 135 m [443 ft]. Pump efficiency tables for Goulds® GS Series pumps suggest most efficient model 18GS30412C, a 10.2-cm [4-in] diameter, 3 HP, 19 stage, 107.7-cm [42.4-in] long (pump + motor) submersible. Motor requires 60 Hz, 230 VAC, single phase with 3 wires.	57 L/min [15 gpm] is based on minimum capacity pump needed to meet the 371 L [98 gal] seven-minute peak water demand for a 2-2½ bathroom home. TDH = Static Head + Drawdown + Head Loss in Piping + Pressure Head Required at House = 88 m + 0.2 m + 4.5 m + 42 m = 135 m. Static Head = depth to static water level. Drawdown at 57 L/min is calculated to be 57 L/min ÷ 500 L/min/m = 0.11 m ÷ 0.7 = 0.16 m is based on estimated specific capacity of 500 L/min/m [40 gal/min/ft[1]] and well efficiency of 70%. Head Loss for 120 m [395 ft] of 3.175-cm ID [1.25-in] PVC pipe is 4.5 m [15 ft]. Pressure Head Required at House corresponds to supply pressure of 400kPa [60 psi].
Casing and Well String	10 m [33 ft] of 15.2-cm [6-in] ID wire-wrap well screen and 98 m [322 ft] of 15.2-cm [6-in] ID steel well casing with wall thickness 7.1 mm [0.28 in]. Steel well casing OD is 16.8 cm [6.625 in].	15.2-cm [6-in] ID well screen required for 2.54-cm [1-in] annular clearance for 10.2-cm [4-in] diameter submersible pump. Borehole will be drilled 20 m [66 ft] into the saturated zone for a total depth of 108 m [355 ft]. Well screen will be set at the bottom of the hole 108 m [355 ft] = 10 m [33 ft] of well screen + 98 m [322 ft] of casing. Pump will be set 1 m [3.3 ft] above top of screen.
Tubing Strings	3.8-cm [1.5-in] OD tremie pipe for installing gravel filter pack, bentonite seal pack, and concrete surface seal.	Standard-sized tremie. Will require minimum 5.1-cm [2-in] annulus around casing string.
Borehole Size	27 cm [10.625 in] diameter for screened section with gravel filter pack	16.83 cm [6.625] OD for casing + 2×5.1 cm [2 in] annulus for gravel pack and tremie = 27 cm [10.625 in] diameter borehole
Surface Casing Size	3-m [10-ft] of 35.6 cm [14 in] temporary surface casing to prevent collapse of upper borehole. Note that driller should encounter an engineered cover consisting of several layers of compacted soil, clay, and coarse gravel and may not need surface casing.	Well logs for the IDF area indicate that original surficial materials consisted of unconsolidated wind-blown sand and alluvial sand and gravel. This material should be removed during disposal cell construction and replaced by an engineered cover at closure.
Surface Water Well Borehole Diameter	35.6 cm [14 in] to 3 m [10 ft] bgs. 27 cm [10.625 in] from 3 m [10 ft] bgs to 108 m [355 ft] bgs.	

[1] From Figure 2-4 "Comparison of Transmissivity, Specific Capacity, and Well Potential" USBR (1985) for T = 35 m/d × 20 m = 700 m^2/d [7,600 ft^2/d]

G.5.3.2 Drilling Method, Drill Bit Diameter, Drill Bit Type, and Rig Operating Conditions

The licensee should demonstrate the adequacy of an intruder barrier using site-specific information on drill rig types, drill bit sizes, drill bit types, and rig operating conditions at the time the intruder barrier is encountered. Table G-7 gives estimates of the diameter of the drill bit being used when the drill string contacts an intruder barrier for the example location at the Hanford IDF. Table G-8 lists the drilling method, drill bit type, and drill bit diameter likely to be used by a domestic water well driller at the Hanford IDF. This table was constructed using aquifer information contained in Table G-5, the likely diameter of the drill bit being used when the drill string encounters the intruder barrier listed in Table G-7, and drilling methods described in Table G-4. Because the formations at the Hanford IDF consist of unconsolidated sands and gravels, it was assumed that direct rotary with foam, water, or mud circulation, would be a common drilling method. Although, a cable tool rig or a sonic rig could be used in unconsolidated formations, these drilling practices are less common than direct rotary. However, the licensee should consider conducting a survey of local water well drilling practices to establish the type of drilling method and bit type the barrier should be required to deter.

Rig operating conditions at the time the drill bit contacts the intruder barrier should be considered when selecting and designing the barrier. Information on the pullback capacity of the draw works, rotational speed of the bit, axial thrust (weight on bit (WOB)), and torque applied to the drill bit can be obtained from drill rig manufacturer brochures and information from standard references on water well drilling. Table G-9 identifies a range of operating conditions for an Epiroc Atlas TH60 direct rotary rig that could be used by a water well driller at the Hanford IDF. As an example, if the licensee were to select an intruder barrier constructed of heavily reinforced concrete, the bit load data in the third column may be useful in selecting the unconfined compressive strength (UCS) of the concrete mix, and the torque on the drill string listed in the fifth column could be useful in selecting the thickness of rebar to accommodate stresses imparted by the bit's teeth. The licensee can use practical engineering approaches to design the intruder barrier, and it should present and support its technical basis when using analytical methods (e.g., finite element, finite difference, distinct element methods (DEM)) to demonstrate the effectiveness of the barrier.

Table G-7 Diameter of Drill Bit Being Used When the Intruder Barrier is First Contacted at the Hanford IDF

Disposal Site	Depth and Diameter of Permanent Well Casing Seal	Diameter of Drill Bit at 5 m [16.4 ft] BGS, the Depth of the Intruder Barrier
Hanford Reservation Integrated Disposal Facility (IDF), Hanford, Washington	Minimum depth of 5.5 m [18 ft], diameter must be 10.2 cm [4 in] greater than diameter of permanent casing.[1]	27 cm [10.625 in]

[1] Washington Administrative Code (WAC)-173-160-231 Water Well Requirements: Standards for Surface Seals. (https://app.leg.wa.gov/wac/default.aspx?cite=173-160-231&pdf=true accessed 1/21/2021)

Table G-8 Drilling Method and Drill Bit Type and Diameter Likely to be Used When the Inadvertent Driller First Contacts the Intruder Barrier at the Hanford IDF

Disposal Site	Drilling Method	Drill Bit Type	Diameter of Drill Bit
Hanford Reservation Integrated Disposal Facility (IDF), Hanford, Washington	Direct rotary with foam, water, or mud	Milled tooth and button bit roller-cone bits	27 cm [10.625 in]

Table G-9 Estimated Drill Rig Operating Conditions When the Intruder Barrier is First Contacted at Hanford IDF[1]

Disposal Site	Diameter of Drill Bit at 5 m [16 ft] BGS, the Depth of the Intruder Barrier	Bit Loads	Rotation Rate of Drill String	Torque
Hanford Reservation Integrated Disposal Facility (IDF), Hanford, Washington	27 cm [10.625 in]	Milled tooth tricone: for material with UCS < 35 MPa [< 5,000 psi]; WOB 4,800 to 19,300 kgf [10,600 to 42,500 lb] TCI button tricone: for material with UCS < 83 MPa [< 10,000 psi]; WOB 4,800 to 19,300 kgf [10,600 to 42,500 lb]	Milled tooth bit: 50 to 100 rpm, higher WOB => lower rpm TCI button bit: 50 to 150 rpm, higher WOB => lower rpm	Epiroc Atlas TH60 direct air and mud rotary drill rig. 5.42 to 7.45 kNm [4,000 to 5,500 ft-lb]

[1] Values are best estimates derived from drill rig manufacturer technical brochures.

G.5.3.3 Barrier Design Versus Drilling Technology

Table G-10 lists the six intruder barriers identified in Section G.5.1.1, describes how the barrier may deter drilling, identifies water well drilling methods that may be deterred by the barrier, and lists water well drilling methods that may be able to penetrate the barrier. As can be seen in Table G-10, based on the common drilling method identified for the Hanford Site in Table G-8 (i.e., direct rotary with foam, water, or mud circulation), all six intruder barriers may be potential candidates for a licensee to consider, depending on the type of drill bit used. Table notes provide limited justification for determining which drill rig or drill bit combinations may or may not penetrate a barrier. Column 2 of Table G-10 shows that these six intruder barriers are anticipated to present at least four deterrent effects: (1) very slow rate of penetration (Slow ROP), (2) excessive drill string deflection (Deflection), (3) excessive loss of drilling fluid (Fluid Loss), and (4) stuck drill string (Stuck String). Parenthetical notes in column 2 and details in the notes from Table G-10 show that different barriers may achieve the same deterrent effect by

different physical attributes and mechanisms. No reports have been identified that describe actual drilling tests of the six barriers. The barriers' deterrent mechanisms were inferred based on the physical properties of the barrier. Similarly, the ability of a barrier to deter or not deter a specific type of drilling technology is inferred from the barriers' physical properties and the physics of drilling, including fluid circulation.

G.5.3.4 Demonstration of Barrier Deterrent Capacity by Field Testing

For field tests to produce applicable results, the licensee may consider constructing test barriers at or near the proposed GTCC waste disposal facility. Conducting experiments on location would allow experienced local drillers to select the appropriate drilling equipment for the local aquifers and overlying formations. To allow the experimental results to be compared, the licensee may consider providing the test drillers and their crews some basic information about the experiments, including, for example, that they are participating in an experiment to investigate whether physical barriers can be designed that will prevent a residential water well driller from penetrating an important subsurface facility. The licensee should consider asking the driller to conduct their work in accordance with their company standards for meeting the customer's requirements without risking injury to crew members or observers or losing money on the job due to excess time onsite or permitting excess wear and damage to their equipment. The licensee may ask drillers to drill through the barrier, without risk to the crew, and later analyze the results including the effort and expense to drill through the barrier or to the point where drilling had to stop. The review would determine at what point in the drilling process it would become unrealistic (e.g., too many broken drill bits) or uneconomical (e.g., too much time or money spent) for a drilling crew under contract to continue.

The licensee should keep in mind that a driller may be disinclined to continue if drill cuttings coming to the surface are composed of man-made material, which would include the intruder barriers listed in Table G-10, except for large rocks. This could cause the driller to relocate the drilling rig and restart drilling or, at least, to inquire about previous human activity at or near the drilling location.

G.5.3.4.1 Assessing the Success and Failure of Drilling Tests

Before conducting tests, the licensee should establish the range of experimental outcomes that would generally indicate whether a specific barrier is likely to deter residential water well drilling. Adams and Wittmeyer (2020) describe interviews with drillers who have constructed many hundreds of residential water wells in the vicinity of the SRS near Barnwell, South Carolina, and INL near Idaho Falls, Idaho. The interviews reveal that drillers have drilled into "impenetrable" materials such as reinforced concrete and 16 mm (0.625 in) thick steel casing shortly after opening a water well borehole, but have not, in all instances, immediately stopped drilling, tripped out (removed the entire drill string from the borehole), and started a new borehole at another location. Although Adams and Wittmeyer (2020) did not discover how long the drillers continued to drill through the reinforced concrete and steel well casing before reentering the formation, it would seem reasonable that a driller who encounters a geologic unit that cannot be penetrated despite drilling for a sustained period of time at a depth of just 5 m (16 ft) would either examine cuttings to determine whether the ROP can be increased with a more suitable bit or start over at another location. For a near-surface disposal facility, the intruder barrier would typically be placed within the top 5 m (16 ft) because 10 CFR 61.52(a)(3) requires waste to be buried at least 5 m (16 ft) below the land surface. In a specialized disposal facility, disposal depths would be at least 30 m (98 ft) and potentially much greater. However, an intrusion

barrier within 5 m (16 ft) of the surface could be an effective part of a multi-barrier system to deterring domestic water-well drilling. Although the performance of a near-surface barrier may degrade after the initial 500-year period required for near-surface disposal, a near-surface barrier with degraded performance could also provide drillers with an early indication that they are drilling into an engineered facility, depending on the barrier design. Guidance on engineered barriers to inadvertent intrusion at specialized disposal facilities is provided in Section G.5.5 and additional considerations for deterring intrusion for long timeframes is provided in Section G.7.

Table G-10 Summary of Hypothetical Domestic Water Well Intruder Barriers, Deterrent Effect of Barrier, Types of Water Well Drill Rigs and Drill Bits that may be Deterred by the Barrier, Types of Water Well Drill Rigs and Drill Bits that may Penetrate the Barrier

Intruder Barrier	Deterrent Effect	Deterred Drill Rigs and Bits	Undeterred Drill Rigs and Bits
Heavily Reinforced Concrete[1]	Very Slow Rate of Penetration (Frequent Drill Bit Replacement, Button Replacement, Tooth Dressing)	Mud-rotary (direct and reverse) with tungsten-carbide (TCI) edged drag bits, roller cone bits with TCI or milled teeth, fixed cutter bits with polycrystalline diamond compact (PDC) inserts[2] Air-rotary with TCI edged drag bits, roller cone bits with TCI or milled teeth, PDC bits, DTH hammer with diamond or TCI buttons [2,3]	Heavy cable tool rig Dual air-rotary with TCI or diamond impregnated/studded casing shoe and DTH hammer with diamond or TCI button bits[4] Sonic rig with TCI studded, PDC studded, diamond impregnated, and surface-set diamond casing shoes and core barrel bits
Large[5] Spherical Rocks of Equal Diameter	Excess Drill String Deflection Excess Loss of Circulation Loss of Drill String	Mud-rotary (direct and reverse) with TCI edged drag bits, roller cone bits with TCI inserts or milled teeth, PDC bits Air-rotary with TCI edged drag bits, roller cone bits with TCI inserts or milled teeth, PDC bits, DTH hammer with diamond or TCI buttons	Heavy cable tool rig Dual air-rotary with TCI or diamond impregnated/studded casing shoe and DTH hammer with diamond or TCI button bits Sonic rig with TCI studded, PDC studded, diamond impregnated, and surface-set diamond casing shoes and core barrel bits
Rubber Tires[6]	Excess Loss of Circulation Loss of Drill String Very Slow Rate of Penetration (Inability of Drill Bit to Shear or Shred Rubber)	Mud-rotary (direct and reverse) with roller cone bits with TCI inserts or milled teeth, PDC bits Air-rotary with roller cone bits, DTH hammer Cable tool rig	Mud-rotary with TCI edged drag bit, PDC bits[7] Air-rotary with TCI edged drag bit, PDC bits [7] Dual air-rotary with TCI or diamond impregnated/studded casing shoe and DTH hammer with diamond or TCI button bits Sonic rig with TCI or diamond studded casing shoes and core barrel bits

Intruder Barrier	Deterrent Effect	Deterred Drill Rigs and Bits	Undeterred Drill Rigs and Bits
Baling Wire or Fencing[8]	Excess Loss of Circulation Loss of Drill String Very Slow Rate of Penetration (Inability of Drill Bit to Shear or Tear Steel Wire)	Mud-rotary (direct and reverse) with roller cone bits Air-rotary with roller cone bits, DTH hammer	Mud-rotary with TCI edged drag bit, PDC bits Air-rotary with TCI edged drag bit Dual air-rotary with TCI or diamond impregnated/studded casing shoe and DTH hammer with diamond or TCI button bits Sonic rig with TCI or diamond studded casing shoes and core barrel bits Cable tool rig
Steeply Angled Hard Surface[9]	Excess Drill String Deflection Loss of Drill String	Mud-rotary (direct and reverse) TCI edged drag bits, roller cone bits with TCI or milled teeth, PDC bits Air-rotary with TCI edged drag bits, roller cone bits with TCI or milled teeth, PDC bits, DTH hammer with diamond or TCI buttons	Dual air-rotary with carbide or diamond casing shoe and DTH hammer Sonic rig with TCI or diamond studded casing shoes and core barrel bits
Steel Plate	Slow Rate of Penetration (Frequent Drill Bit Replacement, Button Replacement, Tooth Dressing)	Mud-rotary (direct and reverse) with drag bits, roller cone bits Air-rotary with drag bits, roller cone bits, DTH hammer	Dual air-rotary with carbide or diamond casing shoe and DTH hammer Sonic rig with TCI studded or diamond casing shoes and core barrel bits Cable tool rig[10]

[1]Standard concrete slab, compressive strength 28 MPa [4,000 psi], tensile strength 2.8 MPa [400 psi]. Reinforced with 1 or more two-way mats of #6 rebar set on 15.2-cm [6-in] centers depending on thickness of barrier. Grade 60 rebar, minimum yield strength 400 MPa [60,000 psi], normal ductility. Rebar should be horizontally spaced at least as closely as the diameter of the smallest drill bit used by local water well drillers.

[2]Despite statements from INL- and SRS-area water well drillers that they have penetrated reinforced concrete with these types of rigs and bits, it is the collective opinion of drilling SMEs (Black et al., 2001) that a very thick [1.5 m] concrete barrier that is heavily reinforced [two-way mats placed every 15.2 cm [6 in] vertically] would eventually discourage and deter even the most persistent driller.

[3]Percussive action of DTH hammer is unlikely to fracture or deform rebar to the point of failure.

[4]Rotating diamond impregnated casing shoe should be able to cut through rebar if driven with enough torque and thrust.

[5]Rock diameters should be about the same as the smallest diameter conductor casing used for residential water wells constructed in the vicinity of the disposal facility. This would force a driller using a dual air rotary rig to drive the casing through rock even in the unlikely possibility that the rocks have a simple cubic packing. Rocks, such as

chert, that have a high fraction of fine-grained quartz (Mohs hardness of 7) would be abrasion resistant, damage the drill bit teeth and buttons, and minimize the rate of penetration.

[6]Tires consist of two beads, which ensure an air-tight fit to the wheel rim, radial fabric plies that form the sidewalls and casing, circumferential fabric and woven steel fiber belts, and the natural rubber, synthetic rubber, and other chemical compounds that form the tire tread and outside coating of the tire. The non-rubber components are made from steel cable, woven nylon and polyester cord, and belts woven from steel wire. Rubber tread is composed of natural rubber; synthetic butadiene and styrene rubber; carbon black and silica for tear resistance, tensile strength, and abrasion resistance; sulfur and zinc oxide to crosslink the rubber molecules (vulcanization); and other additives to prevent deterioration from heat and ozone.

[7]Tire shredders have two parallel, inward-rotating shafts on which are mounted a series of metal disks with short, curved blades that grab and then shear the tires with a scissors' action. Unlike most natural earth materials, which have high compressive strength, but low fracture toughness, tires are a composite that includes materials that have high tensile strength (steel wires, steel cables, nylon, and polyester fibers) in a matrix of rubber, which is quite ductile and has high fracture toughness. Drill bits are designed to penetrate hard, strong, but brittle rock or unreinforced concrete. Roller cone bits, DTH hammer bits, and cable tool bits penetrate brittle rock by fracturing the rock and comminuting the rock fragments. Drill bits that induce a scraping or sawing action are more likely to be able to tear, shear or cut tough ductile materials such as steel fiber, organic fibers, and rubber. Drag bits with TCI or other metal-carbide or metal-nitride blade inserts, as well as thin kerf TCI or diamond impregnated casing shoes and coring bits may be able to saw or cut their way through rubber. PDC bits that have disc-shaped diamond inserts that are aligned perpendicular to the working face may be able to shear or abrade rubber.

[8]Steel baling wire and steel fencing have high tensile strength and may be able to withstand the pulling and twisting action of rotating drill bits. Because the coils of baling wire and fencing are not embedded in a strong matrix, such as concrete, the wire will be snagged and pulled by the drill bit's teeth and wrapped around the drill string. Bits that cut or shear, such as drag bits with metal-carbide blade inserts or diamond impregnated casing shoes and coring bits, may be able to cut through wire and fencing.

[9]Even if a diamond impregnated or diamond set casing shoe is advanced with the drill bit, the driller would have to very carefully control the advance of the casing to ensure it "cuts" the hard metal surface without being deflected down-dip.

[10]By properly selecting the weight and "sharpness" of the cable tool bit, as well as the weight of the remainder of the tooling on the string, an experienced cable tool operator can fine tune the frequency of the rig's stroke to maximize the impact energy delivered to the 1-in thick steel plate. If a large enough number of high-energy impacts of the bit are delivered, the steel plate may be pierced, punched, or fractured sufficiently to penetrate the waste disposal zone. It is not clear if the hole through the steel plate can be made large enough to admit a small diameter sand bailer that would bring contaminated waste to the surface without becoming stuck.

1

Table G-11 Attributes and Associated Physical Properties and Mechanisms of the Barriers (rows) that Deter Drilling (columns) are Described in the Cell Entries

Barrier	Deterrent Effect			
	A Slow ROP	B Deflection	C Fluid Loss	D Stuck String
1 Heavily Reinforced Concrete	A.i Steel reinforcing bars have much higher fracture toughness than concrete matrix. Cutting or grinding through rebar will be slow and require frequent bit replacement if rebar mats have small [15 cm (6 in)] horizontal and vertical spacing.	NA	NA	NA
2 Large Spherical Rocks	NA	B.i Edge of bit strikes convex rock surface. Vertical force on bit and horizontal force from rock bend and drive the string into the adjacent pore space.	C.i Composite assembly of large rocks has high fluid conductivity. C.ii Moderate to high porosity of barrier can capture and retain large volume of drilling fluid.	D.i Drill advances but borehole collapses and adjacent rocks fall against drill rod, trapping the string.
3 Rubber Tires	A.ii Milled tooth and TCI button bit roller cones will not readily penetrate tires unless the rubber has lost its resilience and metal and cloth reinforcements have lost their tensile strength.	NA	C.iii Composite assembly of tires has moderate to high fluid conductivity, while rubber is undegraded and not fully compressed by overburden. C.iv Moderate to high porosity of barrier can capture and retain large volume of drilling fluid while rubber is undegraded and not fully compressed by overburden.	D.ii Woven metallic and cloth cords and belts become wrapped around drill bit making it difficult, but not impossible, to remove the drill string from what is now an undersized borehole.

Barrier	Deterrent Effect			
	A Slow ROP	B Deflection	C Fluid Loss	D Stuck String
4 Baling Wire or Fencing	A.iii TCI button bit roller cones may not be able to snag and tear wire unless the wire's tensile strength has degraded due to corrosion.	NA	C.v Composite assembly of baling wire coils or panels of chain link fencing may have moderate to high fluid conductivity until corrosion has degraded the steel and the assembly has been fully compressed by overburden. C.vi Moderate porosity of barrier can capture and retain a large volume of drilling fluid so long as the steel wire remains uncorroded and not fully compressed by overburden.	D.iii Steel wire and fencing, if not too corroded, become wrapped around drill bit, making it difficult, but not impossible, to remove drill string through what is now an undersized borehole.
5 Steeply Angled Hard Surface	NA	B.ii Edge of bit strikes hard angled surface. Vertical force on bit and resultant horizontal force from angled surface bend and drive string downdip along the angled surface.	NA	D.iv If angle of hard surface is steep enough, a permanent bend will be set (plastic limit exceeded) in the drill rods. This will make the string difficult, but not impossible, to remove.
6 Steel Plate	A.iv Milled tooth and TCI button bit roller cones will not readily penetrate plate steel unless it has undergone substantial material loss due to corrosion.	NA	NA	NA

How long the driller would continue to drill may depend on how long it typically takes to drill to the aquifer and on the driller's options for relocating the rig. For example, if it typically takes 2 days to reach the aquifer, the driller may accept a very low ROP for 2 to 6 hours; however, if the drill bit is damaged every 30 minutes or so, the driller might trip out and relocate. One may infer that a barrier that makes the driller relocate the rig at least once has temporarily deterred the well driller. If the driller is compelled to relocate the rig enough times to be completely off the disposal site, the deterrent effect is more lasting. Conversely, if the driller is accustomed to encountering units with low ROPs and has bid the job such that the cost of additional time and damaged equipment can be absorbed, the barrier may not have effectively deterred the well driller. For barriers such as Barrier 1, "Heavily Reinforced Concrete," that are designed to limit ROP and to damage drill bits, one should be able to determine which physical properties of the barriers deter water well drilling and which do not. Those physical properties that produce a deterrent effect will depend on the type of drilling method used. For barriers designed to cause excessive loss of circulation, bend drill rods, and seize the drill string, such as Barrier 2, "Large Spherical Rocks," the immediate deterrent effect is simply the inability of the drill rig to continue

to operate. One should be able to determine the physical properties of Barrier 2, "Large Spherical Rocks," that deter water well drilling and those that do not. Table G-12 identifies some criteria that may be used to determine whether the results of intruder barrier tests indicate a barrier would deter a water well driller.

G.5.3.5 Demonstration of Barrier Deterrent Capacity by Analysis and Modeling

The NRC staff has not conducted a thorough investigation of analytical methods that may be used to demonstrate the four deterrent effects identified in Table G-11. The staff used Google Scholar and Scopus to identify a few investigations that could be useful to the licensee intending to demonstrate barrier performance using engineering analyses, including computer numerical simulation methods.

For analyzing barriers whose deterrent effect is Slow ROP, the staff has located several journal articles that describe the use of finite difference, finite element, and distinct element methods to simulate the fragmentation and comminution of different types of geologic media using various types of drill bits. Wang et al. (2021) describe the use of the commercial finite element code ABAQUS to simulate rock breakage at the face of the borehole due to the combined actions of shear from a rotating PDC-impregnated bit and percussion from a mud-driven down hole hammer. The authors describe verification of their modeling method and the values of physical properties used in brittle failure constitutive models by comparing numerical results to measured values of rock materials subject to impact tests in a split Hopkinson pressure bar rig. The introduction and reference sections of Wang et al. (2021) describe the work of other investigators who have used finite element and other numerical methods to simulate the rock breakage process from drilling. Rojek et al. (2011) present a numerical model of the rock drilling process using the distinct (or discrete) element model DEMPack. The introduction of Rojek et al. (2011) cites the work of other investigators who have used commercially available DEM codes, such as PFC2D and UDEC, to simulate rock breakage by drilling. The licensee may need to conduct additional research to identify appropriate methods and constitutive models for predicting the damage that different types of drill bits would cause to composite materials, such as reinforced concrete, that exhibit brittle failure (e.g., concrete) and plastic failure under compression and tension (rebar). Similar challenges could be faced by licensees that elect to use analytical methods to evaluate the penetrability of rolled metal fencing and used tires backfilled with compacted sand-silt-clay mixtures when impacted by different types of drill bits.

The staff found several studies that describe the use of finite difference, finite element, and distinct element methods for predicting wear and breakage of drill bit bodies (e.g., steel and tungsten) and drill bit inserts (e.g., milled tooth roller cones, tungsten carbide roller cones, PCD compact teeth). Mazen et al. (2020) report the development of a mathematical model for estimating the life of PDC bits that is based on calculating bit forces, which are then correlated with observed PDC wear rates. PDC bits are commonly used by the oil well drilling industry for deep drilling (more than 3000 m (10,000 ft)) in moderately abrasive formations. It is not clear whether the model developed by Mazen et al. (2020) could be directly applied to estimating bit wear and tooth failure when drilling into barriers constructed of abrasive materials such as reinforced concrete or concrete that contains metallic aggregates.

1 **Table G-12 Potential Success Criteria for Barrier Deterrence Tests**

Barrier	Deterrent Effect			
	A Slow ROP	B Deflection	C Fluid Loss	D Stuck String
1 Heavily Reinforced Concrete	Driller pulls off site and relocates drill rig because of too much drilling time and too much drill bit damage.	NA	NA	NA
2 Large Spherical Rocks	Driller pulls off site and relocates drill rig because of too much drilling time and too much drill bit damage.	Deviation from vertical cannot be corrected, or one or more drill rods is permanently bent. Driller attempts to recover the drill string and relocates rig.	Drill cuttings are not returned to the surface and ROP becomes too slow. Driller pulls off site and relocates drill rig. Available mud is lost, and driller trips out before becoming stuck. Air compressor cannot keep up with rate of air loss and driller trips out to avoid becoming stuck.	Drill rig has insufficient torque and hoist capacity to extract drill string from downhole. Driller cuts off drill string at surface and relocates rig.
3 Rubber Tires	Driller pulls off site and relocates drill rig because of too much drilling time. Compliance of tire layer observed as “bounciness” of drill string and driller pulls off site and relocates.	NA	Drill cuttings are not returned to the surface and ROP becomes too slow. Driller pulls off site and relocates drill rig. Significant fraction of available mud is lost, and driller trips out before becoming stuck. Air compressor cannot keep up with rate of air loss and driller trips out to avoid becoming stuck.	Drill rig has insufficient torque and hoist capacity to extract drill string from downhole after fabric and cable from tires wrap around drill bit. Driller cuts off drill string at surface and relocates rig.

Barrier	Deterrent Effect			
	A Slow ROP	B Deflection	C Fluid Loss	D Stuck String
4 Baling Wire or Fencing	Driller pulls off site and relocates drill rig because of too much drilling time.	NA	Drill cuttings are not returned to the surface and ROP becomes too slow. Driller pulls off site and relocates drill rig. Significant fraction of available mud is lost, and driller trips out before becoming stuck. Air compressor cannot keep up with rate of air loss and driller trips out to avoid becoming stuck	Drill rig has insufficient torque and hoist capacity to recover drill string after wire and fencing wrap around drill bit. Driller cuts off drill string at surface and relocates rig.
5 Steeply Angled Hard Surface	NA	Deviation from vertical cannot be corrected, or one or more drill rods is permanently bent. Driller attempts to recover drill string and relocate rig.	NA	Drill rig has insufficient torque and hoist capacity to extract drill string from downhole. Driller cuts off drill string at surface and relocates rig.
6 Steel Plate	Driller pulls off site and relocates drill rig because of too much drilling time.	NA	NA	NA

The staff located several studies in which numerical modeling methods simulated the application of bottom hole assemblies to gradually deflect drill strings for horizontal drilling in tight gas shales of other thin, but laterally extensive, unconventional oil and gas reservoirs. However, it is not known how applicable these studies may be to predicting the performance of intruder barriers designed to deter water well drilling by severely deflecting and permanently bending the drill string (Deflection). The investigation by Zifeng and Jingyuan (2008) defines the forces between the borehole and a string steered by bottom hole assemblies and could be used to guide a licensee in analyzing barrier geometries that may bend and permanently deform a water well drill string. There may be other relevant journal articles and reports that the licensee could use to develop an analytical method for estimating the likelihood of deflecting and permanently deforming a relatively short water well drill string that has penetrated a zone of material designed to deter drilling.

The staff identified several studies in the oil and gas drilling literature in which numerical methods were used to predict the severity of a loss of circulation incident and to determine measures that could prevent further fluid loss. Some of these studies may be useful for analyzing the Fluid Loss deterrent effect. Albattat et al. (2022) describe the development of semi-analytical methods and special borehole hydraulics-type curves to model lost circulation in

natural fractures. Albattat et al. (2022) identify other investigations that use analytical and numerical methods to quantify various aspects of the fluid loss problem, including determining the type of formation responsible for fluid loss; managing pressure during drilling to control fluid loss; and adjusting the type, viscosity, and density of drilling fluids to reduce loss rates. If the licensee chooses to use methods developed in the oil and gas industry to evaluate the effectiveness of a proposed barrier, it should identify and evaluate the important differences between oil well drilling and domestic water drilling, including the smaller volumes of drilling fluid used for water wells, the smaller total volume of void space associated with barriers designed to cause fluid loss, and the relative ease with which a water well driller can stop drilling and start a new well if severe drilling problems are encountered when compared to the complexity and expense of moving a large oil drilling operation.

Column 5 of Table G-12 shows four hypothesized mechanisms that may stop domestic water well drillers by the Stuck String deterrent effect. There may be many more than four mechanisms, depending on how many types of barriers the licensee considers that are designed to induce the Stuck String deterrent effect. Johnson et al. (2013) describe and illustrate 14 drill string sticking mechanisms that affect oil well drilling, some of which could be mimicked by a well-designed intruder barrier. For used tire and baling wire and fencing barriers, for which it is postulated that the drill string will be difficult to remove if material wraps around the bit, the mathematical model by Miyoshi et al. (2022) for pull-out of a drill string immersed in a cuttings bed could be useful in determining whether a rig has sufficient pullback capacity to extract the string through the now undersized borehole. Hu et al. (2019) used a light-duty horizontal directional drilling (HDD) machine to install horizontal leachate drains in a municipal solid waste landfill in Tianziling, China. The torque and pullback capacities of the HDD rig (6,000 newton meters (Nm) (4,400 ft-lb) and 200 kilonewtons (kN) (45,000 lb)) were not adequate to advance drilling where “[p]lastics and textiles…tend to become wrapped around the drilling rod, increasing the resisting torque and ultimately preventing rotation” (Hu et al., 2019). Because HDD machines used to install piping and electrical conduits beneath existing infrastructure and natural features (e.g., roadways, canals, rivers) operate differently than domestic water well drilling rigs, a licensee applying the results of Hu et al. (2019) and of other similar field investigations should exercise caution.

G.5.4 Effects of Intruder Barriers on Site Stability

Guidance on site stability provided in Chapter 5.0 and Appendix E of this document remain relevant for near-surface GTCC waste disposal systems whose final cover system include a domestic water well intruder barrier overlain by an engineered surface cover, which together ensure the top of the GTCC waste is disposed at least 5 m (16 ft) below the land surface. The important functions of the cover system are protecting surface material from erosion, minimizing infiltrating water from reaching the waste, minimizing the effects of long-term settlement and subsidence, and ensuring the long-term stability of the site. In general, engineered surface covers are expected to be less important for specialized land disposal facilities than they are for near-surface disposal facilities because surface processes are expected to have no effect or a significantly reduced effect on the deeper zone of waste emplacement in a specialized facility. For example, surface erosion processes are not expected to affect waste disposed more than 30 m (98 ft) below the land surface. Similarly, infiltration is not expected to significantly affect waste disposed below the uppermost aquifer. However, inadvertent human intrusion could affect site stability if an inadvertent intruder (e.g., a well driller) compromises the physical integrity of the disposal site, especially in the zone of waste emplacement. As stated in Chapter 5.0 of this document, guidance on site stability in NUREG-1200, “Standard Review Plan for the

Review of a License Application for a Low-Level Radioactive Waste Disposal Facility," issued April 1994 (NRC, 1994), addresses waste stability, erosion protection, and geotechnical issues, such as slope stability, settlement, and subsidence for near-surface LLW disposal. Additional guidance for specialized land disposal facilities is provided in Section G.5.5.

Section G.5.1 presents six potential physical barriers identified in the literature that may be capable of deterring water well drillers. Table G-11 identifies some of the physical attributes of the six barriers that may deter water well drilling. Physical properties, the mechanisms presumed to deter drilling, and the design for each of the barriers vary significantly. In addition to a barrier's deterrent capacity, the site geology and subsurface strata, the planned depth of the cover system for a near-surface facility, and the depth to the water table or piezometric surface, may affect the design of the intruder barrier. In addition, the choice of a barrier design may affect the long-term stability of the site including the structural stability of vaults or other structures (e.g., boreholes) containing waste disposal containers.

G.5.4.1 Long-term stability

The guidance on long-term site stability in NUREG-1200 focuses on surface drainage and erosion protection, stability of slopes, and settlement and subsidence. Therefore, that guidance is more relevant to a near-surface disposal facility than it is to a specialized land disposal facility. For all land disposal facilities, a long-term stability assessment may involve the structural stability of disposal units (e.g., concrete vaults) and waste containers. Subsidence and differential settlement can cause cracking of the intruder barrier or barriers. Subsidence and differential settlement can also create depressions and localized ponding, which can increase infiltration to the waste horizon of a near-surface disposal facility. As stated in Chapter 5.0, NUREG-1200 and NUREG-1199, "Standard Format and Content of a License Application for a Low-Level Radioactive Waste Disposal Facility," issued January 1991, address subsidence and stability analyses; however, the licensee may need to conduct additional geotechnical evaluations under static and seismic loading to assess the effects of the intruder barriers on long-term stability. The following subsections include guidance for the licensee on some analyses that it may need to consider when using barriers composed of (1) reinforced concrete and (2) used rubber tires. Heavily reinforced concrete is rigid and can impose heavy loads on a near-surface disposal cell, while used tires are likely to be deformable and impose relatively light loads on near-surface disposal cells.

G.5.4.2 Reinforced Concrete

Section G.5.1.7 provides an example of a concrete cover consisting of a 1.5 m (5 ft) thick heavily reinforced slab above near-surface waste disposal vaults with a 6.1 to 9.1 m (20 to 30 ft) thick cover cap placed on top of the slab. The steel rebar in the concrete is expected to provide resistance to rotary drill bits trying to drill through the concrete. The steel rebar can be placed strategically to maximize the likelihood that the drill bit will encounter rebar, perhaps multiple times, and minimize the likelihood that the drill bit will pass through the spaces between the rebar. Section G.5.1.7 suggests placing two-way horizontal rebar mats at each 15 cm (6 in) elevation increment in the concrete slab. Rebar mats may be horizontally offset from one another to further decrease the likelihood of drilling through the concrete without encountering rebar. Construction of slabs with heavy reinforcement should follow American Concrete Institute Standards 207.1, 207.2, and 207.4 for current practice and guidance on mass concreting for controlling the heat of hydration and minimizing thermal cracks in concrete.

For a near-surface disposal facility that uses vaults for the disposal units, a reinforced concrete intruder barrier may be constructed as the roof of the vault disposal structure or constructed onsite as a concrete mat above the vaults' lids. The 1.5 m (5 ft) thick concrete mat may result in heavy loads that must be borne by the underlying vault structure(s). Hence the structural performance of the concrete barrier may depend on the stability of the underlying waste container vaults. The integrity of the vault structure should be evaluated, and soil/rock-structure interaction analyses may be needed, depending on the bearing strength of the subsurface soil and its capacity to resist settlement under static and seismic loading. Differential settlement can cause mechanical cracks and loss of integrity of the concrete barrier mat. Long-term site stability assessments may need to consider the chemical degradation of cementitious material and corrosion of rebar that may affect the structural integrity of the concrete slab. The stability assessment should evaluate settlement of the fill material above the concrete slab for potential subsidence and cracking or ponding of the top cover.

G.5.4.3 Layer of Rubber Tires

Tolan (1993) indicates that the suggestion of using tires to deter drilling came from drillers who experienced difficulties advancing boreholes when drilling into tires disposed in municipal solid waste landfills. Tolan did not state whether the tires were loosely scattered in one or more landfill layers, consisted of many tires in a "tire-dense" deposit, or were bound together as bales or some other assemblage. Wittmeyer (2026) found that air and mud rotary drilling both penetrated a barrier made of four layers of tires backfilled with a low-strength flowable material. Until additional field tests are conducted on a prototype intruder barrier constructed from tires, it may be difficult to determine how to construct a tire barrier to replicate the deterrent behavior observed in the municipal waste landfill.

Layers of loose used tires, horizontal layers of interconnected and backfilled tires, and compacted bales of waste tires have been used in civil engineering applications, especially road engineering, embankment construction, slope stabilization, road foundations on soft ground, gravity retaining structures, and backfill material for retaining walls. Garga and O'Shaughnessy (2000) and Sayao et al. (2002) investigated the behavior of soil-tire gravity walls. In this study, tires were placed side by side in successive horizontal layers and tied together with polypropylene rope to form a mat. The layer was then filled with sand. Layers of tire mats were used to construct a 60 m (200 ft) long and 4 m (13 ft) high retaining wall. The authors do not provide information on the design process; however, they did evaluate the performance of the retaining wall by monitoring its deformation over time. Waste tire bales are sometimes used in lieu of conventional construction materials for certain applications because of their (1) unit (specific) weight being less than typical soils, (2) high frictional resistance, (3) favorable damping properties, and (4) good drainage properties (Duda and Siwowski, 2021). Tire bales are made by compressing approximately 100 tires into a large block about 150 cm (5 ft) wide, 150 cm (5 ft) long and 75 cm (2.5 ft) high, bound with galvanized or stainless steel wire or steel bands (Zornberg et al., 2005). The bales are laid horizontally in layers and stacked vertically to form the desired geometry of the structure. The bales are typically assumed to act as discrete blocks for short-term conditions, in which the controlling mode of failure is sliding along the jagged surfaces of the bales.

For the design of transportation structures using tire bales, Freilich and Zornberg (2009 and 2010) studied the index, mechanical, and hydraulic properties of tire bales using field and laboratory test methods. Engineering properties of tire bales were also studied by Zornberg and

La Rocque (2006), Zornberg et al. (2005), and Driscoll and Siwowski (2021a and 2021b). These tests typically addressed (1) compressibility characteristics based on deformation measurements of tire bales, (2) confined compressibility of tire bales (with a dry sand infill material), including static and cyclic modulus evaluation, (3) rebound and potential lateral expansive pressure, (4) time-dependent deformations of single bales under sustained load (i.e., creep) for up to 1,000 hours, and (5) shear strength along the interface between two vertically stacked bales (Zornberg et al., 2005). Driscoll and Siwowski (2021a) conducted stability and settlement analyses of embankments constructed with tire bales. The bales were filled with medium sand and rubber aggregate. They used equilibrium and finite element methods to model settlement with engineering properties and model parameters determined by experiment. They performed a comparative study between embankments constructed with common sand and tire bales. The study concluded that the application of tire bales generally enhanced the embankment stability while reducing its settlement and the normal stress in the subsoil.

Based on limited analytical studies, using tire bales in embankments has some advantages over standard earthwork support methods with respect to subsidence control and stability; however, the compressibility of tire bales may affect their mechanical and hydraulic characteristics under long-term sustained loading. Licensees that elect to use tire bales for their intruder barrier may be able to use the studies identified previously to guide field and numerical studies of the effect of tire-based intruder barriers on site stability. Zornberg et al. (2005) note that tire bales show no biological or chemical degradation during 14-20 years of use; however, they also note there are no comprehensive studies addressing long-term durability of tire bales under deleterious environmental conditions. Licensees that adopt tire-based intruder barriers may consider conducting accelerated environmental testing or analyses to demonstrate the ability of the tire barrier to deter water well drilling for up to 500 years following closure for use in a near-surface disposal facility. Because of the greater biological and chemical degradation of tire barriers compared to other types of intrusion barriers (e.g., thick reinforced concrete or metal barriers), tire barriers are not expected to be an effective part of a multi-barrier system for the longer timeframe required for inadvertent intruder prevention for a specialized land disposal facility with significant quantities of long-lived radionuclides.

G.5.5 Engineered Barriers to Inadvertent Intrusion at a Specialized Land Disposal Facility

The primary difference between inadvertent intruder barriers for near-surface and specialized land disposal facilities is the length of time intruders must be prevented from disrupting or contacting the waste. Under 10 CFR 61.52(a)(3), a near-surface disposal facility must include an intruder barrier that deters inadvertent intrusion for at least 500 years after site closure. In contrast, because of the greater hazard posed by the waste that may be disposed in a specialized land disposal facility, 10 CFR 61.13(b)(4) requires the licensee to demonstrate that an inadvertent intruder will be prevented from disrupting or contacting the waste during any part of the compliance period in which the waste remains a radiological hazard. Because a specialized land disposal facility may receive significant quantities of long-lived radionuclides, this requirement could extend to a compliance period of up to 10,000 years. Given the difficulty of demonstrating that a single engineered barrier can maintain performance over such a long timeframe, 10 CFR 61.13(b)(4) requires that the demonstration of intruder protection for a specialized land disposal facility must rely on a combination of engineered barriers and natural features.

This section addresses engineered barriers that can serve as one part of a multi-barrier system to prevent an inadvertent intruder from contacting or disrupting the waste while it remains a radiological hazard during the compliance period. Additional considerations, controls, and designs to deter inadvertent intruders from contacting waste over long timeframes are discussed in Section G.7 (e.g., significant disposal depth, disposal site locations away from natural resources, limited areal footprints, advanced wasteforms).

Results from drilling barrier tests described in Wittmeyer (2026), suggest that heavily reinforced concrete can deter medium-duty-direct-mud rotary and direct-air rotary rigs. The reinforcing bars used in these tests were made from standard carbon steel, which has a low corrosion rate while the concrete remains intact and the aqueous chemistry of the concrete-rebar system has high pH conditions (10.5 to 13.5). However, once fractures or other fast-flow pathways develop in the reinforced concrete, the pH of the water contacting the carbon steel will decrease and the rebar is susceptible to corrosion. Therefore, the intruder deterrence capability of a traditional reinforced concrete intrusion barrier in the near surface is likely to degrade after centuries of service, and is unlikely to prevent a significant barrier to a driller for a full 10,000-year compliance period. However, as part of a multi-barrier system, a thick (e.g., meters) reinforced concrete barrier in the near surface could provide an early indication to a well driller that they are drilling into an engineered structure.

Reinforced concrete could also be used in configurations other than a near-surface slab. For example, if reinforced concrete is used to backfill and seal the upper 500 m (1,650 ft) of a borehole, careful design of the concrete mixture and proper selection of the steel or other metal used for reinforcing could sustain its capacity to deter penetration by drilling for many centuries following permanent closure. Even if the reinforcing is degraded enough that the drill rig can penetrate the upper portion of the borehole seal, after drilling down to a depth of one drill rod, or approximately 6 m (20 ft), the concrete and other manmade material in the cuttings should cause a driller to cease operations because of encountering unexpected subsurface conditions.

Additional barriers that are generally not considered for a near-surface facility could be appropriate for a specialized facility and could provide a more significant barrier to intrusion over a longer timeframe. For example, as discussed in Section G.5.1.4, the subject matter experts elicited by Black et al. (2001) indicated that a steeply-angled shield made of a hard metal such as titanium would be almost certain to deflect a drill bit and prevent a well from penetrating the waste area. In addition, the corrosion resistance of titanium could ensure it provides a barrier to intrusion for hundreds to thousands of years after site closure, depending on the geochemical conditions of the site. Although the DOE did not consider a titanium plate further because it was cost prohibitive for the application at that time, such a barrier may not be cost prohibitive for a disposal unit with a much smaller footprint, such as a borehole. Similarly, a licensee could propose an angled shield made of another material, such a specialized alloy, that would also exhibit the strength and corrosion resistance needed to present a significant barrier to a well driller for the longer times needed for significant quantities of long-lived waste.

G.6 Operational Safety and Security Evaluations

G.6.1 Operational Safety Assessment

A licensee's assurance of radiological safety during the operational period should consider occupational and public radiation exposures from routine operating and accident conditions. Radiation exposures under routine conditions and likely accidents (i.e., those expected to occur

one or more times during the operational period) are primarily addressed by a radiological protection plan that complies with the radiation protection regulations in 10 CFR Part 20. Because GTCC waste streams, by definition, include higher radionuclide inventories than typical (Class A, B, and C) LLW, they can lead to higher doses under routine operations and under non-routine accident conditions; in particular, under accident conditions involving the release of radionuclides. Exposures from non-routine unlikely accidents must be evaluated to gain insights into the hazards associated with specific waste streams and assess whether additional controls may be needed to reduce the likelihood or consequence and, therefore, the overall risk of these events. This section discusses the additional technical analyses that may be needed to address the radiological hazards that are unique to GTCC waste and that warrant additional consideration by licensees interested in disposing these materials at a commercial LLW facility.

This section refers to the technical analysis required under 10 CFR 61.13(c) as an operational safety assessment. Although operational safety is primarily addressed through compliance with the radiation protection standards in 10 CFR Part 20, the requirement in 10 CFR 61.13(c) incorporates and extends the operational safety considerations of licensees proposing to receive, handle, store, transfer, and dispose of GTCC waste to ensure they include analyses of operational exposures and exposures from unlikely accidents. The NRC staff understands that these additional technical analyses to address the requirement in §61.13(c) are closely related to and integrated with a licensee's description of its radiation protection program and related operational safety analyses. Therefore, the staff considers that the licensee would likely document the 10 CFR 61.13(c) operational safety assessment as an additional 10 CFR 61.13(c) operational safety analysis within the part of a license application or amendment request that addresses radiation protection and compliance with 10 CFR Part 20 and not necessarily as a separate document. Licensees who meet the criteria in 10 CFR 61.1(b) must complete an operational safety assessment in accordance with 10 CFR 61.13(f).

G.6.1.1 Exposures from Routine Operations

A radiation protection plan that complies with 10 CFR Part 20 adequately addresses the radiological protection of workers and the public from normal operations at an LLW facility that involves GTCC waste. This section highlights the unique circumstances associated with some expected GTCC waste streams that may require additional technical analyses to demonstrate that proposed operations can be conducted in accordance with the applicable radiation safety standards.

Under routine operations involving the receipt, storage, transfer, and emplacement of GTCC waste, occupational exposures would primarily be a function of the external radiation fields associated with packaged waste, the distance to nearby workers, and the duration of worker activities and exposure. External gamma exposure rates for packaged GTCC waste streams result from radionuclide inventories and shielding and would vary by waste stream. Most packaged GTCC waste streams that the NRC staff considers acceptable for near-surface disposal would involve dose rates that are typical of existing LLW or are elevated but would not be expected to present unique challenges for operators that apply typical controls to limit occupational exposures. An exception is that some canisters containing activated metals[3] and GTCC wastes originating from the reprocessing of spent fuel could have significant surface gamma dose rates well in excess of 100 roentgen (R) per hour (hr) and should consider

[3] The material of the canisters is not activated metals; rather, canisters are used to package or contain activated metals. Sometimes those canisters are referred to as activated metal canisters.

approaches for minimizing the exposure rates such as the use of shielding and/or increased storage times by the waste generator. Other waste streams in DOE's GTCC FEIS that are labeled for remote handling have unshielded gamma dose rates in the 10 to 100 R/hr range.

For operations involving waste packages with surface gamma dose rates above 2 millisieverts (mSv)/hr (200 millirem (mrem)/hr), licensees should describe structures, systems, and components important to safety and related detailed operational procedures for remote handling. Facilities that would receive shipments of canisters containing waste materials having a high dose rate evaluate occupational doses from the planned receipt, storage, transfer, and emplacement operations to demonstrate that operations can be conducted within occupational limits (and related action levels) or provide a technical basis for how occupational exposures would be controlled to preclude the need for further analysis. For a representative and bounding operation, licensees should evaluate the dose rates and estimate and document occupational doses for operations that could result in elevated occupational exposures. This includes describing the operational activities in sufficient detail to identify the structures, systems, and components and other relevant items (e.g., plans and procedures) that are relied upon for safe operations. Licensees should consider likely operational occurrences (e.g., likely accidents) and how such conditions could affect operational safety, external dose rates, or the release of radioactive material.

Licensees should document the controls that would be applied to address occupational dose limits. Controls to reduce occupational exposures include use of shielding or other design-related measures, increased distance between the source and workers, reduced time to complete the operation (e.g., efficiency), and enhanced planning for special shipments (e.g., additional training, dry runs, or curtailing other activities during placement). Licensees should include these considerations, as needed, in developing waste acceptance criteria to optimize operational activities involving activated metal or high-gamma remote-handled wastes to effectively manage occupational exposures during receipt, storage, transfer, and emplacement. An example is the use of lead-shielded canisters. For activated metal wastes, the DOE has described a half-shielded activated metal canister fitted with a 9.71-cm (3.825-in.) lead shield that could reduce radiation rates at the surface to less than 2 mSv/hr (200 mrem/h) (DOE, 2016a,b).

An additional consideration for evaluating operations involving unshielded high-gamma-dose-rate activated metal waste or other remote-handled waste streams is the potential contribution to occupational doses from secondary (air-scattered) radiation. The DOE evaluates the contribution of air-scattered gamma radiation in its GTCC FEIS using the computer code MicroSkyshine[4] (Grove Software, Inc., 2008) to examine the activated metal waste containers and the conceptual geometric configurations of a vault, trench, and borehole. The DOE uses this computer code to address radiation exposures from secondary radiation when there is shielding between the radiation source (waste packages) and a potentially exposed individual (nearby worker). Although the shielding would greatly reduce the dose from direct radiation, the dose from air-scattered radiation could be significant. The DOE described how the gamma radiation would be emitted from the waste packages to the air above the disposal unit and would be scattered by air molecules in the atmosphere, and then a small fraction of the scattered radiation would be directed toward a nearby worker. Although significantly lower than the direct exposure associated with the activated metal waste

[4] The NRC does not endorse commercial products but will note software products that are available and have been used.

containers, the DOE found the exposure rates from skyshine radiation could exceed 10 mR/h and approach 100 mR/h close to the disposal facility if several waste containers were grouped together before placement of the overlying cover. The DOE concluded the exposure rates indicate the need for shielding to protect individuals working at the site under these conditions. NUREG-2215, "Standard Review Plan for Spent Fuel Dry Storage Systems and Facilities, Final Report," issued April 2020 (NRC, 2020), includes additional NRC guidance on shielding calculations. NUREG-2215 applies to storage of spent nuclear fuel and GTCC waste. The technical information and guidance on shielding calculations applies to shielding calculations required for technical analysis of GTCC waste handling, storage, and disposal.

Licensees of facilities that would receive shipments of unshielded canisters of high-dose-rate waste materials should evaluate the annual public dose to a hypothetical member of the public at the nearest facility boundary from the operations area. NUREG-2215 includes applicable guidance related to calculating public external radiation doses from operations that may be informative in this context.

G.6.1.2 Exposures from Unlikely Accidents

Unlikely accident conditions are less likely to occur than likely accidents but present the potential for increased radiation exposures to workers and the public. The increased hazards associated with GTCC waste streams relative to typical LLW necessitate a more thorough analysis of potential accidents, their likelihoods, and their consequences to gain insights about the adequacy of proposed operational controls to maintain safety and demonstrate compliance with applicable requirements.

The operational safety assessment of unlikely accidents involving GTCC waste is focused on estimating dose consequences to a hypothetical member of the public at the nearest downwind facility boundary. While the NRC staff recognizes that the consequences of accidents on workers would be higher than for members of the public, the analysis is aimed at providing insights that can generally limit accident risks. The analysis considers both the likelihood and consequences of accidents and, therefore, would limit accident risks to both workers and the public.

Within the context of an operational safety assessment for the disposal of GTCC waste, the NRC staff considers it reasonable to define likely accidents as having at least once chance of occurring during operations (e.g., greater than or equal to approximately 2.5×10^{-2} per year for a facility with an operational period of 40 years). Likely accidents should be evaluated as part of normal operations and compliance with the radiological protection requirements in 10 CFR Part 20. Unlikely accidents typically can span a range that the NRC staff generally expects to have less than one chance of occurring during operations and at least one chance in 1,000 of occurring in any year within the period of operations. Unlikely accidents may be defined and justified by licensees.

In accordance with the requirements in 10 CFR 61.13(c), analyses of the protection of individuals during operations must include assessments of expected exposures due to unlikely accidents during handling, storage, and disposal operations involving GTCC waste. Accidents that should be evaluated include accident sequences applicable to LLW facilities such as fire, handling events, and other credible accidents. An assessment of expected exposures from operational accidents involving GTCC waste identifies structures, systems, and components important to safety. Licensees should consider the initiating events for unlikely accidents in

determining event sequences only if they are reasonable (i.e., based on the proposed operations, the characteristics of the natural and the human environment, and consistent with precedents adopted for nuclear facilities with comparable or higher risks to workers and the public).

G.6.1.2.1 Description of Waste and Facility Characteristics

To support the identification of hazards, initiating events, and accident sequences, licensees should describe (1) the GTCC waste characteristics and (2) the site and facility characteristics, including the facility design and the structures, systems, components, and other items relied upon for safety. GTCC waste characteristics include radionuclide inventory, dose rate, wasteform composition and amount, packaging and drop fracture resistance, shielding, and thermal characteristics. Applicable site characteristics vary but should include those that inform the potential for and consequences of natural events (e.g., tornados, earthquakes), meteorology and regional climatology, regional and local surface and groundwater hydrology, site geology and seismology, geomorphology, land use, structures and facilities, and local population characteristics. Natural events may act as an initiator for an accident sequence or may impair operational and management controls applied in response to a release.

Facility characteristics include general arrangement drawings of buildings; design codes and standards employed; materials of construction; equipment layout; electrical systems; crane systems; heating, air conditioning, and ventilation and filtration systems; confinement systems; decontamination systems; radiation monitoring; fire suppression; waste package receipt, transfer, and handling systems; loading and unloading systems; remote handling systems; emergency and radiological safety systems; communications and control systems; power distribution and backup power; shielding; and water supply systems. Other items relied upon for safety include operational processes and procedures that address normal operations, maintenance, and off normal or emergency conditions. Operational processes and procedures should describe basic operational process function and theory related to receipt, storage, transfer, and emplacement of GTCC waste streams, including descriptions of ranges and limits for measured variables to ensure safe operation.

G.6.1.2.2 Identification of Hazards and Initiating Events

Other NRC guidance can inform methods used to identify hazards and initiating events and should be consistent with standard industry practice. Examples include NUREG/CR-2300, "PRA Procedures Guide—A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants," issued January 1983 (NRC, 1983); NUREG-1513, "Integrated Safety Analysis Guidance Document," issued May 2001 (NRC, 2001); and NUREG-1520, "Standard Review Plan for Fuel Cycle Facilities License Applications, Final Report," Revision 2, issued June 2015 (NRC, 2015b). DOE guidance and technical reports provide useful and detailed information related to the identification of hazards and initiating events (DOE, 2016a,b; 2015a; 2007). The NRC staff recognizes that guidance applicable to other types of facilities that are more complex or involve higher hazards or risks relative to proposed GTCC waste operations at an LLW disposal facility may go beyond what is necessary to evaluate GTCC waste operations. This guidance is useful to provide concepts and methods, where applicable. Licensees should clearly describe and justify the methods they use to identify hazards and initiating events. Included hazards and initiating events should be consistent with the proposed GTCC waste operations and supporting information. Licensees should provide technical bases for excluding specific hazards and initiating events (e.g., based on waste characteristics, site and facility

information, and operating procedures). Licensees should address whether hazards and initiating events that involve a release are credible and should provide technical bases for these determinations.

G.6.1.2.3 Identification of Event Sequences

Hazards and initiating events should be explained and justified and appropriately considered in the identification of event sequences. Licensees should consider reasonable combinations of initiating events and associated event sequences that could lead to exposure of the public to radiation. Methods for event sequence identification are selected consistent with site-specific information, NRC guidance or standard industry practices, or other adequate justification. Assumptions made in identifying event sequences must be valid and reasonable. The probabilities of applicable event sequences should consider relevant information applicable to similar operations or sites. The DOE GTCC FEIS includes detailed information about a variety of applicable accident sequences and initiating events and provides probability estimates and associated references that may be relevant to other sites and proposed disposal methods.

G.6.1.2.4 Consequence Analysis

Accident consequence calculations for GTCC waste disposal are acute (short-term) offsite public individual dose estimates from estimated releases that result from credible accident or event sequences. Typically, inhalation will be the limiting pathway and other pathways can be omitted. Exceptions include circumstances in which radionuclides may be released that produce a higher external dose relative to the inhalation dose; under these circumstances, pathways such as cloudshine and groundshine should be considered as well. A simple accident source term and acute inhalation dose calculation should be sufficient and is explained further in this section. More detailed consequence modeling is an option if licensees would prefer to conduct more realistic (less conservative) site or facility-specific air release, transport, and dose calculations.

<u>Conceptual and Mathematical Models</u>

Conceptually, the GTCC waste operations accident of greatest importance involves (1) a breach of containment, (2) subsequent fractional release of the radioactive material inventory in a package or packages, (3) airborne transport and dispersion of a plume, and (4) a downwind air concentration of radioactive material at the facility boundary where a public receptor is assumed to be located. The receptor is exposed by inhalation to the passage of a plume of released GTCC waste material. The receptor breathing rate and exposure duration determine the intake of radioactive material, and inhalation dosimetry (dose coefficients) converts the intake to an individual dose (typically an effective dose equivalent or other approved metric). Calculations that implement this conceptual model involve evaluating the source term and then estimating the dose. Because the source term is an important part of the dose calculation, this section first describes the source term calculation, followed by the remainder of the dose calculation.

Evaluating the risks and consequences of accident scenarios involves an understanding of the barriers to radioactive materials release, the degree that barriers can fail under specific accident scenarios, and the estimated fractional release of the radioactive material inventory. Technical reports that provide guidance on how to estimate releases of radioactive materials from nonreactor nuclear facilities include NUREG/CR-6410, “Nuclear Fuel Cycle Facility Accident Analysis Handbook,” issued March 1998 (NRC, 1998), and DOE-HDBK-3010-94, “DOE

Handbook: Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities, Volume I—Analysis of Experimental Data," issued 1994 (DOE, 1994). These references evaluate source terms as follows:

$$Source\ Term = MAR \cdot DR \cdot ARF \cdot RF \cdot LPF \quad (1)$$

MAR Material at Risk: the amount of radioactive material available to be acted upon by a physical stress that could result in a release

DR Damage Ratio: the fraction of Material at Risk that is impacted by the accident conditions

ARF Airborne Release Fraction: the fraction of the affected material that is released to air (for discreet release events) or the airborne release rate (for continuous release events)

RF Respirable Fraction: the fraction of material released to air that can be inhaled into the human respiratory system (typically, particles of 10 micrometer (μm) diameter or less)

LPF Leak Path Factor: the fraction of released material that transports through confinement, which could involve several layers (e.g., wasteform, packaging, facility structures)

For a specific accident scenario, the product of *DR*, *ARF*, *RF*, and *LPF* is the release fraction. The DOE has used this methodology to evaluate the source terms for several types of accident scenarios applicable to the disposal of GTCC waste streams at LLW sites and derived a set of release fractions summarized in Table G-13 (DOE, 2016a,b).

Table G-13 Example Estimated Release Fractions for Hypothetical Facility Accidents

Accident Scenario	DR	ARF	RF	LPF	Release Fraction
Single drum drops, lid failure, in building	0.25	0.001	0.1	0.001	2.5×10^{-8}
Single small white box drops, lid failure, in building	0.25	0.001	0.1	0.001	2.5×10^{-8}
Three drums drop, puncture, lid failure, in building	0.15	0.001	0.1	0.001	4.5×10^{-8}
Two small white boxes drop, puncture, lid failure, in building	0.175	0.001	0.1	0.001	3.5×10^{-8}
Single drum drop, lid failure, outside	0.25	0.001	0.1	1	2.5×10^{-5}
Single small white box drop, lid failure, outside	0.25	0.001	0.1	1	2.5×10^{-5}
Three drums drop, puncture, lid failure, outside	0.15	0.001	0.1	1	4.5×10^{-5}
Two small white boxes drop, puncture, lid failure, outside	0.175	0.001	0.1	1	3.5×10^{-5}
Fire inside the building, one small white box affected	1	0.0005	1	1	5.0×10^{-4}
Single remote handled waste canister breach	0.01	0.001	0.1	0.001	1.0×10^{-9}
Earthquake affects 18 pallets, each with four contact handled drums	0.1	0.001	0.1	1	7.2×10^{-4}
Tornado missile hits one small white box, contents released	1	0.001	0.1	1	1.0×10^{-4}

The release fractions are highly variable and are lower for less severe accidents or accidents involving more robust containment (or both) and higher for more severe accidents or accidents involving less robust containment (or both). The DOE analysis results, with release fractions that range from 1.0×10^{-9} to 5.0×10^{-4}, demonstrate the importance of accounting for the range and characteristics of possible accidents based on facility design and implementation, as well as the characteristics of the wasteform and applicable additional containment layers in estimating the consequences of potential accidents associated with GTCC waste disposal. The values found in Table G-13 should not be used without justification. The release fractions can have a very large range and are dependent on the characteristics of the waste, wasteforms, engineered barriers, controls, and accident. The wasteform can have a significant impact on some of the release factors (e.g., activated metal waste would be less sensitive to drops and fires than other brittle or combustible wasteforms). The values used should be consistent with the characteristics of the waste.

The source term that includes the *MAR* and the release fraction (the product of the first four factors in Table G-13) is part of an acute offsite dose calculation that evaluates the short-term dose to a member of the public at the facility boundary during an accidental release of GTCC waste material. A simple acute offsite public exposure calculation assumes a member of the public is located at a specified distance downwind from a release event of specified duration

that releases a specified percentage of a waste package contents into the air. The location of the receptor is typically at the nearest downwind offsite location at a specific site or facility.

Equation 2 describes the mathematical model for an acute offsite public exposure scenario:

$$Dose_p = \sum_i \frac{C_i\, V_{pkg}\, F_{rel}}{T_{rel}} \frac{\chi}{Q_{acc}} \quad B_{rate}\, E_{time}\, DC_{inh,i}\, U_{conv} \tag{2}$$

$Dose_p$ acute total offsite public dose for an accidental release by GTCC waste stream (mrem)

C_i inventory concentration for a GTCC waste stream for radionuclide i (Ci/m^3)

V_{pkg} package volume (m^3)

F_{rel} release fraction (unitless)

T_{rel} release duration (seconds (sec))

χ/Q_{acc} downwind centerline ground-level time-integrated air concentration at the receptor location relative to release rate (sec/m^3)

B_{rate} inhalation rate (m^3/sec)

E_{time} exposure duration (sec)

$DC_{inh,i}$ inhalation dose coefficient for radionuclide i (Sv/Bq)

U_{conv} unit conversions for mrem/Sv and Bq/Ci

The product of the terms C_i and V_{pkg} represent the *MAR* in Equation 1, and F_{rel} is the product of *DR*, *ARF*, *RF*, and *LPF*, as shown in Table G-13. The calculation in Equation 2 is a conservative screening calculation to gain insights into the hazard presented by the materials in different waste streams.

Input Parameters

This section provides additional information about input parameters applicable to acute offsite public dose calculations that may be helpful to licensees implementing these calculations.

Relative Downwind Air Concentration (χ/Q) (sec/m^3)

The downwind concentration is estimated based on applying an available χ/Q value from a Gaussian plume calculation that involves site-specific or reasonably conservative dispersion assumptions. The χ/Q is the relative ground-level time-integrated downwind centerline air concentration at the receptor location. The χ/Q can be calculated by executing an atmospheric dispersion model or can be selected from references that provide previously calculated values under specific conditions or assumptions that are either representative or conservative for the site-specific calculation. If detailed dispersion modeling software is used, then the software may include the remainder of the dose calculation in Equation 2, thereby omitting the need for further calculations outside the software. The DOE has documented χ/Q values that may be applicable

to some site-specific calculations (DOE, 2015b). Licensees should provide the bases for selection or modeling of χ/Q values as either representative or conservative for site-specific conditions. A conservative approach to atmospheric dispersion modeling or selection of χ/Q values involves assumptions about relatively stable atmospheric conditions (e.g., F stability class), low wind speed (e.g., 1 m/sec), and the shortest distance to a publicly accessible offsite receptor location. The conservative selection of χ/Q values would be expected to produce higher consequence estimates relative to site-specific dispersion modeling but may be a favorable option for licensees aiming to avoid the complexities associated with site-specific atmospheric dispersion modeling.

Release Fraction

For a specific accident scenario, the product of DR, ARF, RF, and LPF from Equation 1 is the release fraction (examples provided in Table G-13). Technical reports that provide guidance on how to estimate releases of radioactive materials from nonreactor nuclear facilities include NUREG/CR-6410 and the DOE handbook DOE-HDBK-3010-94. These release fractions are very uncertain and, if important to the assessment, are best if measured in controlled experiments. Though the references provide values for different scenarios, release fractions are problem specific and can change significantly. Conservatism is used when developing handbook values such that site-specific values will more likely be lower than documented values but can sometimes be higher.

Inhalation Rate (m^3/sec)

Inhalation rates, also called breathing rates, specify the volume of human air intake per unit time. Inhalation rates vary depending on various factors including age, gender, weight, health status, and the level of physical activity (EPA, 2011). An offsite public receptor should be consistent with or conservative for typical local practices surrounding a site. For example, a typical adult member of the public engaging in general outdoor activities would have a breathing rate of about 1.4 m^3/hr (49 ft^3/hr) (NRC, 1999c). Higher intensity activities such as construction work or active recreational activities might lead to higher values. EPA (2011) provides a complete description of available data with recommendations that can be informative.

Exposure Duration (sec)

For short-duration accidental releases, the duration of exposure of the public receptor to the plume can be set to the same value as the release duration to allow exposure to the full duration of the plume passage. For longer duration events such as a fire, other considerations may limit the amount of time a receptor would be exposed to the plume. The exposure duration, therefore, depends on the accident scenario, and licensees should provide the rationale for selected values.

Waste Package Volume (m^3)

The waste package volume is part of the estimation of *MAR* and should be based on the types of packages expected to be used for specific waste streams and how the packages would be filled. For example, the volume of a 55-gal drum of 0.61 m (2.0 ft) diameter filled to a depth of 0.5 m (1.6 ft) with waste material would be 0.15 m^3 (5.3 ft^3).

Inhalation Dose Coefficients (Sv/Bq)

Inhalation dose coefficients convert the amount of inhaled radioactivity to dose. The dose coefficients are typically selected from published results of internal dosimetry modeling (e.g., ICRP, 1996; EPA, 1988). This involves a biokinetic model that evaluates the partitioning of inhaled radioactive elements within the body supported by available biokinetic data and a dosimetric model that calculates dose to specific tissues. These primary components of dosimetry modeling (biokinetic model, biokinetic data, dosimetric model) have evolved over time, resulting in differences among available methods for converting human intakes of radioactive material to dose. The NRC accepts inhalation dose coefficients documented by EPA (1988) that calculate effective dose equivalents. The NRC has approved other methods (e.g., effective dose coefficients from the International Commission on Radiation Protection (ICRP, 1996)) on request (See Section 3.4.2 of the main report). If basic information about the chemical form of the released material is known, it can be used to select the most applicable coefficients based on that information. In the absence of such information, the conservative approach is to select the highest values. Licensees should document the source and the approach to selecting the dose coefficients used in the accident dose calculations.

More Advanced Consequence Models

Limited options are available for modeling software capable of conducting more detailed public dose calculations from user-specified acute air releases of a wide variety of radionuclides. The NRC staff expects that more detailed models would not typically be necessary in the context of these GTCC waste accident analyses, but they are an available option for licensees that may prefer to conduct more detailed but less conservative calculations. These more detailed calculations would typically require greater effort and resources relative to the previously described screening-style calculations.

One available program that has these capabilities is the GENII Version 2.10 software developed by Pacific Northwest National Laboratory for the DOE. GENII Version 2.10 is a documented (Napier, 2010; Napier et al., 2009) and peer-reviewed set of Windows-based programs for calculating radiation dose and risk from radionuclides released to the environment. The code is part of the Radiation Protection Computer Code Analysis and Maintenance Program (RAMP) at the NRC. It is part of a set of quality-assured and configuration-controlled safety analysis codes managed and maintained for the DOE Safety Software Central Registry and for the NRC.

GENII Version 2.10 allows users to input radionuclide-specific air release fluxes and offers a straight-line Gaussian plume model to calculate downwind air concentrations based on the specified releases and commonly known inputs for the Gaussian plume model (e.g., windspeed, meteorological data, dispersion coefficients). Plume rise calculations are user selectable and consider both momentum and thermal buoyancy-based plume rise. The air model calculates time-integrated ground-level air concentrations (e.g., Ci-sec/m^3) at specified downwind distances from the source. The air model also calculates the receptor's external exposure to the passing plume as well as deposition of particulates to the ground surface for groundshine calculations. A puff air transport model is available in addition to the Gaussian plume model.

GENII Version 2.10 includes complex exposure and intake models that involve a variety of potential pathways. Exposure pathways are user selectable and controlled by additional input parameters. Many of the complex and input-parameter-intensive pathways pertain to terrestrial exposure pathways (soil contamination, food, livestock production, and human intake) that

would not typically be needed for acute accident dose calculations (thereby simplifying input needs in this GTCC waste modeling context). The models provide the flexibility to compute time-averaged air concentrations and specify receptor exposure time and can calculate acute doses from air releases based on the product of the time-integrated air concentration (Ci-sec/m^3) from an instantaneous air release (Ci/sec), the receptor breathing rate (m^3/sec), and the inhalation dose coefficient (rem/Ci-inhaled). This approach is conceptually consistent with an exposure scenario in which the receptor is exposed to the full passage of the plume as previously described.

G.6.1.2.5 Analysis of Results

Reasonable measures should be implemented to prevent the occurrence of unlikely accidents. The applicable review criteria are in NUREG-1200, specifically, SRP 6.1.4 "Radionuclide Release – Accidents or Unusual Operational Conditions" and SRP 6.1.6 "Assessment of Impacts and Regulatory Compliance." The results of the accident analysis should provide insights about the types of accidents that may present project-specific risks that need to be further constrained. Because risk is the product of probability and consequence, it is necessary to define reasonable constraints on these factors to manage the overall risk. An example approach acceptable to the NRC staff to assess GTCC waste operations is to define the lower bound of the unlikely probability range and an upper bound on the potential dose consequence. This provides a construct for evaluating accident analysis consequences to determine whether additional controls are needed to limit either the accident probability or consequence (and therefore limit the risk). Licensees may propose other thresholds or analysis approaches with adequate justification.

Consistent with NUREG-1200, SRP 6.1.4, a licensee may utilize their operational procedures such as waste acceptance, facility design, and operational evaluations and planning in identifying and precluding representative scenarios. The impact from some scenarios may be bound by the consequences of other assessed scenarios. The licensee may eliminate from further consideration scenarios for which the impacts are bounded by assessed scenarios.

Protection of the public during operations is provided in the performance objective 10 CFR 61.43. An operational safety assessment must be completed as required by 10 CFR 61.13(c). Occupational dose limits are set at 50 mSv (5 rem) *total effective dose equivalent* (TEDE) in 10 CFR 20.1201. The public dose limit is set at 1 mSv (0.1 rem) in 10 CFR 20.1301 or at 5 mSv (0.5 rem) for special circumstances, under 10 CFR 20.1301(c). An assessment of expected exposures from operational accidents involving GTCC waste should demonstrate that until cessation of operations, no individual located on, or beyond, any point on the boundary of the site will receive, because of any single event sequence, a probability-weighted TEDE that exceeds the relevant limits in 10 CFR Part 20. Expected event sequences (sometimes referred to as operational occurrences) should not be probability-weighted whereas the assessment of unlikely event sequences may use probability-weighting. An accident is unlikely if it has less than one chance of occurring during the operational period of the facility and more than one chance in 1,000 of occurring in any year within the period of operations. The results of the accident analysis should be evaluated to determine whether the proposed engineered controls, administrative controls, or both provide reasonable assurance the accident is less probable than unlikely, or its probability-weighted consequences are less than the limits established in 10 CFR Part 20.

The licensee may propose a range of facility design, waste acceptance criteria (e.g., wasteform, inventory controls), operational procedures or engineered controls to mitigate either the probability or consequence of accident sequences. As the unweighted consequence potential rises, it would be expected that mitigation measures would rely on an appropriate mix of all four instead of only operational procedures.

G.6.1.3 Fire Protection

The objectives for fire prevention at LLW disposal facilities are similar to those for other non-radioactive waste processing facilities. However, there is an increased caution with fires involving LLW because radioactive particles can be entrained and dispersed in the atmosphere resulting in contamination. The purpose of this section is to establish a general understanding of fire protection and provide guidance on fire protection during the operational phase of LLW disposal facilities (e.g., handling, treatment, processing, storage, disposal).

In fire protection analyses, postulating fire events is an important component in determining the acceptable and reasonable systems, controls, and practices to protect from accidental fire events. In these analyses, the postulated fire events may be assumed to occur in all important areas of the facility, which include, but are not limited to, waste receipt, handling, and storage areas. Additional considerations when assessing acceptable and reasonable fire protection systems, controls, and practices include determining whether there is appropriate fire protection equipment and evaluating the facility's emergency response plan. NUREG-1200 describes fire protection considerations. A goal of this guidance is to discuss certain concepts and principles that may impact the development of postulated fire scenarios.

Treatment and stabilization of LLW usually occurs prior to receipt of waste at a disposal facility but may in some cases be performed at a disposal facility before disposal. Treatment and stabilization typically involve handling and movement of waste, including the introduction of energy and fuel sources. Some stabilization processes are thermal and, therefore, may operate at high temperatures. Licensees should be familiar with additional relevant documents such as "Technical Position on Waste Form (Revision 1)," dated January 24, 1991 (NRC, 1991), and National Fire Protection Association (NFPA) 801, "Standard for Fire Protection for Facilities Handling Radioactive Materials" (NFPA, 2020). NFPA 801 provides fire protection guidance for waste handling and storage facilities. Relevant chapters found in NFPA 801, regarding waste processing and treatment include Chapter 7, "Facilities, Processes, and Special Hazards."

G.6.1.3.1 General Concepts of Fire Protection

Fire is a chemical reaction involving solids, liquids, and gases that ignite and undergo a rapid, self-sustaining oxidation process, accompanied by the evolution of heat and light of varying intensities. However, the chemical and physical reactions that take place during a fire are extremely complex and often difficult to describe completely. The most common fires start as a result of the ignition of solid or liquid fuels (combustible materials). They typically become volatile and serve as suppliers of gaseous fuel to support combustion.

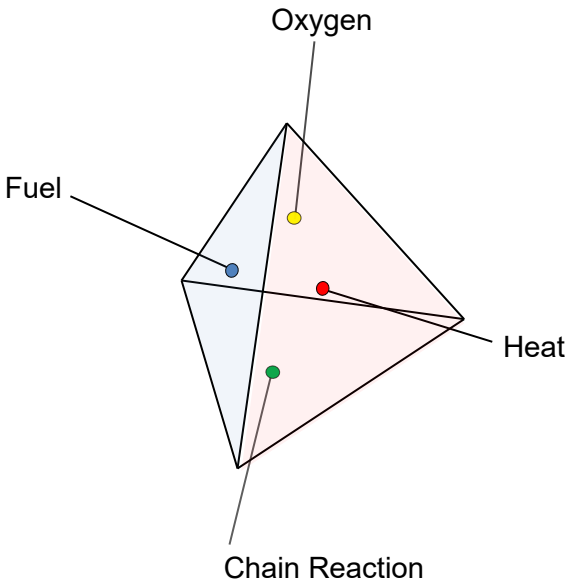


Figure G-3 Conditions of the Fire Tetrahedron

The process of fire development begins when the fuel surface starts to heat up because of heat transfer from the adjacent surroundings. As the temperature increases in response to this heat input, the fuel surface begins to emit fuel vapors. The fuel vapors mix (by convection and diffusion) with oxygen in the adjacent boundary layer, ignite (through a chemical reaction), and release additional heat. Some of this liberated heat energy may further increase the surface temperature of the fuel and thereby accelerate the fire growth process. To sustain combustion, four conditions of the fire tetrahedron need to be met: oxygen, fuel, heat, and an uninterrupted chain reaction (shown in Figure G-3), as follows:

- Oxygen: Oxygen is required to mix with the fuel vapors. Fuel vapors mix with oxygen and ignite from surrounding heat to produce a visible flame.

- Fuel: Fuel is required to sustain the combustion process by providing the fuel vapors used in ignition.

- Heat: Heat is required to initially heat the fuel component to its ignition temperature. The ignition temperature is the temperature at which the fuel ignites. Heating of the fuel produces fuel vapors that then mix with oxygen to cause a visible flame.

- Uninterrupted Chain Reaction: An uninterrupted chain reaction is required to sustain the combustion process. For example, some fire suppression agents, such as Halon, work by disrupting the chain reaction.

If any one of these conditions is not met, a fire cannot be sustained. This concept is important in fire prevention. Since it is impractical to remove oxygen from a given space without using expensive or complex systems, or to interrupt the chain reaction, many fire prevention practices revolve around fuel and heat management. On the other hand, some fire suppression systems will eliminate oxygen and interrupt the chain reaction, such as carbon dioxide and Halon-based systems. If an adjacent heat source can be prevented or removed from an area of combustible

material, the likelihood of ignition significantly decreases. The same logic can be applied if fuel is removed. For example, if there is no fuel or limited fuel, a fire cannot start or be sustained even if there are external heat sources. NUREG-1805, "Fire Dynamics Tools (FDTs) Quantitative Fire Hazard Analysis Methods for the U.S. Nuclear Regulatory Commission Fire Protection Inspection Program," issued December 2004 (NRC, 2004), further explains fire protection concepts and calculations.

G.6.1.3.2 Housekeeping Practices

Fuel management is an important concept for fire prevention because without sufficient fuel, a fire cannot occur or be sustained. One way to manage unwanted fuel is through good housekeeping practices. As such, good housekeeping practices are not only essential in preventing contamination but are also an important aspect of fire prevention and protection. For example, good housekeeping practices minimize fire spread, preserve clear escape paths, and maintain the ability to access areas to fight a fire if one were to occur. Good housekeeping practices can be defined as procedures, inspections, and management controls that contribute to the organizational health of the facility. For example, good housekeeping relates to keeping everything in its place, properly labeling materials and containers, and ensuring processing areas are free of accumulated materials. Poor housekeeping practices contribute to the increased potential and severity of a fire or explosion in several ways:

- greater quantity of combustible fuel for the initial fire to consume, which in turn leads to a larger initial fire
- greater quantities of material around the initial fire area, which allow the fire to spread more easily
- potential for flash fires or dust explosions to occur if layers of lint or dust accumulate
- increased potential for spontaneous ignition

Proper layout and storage of materials is an essential aspect of good housekeeping because it helps minimize the quantities of materials and overcrowding of the work area. Overcrowding is a major challenge because it restricts aisle usage, can slow egress in an event of an evacuation, and can be detrimental to fire protection equipment such as sprinklers, fire doors, fire extinguishers, and ease of access by the fire brigade. Additionally, overcrowding of materials leads to a larger fire load in the area and an increased likelihood of the fire spreading to other areas of the facility. Proper labeling is an important aspect of storage to prevent contamination. Labeling materials is important in identifying whether materials are combustible or not, which aids in preventing fires and the possible dispersion of radioactive particles due to fires.

Proper housekeeping can prevent the buildup of dust and lint. This section summarizes the explosive hazards of dust buildup and the factors of a severe explosion, as discussed in Schwab (2008) and Cashdollar (2000). Dust buildup on system components can create a fire or explosion hazard. Some believe that a depth as little as 0.8 mm (0.08 in) of finely divided dust can lead to a flash fire (Schwab, 2008). Relative reactivity of a material can be estimated by the peak explosion pressure, or the maximum rate of pressure rise (Cashdollar, 2000). For example, Figure G-4 shows that aluminum has the highest reactivity, followed by magnesium, polyethylene, and coal dust. Typically, fine metal particles are reactive because of their ability

to oxidize. However, organic materials, such as wood shavings, wood flour, and cornstarch, can also present an explosive hazard. Any fine particles with reactivity can represent an explosion hazard. An explosibility index is a way to measure and rate the relative explosive hazards of different dusts (Schwab, 2008). A larger index results in a more severe type of explosion. An example of various types of dust are listed in Table G-14. As shown, cornstarch dust can pose as much of an explosive hazard as magnesium or aluminum.

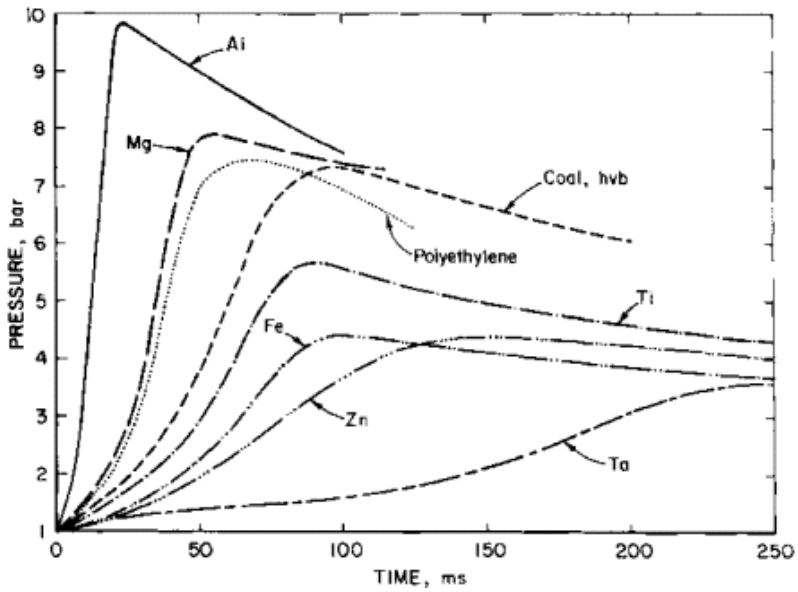


Figure G-4 Peak Explosion Pressures of Different Material Types (Cashdollar, 2000)

Reactivity is not only influenced by the intrinsic properties of the material, such as how well it can oxidize, but also the specific particle size of the dust, which influences the pressure of the explosion. At lower particle sizes, the specific unit surface area per particle increases. A larger surface area has an effect on the rate of pressure rise, which influences the size of the explosion. Additionally, dislodged dust particles in the air allow a dust cloud to occur, which can then propagate into an area-wide explosion. Therefore, proper control and routine cleaning of fine particles can help mitigate the hazard of a flash fire or explosion. Generally, facilities are equipped with dust collection equipment to pick up dust at collected areas, such as in areas containing processing machinery, and convey them to dust collectors outside the building. However, the dust collection equipment should be inspected to ensure it is working properly and that there is no collection of dust in unwanted areas. Failure to ensure proper maintenance of equipment can lead to dust collecting within the equipment and to an increased occurrence of a fire or explosion.

G.6.1.3.3 Controlling Spontaneous Ignition

Spontaneous ignition can be minimized by understanding and practicing good housekeeping that revolves around controlling ignition sources. The following information summarizes the different types of common ignition sources found in LLW disposal facilities, with further expansion of detail found in Mattern (2008).

Common ignition sources include, but are not limited to, the following:

- friction
- hot work
- static electricity

Friction is a common source for ignition because it is prevalent in many processing and industrial facilities, even those that attempt to eliminate combustion. Preventive maintenance programs can identify and eliminate potential sources of friction. In areas where there is a potential for friction, it is important to implement good housekeeping practices to ensure that debris and transient combustibles do not act as a source of fuel for a potential fire. For example, a common occurrence is friction induced by a track hoe, which can spark nearby debris.

Table G-14 Types of Dusts and their Properties that May Represent an Explosion Hazard (Schwab, 2008)

Type of Dust	*Explosibility Index*	*Ignition Sensitivity*	*Explosion Severity*	*Maximum Explosion Pressure (psig)*[a]	*Maximum Rate of Pressure Rise psi/sec*[a]	Ignition Temperature[b] *Cloud (°C)*	Ignition Temperature[b] *Layer (°C)*	*Minimum Cloud Ignition Energy (joules)*	*Minimum Explosion Conc. (oz/ft³)*[c]	*Limiting Oxygen Percentage*[d] *(Spark Ignition)*
Agricultural Dusts										
Cellulose, alpha	>10	2.7	4.0	117	8,000	410	300	0.040	0.045	—
Cornstarch commercial product	9.5	2.8	3.4	106	7,500	400	—	0.04	0.045	—
Lycopodium	16.4	4.2	3.9	75	3,100	480	310	0.04	0.025	C13
Wheat starch, edible	17.7	5.2	3.4	100	6,500	430	—	0.025	0.045	C12
Wood flour, white pine	9.9	3.1	3.2	113	5,500	470	260	0.040	0.035	—
Carbonaceous Dusts										
Charcoal, hardwood mixture	1.3	1.4	0.9	83	1,300	530	180	0.020	0.140	—
Coal, Pennsylvania, Pittsburgh (experimental mine coal)	1.0	1.0	1.0	90	2,300	610	170	0.060	0.055	—
Chemicals										
Adipic acid	1.9	1.7	1.1	84	2,700	550	—	0.060	0.035	—
Bis-phenol A	>10	11.8	2.8	89	8,500	570	—	0.015	0.020	C12
Phthalic anhydride	6.9	11.9	1.4	72	4,200	650	—	0.015	0.015	C14
Stearic acid, aluminum salt (aluminum tristearate)	>10	21.3	1.9	87	6,300	420	440	0.015	0.015	—
Sulfur	>10	20.4	1.2	78	4,700	190	220	0.015	0.035	C12
Drugs										
Aspirin, acetylsalicylic acid, $C_9H_8O_4$	>10	2.4	4.3	88	>10,000	660	Melts	0.025	0.050	—
Vitamin C, ascorbic acid, $C_6H_8O_6$	2.2	1.0	2.2	88	4,800	460	280	0.060	0.070	C15, N12
Metals										
Aluminum flake, A 422 extra fine lining, polished	>10	7.3	10.2	127	20,000+	610	326	0.010	0.045	—
Magnesium, milled, Grade B	>10	3.0	7.4	116	15,000	560	430	0.040	0.030	—
Alloys and Compounds										
Aluminum-cobalt alloy (60–40)	0.4	0.1	3.5	92	11,000	950	570	0.100	0.180	—

The act of cutting, whether through saw or torch, can produce sparks that can act as a possible ignition source. NFPA 51B, “Standard for Fire Prevention During Welding, Cutting, and Other Hot Work” (NFPA, 2019), states that transient combustibles need to be distanced from any type of hot work. This distance extends spherically from the point of the hot work and not just vertically or horizontally. Often, a fire watch must be present during hot work activities. A fire watch is a trained individual(s) responsible for continuously observing the hot work area for signs of a fire, maintaining fire-safe conditions, and responding to emergencies during hot work. Figure G-5 shows the main components in implementing a fire watch for hot work activities (NFPA, 2019).

Static can be produced by dissimilar materials moving past each other, one or both of which are usually a poor conductor of electricity. For static electricity to be a source of ignition, four criteria must be met (Scarbrough, 2008):

(1) an effective means for separating charge

(2) a means of accumulating and maintaining the separated charge, such that there is a suitable difference in electrical potential between them

(3) a release or discharge of adequate energy

(4) a discharge that takes place in an ignitable mixture

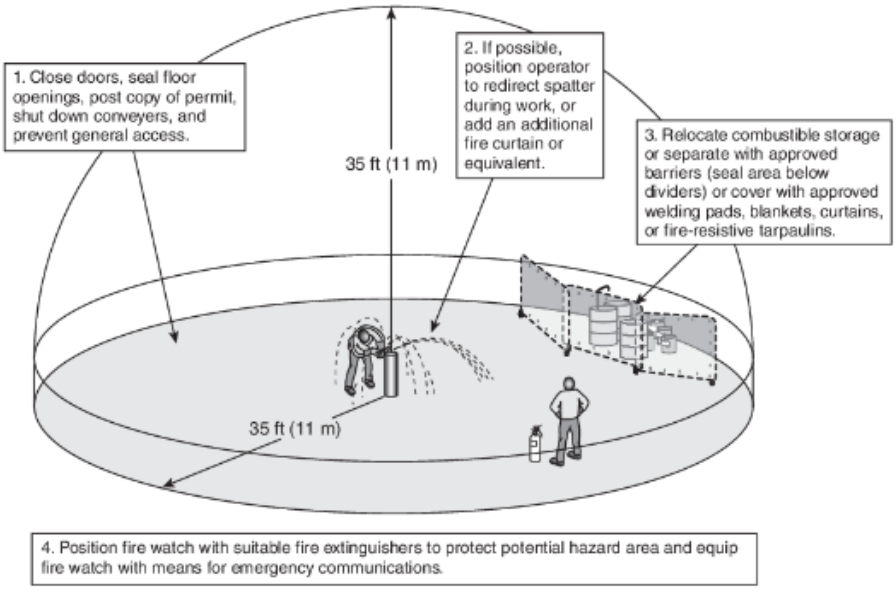


Figure G-5 Main Components to Implementing a Fire Watch (NFPA, 2018)

Liquid or dust moving through a pipe can produce an electrical potential. If a static discharge occurs, it can ignite the fluid or dust. Static charge can be generated when liquids with low electrical conductivity move in contact with other materials, such as flammable liquid dispensed from a container. This also occurs commonly in liquid flows through pipes. Static can accumulate on the body of liquid hydrocarbons if it is flowing quickly through nonconductive materials such as resin-bonded media. Therefore, it is important that the dispensing object,

receiving object, and ground device are all interconnected. Otherwise, a static discharge could result in the ignition of flammable liquid. There are instances where it may be impractical or impossible to completely ground all interconnected parties to prevent static discharge. Therefore, other options, such as inerting the environment, may be more practical.

G.6.1.3.4 Metal Fires

Combustible metals are types of metals that are either combustible or explosive. Certain metals such as calcium, lithium, magnesium, plutonium, uranium, zinc, zirconium, thorium, and titanium, are referred to as combustible metals because of their tendency to easily ignite as thin sections or as fine particles. Specific surface area is a controlling parameter in the propensity of certain metals to combust. This is important because in LLW disposal facilities, small quantities of these metals may be present or may be produced during processing or stabilization. Therefore, understanding the general combustible properties of these metals is important in proper handling and fire prevention. NFPA 484, “Standard for Combustible Metals” (NFPA, 2022), includes in-depth guidance on combustible metals, with each metal having its own chapter. This section covers some general precautions and guidance specifically related to LLW disposal facilities. The following information is further explained in Christman (2008).

Examples of combustible metals used in the nuclear field are uranium, zirconium, and plutonium. However, some metals, such as iron and steel, although not normally combustible, can ignite and burn under the right conditions. For example, fine steel wool can be easily ignited. It is important to consider the particle size, shape, quantity, and alloy when determining combustibility hazard. Clouds of most metal dusts in air are explosive. Therefore, it is important that dust be properly controlled to prevent a fire or explosion. Good housekeeping practices, as discussed above, are essential to mitigating the hazards from metal particles. The housekeeping practices must be used regularly and at an appropriate frequency. If a significant quantity of metal particles has accumulated, the housekeeping activity itself may initiate a metal fire.

Alloy metals may react differently depending on the composition of the alloy. Some alloys may burn hot with little smoke, some may burn with high temperatures, some can sustain burning in low-oxygen environments, and some may react violently in contact with other materials. Because of the wide properties in burning metals, it is imperative that the correct extinguishing agent be used if a fire were to occur. For example, normal extinguishing agents that would be used for ordinary Class A combustibles or flammable liquids may react violently with burning metals. Additionally, burning certain metals, such as thorium, uranium, and plutonium, adds a layer of complexity when firefighting because of the potential for radiological exposure.

Extinguishing a metal fire is a difficult task. The basic principle of extinguishing a metal fire is to completely cover the surface of the burning metal with the appropriate agent until the temperature of the metal cools. Portable Class D fire extinguishers are designed specifically for small metal fires; however, they have a limited quantity of the extinguishing agent. Thus, the amount of burning metal needs to be significantly less than the agent to effectively extinguish the fire. Additionally, applying water to a metal fire will most likely result in violent reactions. The addition of water leads to an exothermic reaction and the production of hydrogen and oxygen, which could lead to an explosion.

Considerations for different metals include the following:

- Sodium: Sodium has only been reported to spontaneously ignite when it is finely divided, such as in a powder form. The fire hazard associated with sodium is that it is less dense than water. Additionally, sodium burns vigorously and reacts with water. Hydrogen may be a byproduct of this reaction and may ignite from the heat generated by the chemical reaction. Similar to other metal fires, sodium reacts violently with halogens, halogenated hydrocarbons, and acids.

- Potassium: Potassium has similar properties to sodium, except potassium reacts more violently with halogens than sodium does. For example, a reaction with bromine may result in a detonation. There can also be an explosive reaction with potassium and sulfuric acid. Additionally, the byproducts of potassium combustion are peroxides that may react violently with organic materials.

- Sodium-Potassium Alloys: Sodium-potassium (NaK) alloys possess the same fire hazards as discussed for sodium and potassium above. Under pressure, NaK leaks have ignited spontaneously.

- Lithium: Lithium also reacts with water to form hydrogen. However, unlike for sodium, the reaction does not generate sufficient heat to ignite the hydrogen. Therefore, lithium-water reactions are typically less violent. The hazard with a lithium-water reaction is the buildup of hydrogen. In poorly ventilated areas, hydrogen could build up and be ignited from an external ignition source, which would lead to a hydrogen explosion. Therefore, proper ventilation should be ensured when handling lithium.

Alkali metals should be stored in dry areas with proper ventilation. Because of their high reactivity with water, alkali metals should be stored in rooms exclusively for alkali metals. Since sprinkler protection of alkali metals would be undesirable, other combustible materials should not be stored in the same area as alkali metals. Additionally, if large quantities of these metals are stored outdoors, there should be sufficient weatherproof enclosures to prevent water from entering and reacting with the alkali metals. Special dry powders were developed to fight alkali metal fires. These special powders usually consist of dry sand, dry sodium chloride, dry lithium chloride, or dry soda ash. These powders line the surface of the burning material and blanket the fire until the ignition temperature of the metal decreases.

The hazards associated with radioactive metals are contamination and radiologic exposure to workers or the fire brigade (responders) and, in some cases, transport of radioactivity to offsite members of the public. The smoke from fires can transport radioactive particles to the surrounding area, as well as allow these radioactive particles to be inhaled. Uranium, thorium, and plutonium are easily ignited in the form of chips, turnings, or fines. The fines generate heat and with the addition of water to cool them, also slowly produce hydrogen. The ratio of fines to water should be appropriate, such that the amount of heat generated is not able to boil off the water and ignite any of the remaining surrounding fines (Christman, 2008). In most cases, because oxidization is also an issue, these materials are stored in an inert environment or in a non-oxidizing substance. Radioactive metals should be adequately controlled and stored (e.g., temperature, pressure) in an environment where the intrinsic properties of the stored fines combined with the storage conditions prevent a fire from occurring, which may be important for some wastes from reprocessing of spent nuclear fuel.

G.6.1.3.5 Historical Fire Events

This section describes fire events that involve radioactive waste and examines what general topics and principles were missed that ultimately led to a fire event.

Beatty Nevada Fire Event

A fire occurred at the Beatty LLW disposal facility in Nevada (Nevada Department of Public Safety, 2015). The facility sits on State-owned land that had disposed of LLW from 1962 to 1992. The facility has been closed since 1992. The LLW disposal facility is located adjacent to an active Resource Conservation and Recovery Act (RCRA) hazardous waste site. Multiple efforts were made to distinguish the area between the two sites, including both digital and physical barriers. The LLW disposal facility was located within a segregated and fenced area. The site had a unique EPA identification number to differentiate it from the RCRA facility. Figure G-6 is an aerial snapshot of the area.

The event occurred on Sunday, October 18, 2015. Small popping sounds, similar to fireworks, were reported. A few minutes later, a section of the ground within the LLW disposal site erupted with flames, smoke, and flying debris. The duty officer of the radiation control program began contacting and notifying appropriate staff and agencies, including local, State, and Federal partners. There was no attempt to extinguish the fire. Given the remote location of the site, the decision was made to allow the fire to self-extinguish several hours after the incident. After the fire went out, an emergency review of monitoring data determined that there was no detectable radiation two times above background.

Independent parties conducted a detailed review, which concluded that water entered a disposal trench through observed cracks and subsidence in the landfill cover cap. Additionally, at least two corroded drums of sodium metal were disposed at this trench location.[5] The water likely reacted with the sodium metal to produce hydrogen gas and sodium hydroxide. This resulted in a violent metal fire that then self-extinguished. It is speculated that a greater than normal amount of precipitation likely prompted the event.

From the standpoint of technical guidance for fire protection, a few key factors may have prevented this fire from occurring. The combination of sodium and water, which should not be mixed, produced hydrogen gas and resulted in a metal fire. Current LLW disposal regulations include restrictions that prevent the disposal of reactive or flammable materials. Although an engineered cover was placed on top of the trench, it ultimately degraded due to surficial geologic processes. The facility was placed into operation before the NRC issued its LLW disposal regulations (10 CFR Part 61). Therefore, modern regulations should prevent this from happening. However, this case further emphasizes the importance of housekeeping practices, such as proper layout and storage, as well as provides context for sites that may have operated in a similar manner and time. Maintenance of historical records, including a description of waste disposed, may be important to help respond to emergency events.

[5] Sodium metal is reactive and does not meet the waste characteristics requirements found in 10 CFR 61.56



Figure G-6 Aerial View of the Beatty LLW Disposal Facility

Idaho National Laboratory Overpressurized Drum Event

This incident occurred at the INL site. A thermal event led to the energetic release of radioactive material from four 55 gal drums of suspect TRU waste (DOE, 2019b). There was no detectable release to the environment. During the initial response, firefighters observed no flames but saw smoke emanating from the top of a drum and the drum's lid was off. The firefighters were unable to extinguish the hot spots of the affected drum and set up 24-hour surveillance, fire watch, radiological surveys, and ventilation monitoring. Eight days later, workers entered the facility and discovered four drums that had experienced exothermic chemical reactions, which resulted in the ejection of their lids and part of their contents.

Further technical analysis of the event led to the identification of two related mechanisms:

(1) Oxidation of pyrophoric non-roster-oxide depleted uranium metal powder supported a secondary reaction.

(2) The secondary reaction was the generation of substantial amounts of methane by hydrolyses of beryllium carbide.

The identified direct cause was oxidation reactions that occurred during drum waste repackaging, resulting in the breach of four suspect TRU waste drums. The report also provided a root cause analysis (DOE, 2019). The investigation identified two root causes and eight contributing causes.

The two root causes were as follows:

(1) Management failed to fully understand, characterize, establish, and implement adequate process control for undocumented origin or process information.

(2) Management failed to continue to develop the safety culture over number of years.

The event can be seen as having a technical cause driven by root cause(s). The technical cause involves the actual chemical reaction that took place. The root causes were the underlying causes that allowed these events to occur in the first place.

Waste Isolation Pilot Plant Salt Haul Truck Fire

The WIPP is a DOE research and development facility located in New Mexico that provides an underground repository for TRU waste. The DOE is responsible for demonstrating the safe and permanent disposal of defense TRU waste. Disposal operations began in 1999. The underground repository is located 650 m (2,150 ft) underground and has four main shafts that connect the underground (waste disposal and construction areas) to the surface. On February 5, 2014, an underground mine fire involving a salt haul truck occurred at the WIPP facility at the location shown in Figure G-7. After the event, an accident investigations board issued a report to identify the direct, root, and contributing causes of this event (DOE, 2014). The full report includes a variety of analyses that were performed.

The report includes an investigation of the salt truck and reviews the fire hazard analysis (FHA) of the facility. The board determined that maintenance of the salt truck was a contributing cause of the event and that there was a significant difference in the preventive maintenance prescribed in the service manual and what the facility performed. Additionally, the FHA of the underground site did not comprehensively address all credible underground fire scenarios. It did not consider various properties of the site, such as smoke movement through ventilation, various unique hazards underground, and other key details expected in a comprehensive FHA. Therefore, limited analyses and development of the fire program requirements resulted in a lack of adequate information to ensure risk-informed decisions regarding the actual fire protection program. The events that led up to the WIPP salt truck fire occurred through a culmination of improper housekeeping procedures and inadequate fire protection analyses, as well as other contributing factors, as detailed in DOE (2014).

Los Alamos National Lab Radiological Release Event

A radiological event, involving an exothermic reaction occurred on February 14, 2014, at the Los Alamos National Lab (LANL) facility in New Mexico (DOE, 2015c). A drum mixture of organic materials and nitrate salts resulted in an exothermic reaction that caused pressurization of the drum, failure of the drum locking ring, and displacement of the drum lid. The energetic release also resulted in TRU waste being expelled onto adjacent containers. As a result, radiological release was detected through airflow monitoring and filtration was automatically initiated. However, it was reported that a small portion of the airflow bypassed the filters and was exhausted directly to the atmosphere. The accident report concluded that the direct cause was incompatible materials being stored in the same drum.

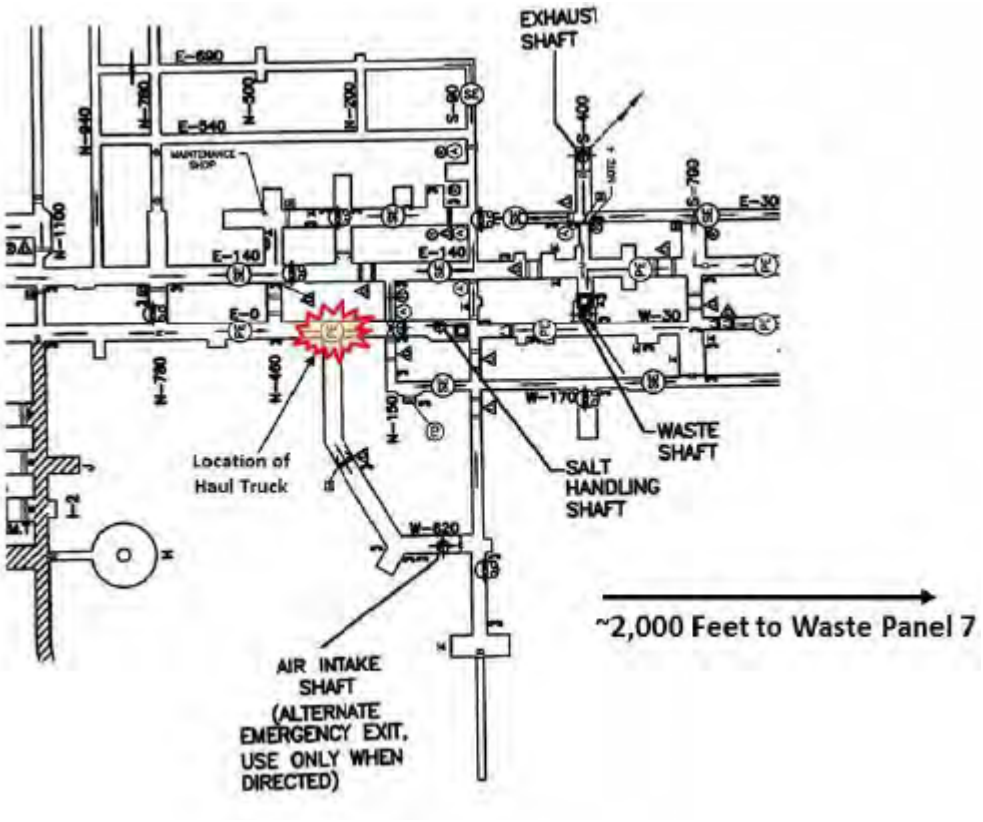


Figure G-7 Location of the Truck Fire in the WIPP Disposal Facility

The external reaction led to internal heating of the drum, which caused an internal pressure buildup of combustible gases. Once the pressure exceeded the drum's capability, the drum lid was displaced and the combustible gases and solids were ignited, which led to the spread of other combustible materials. The report concluded that "If LANL had adequately developed and implemented repackaging and treatment procedures that incorporated suitable hazard controls and included a rigorous review and approval process, the release would have been preventable" (DOE, 2015c). Proper waste characterization and waste acceptance criteria could also have prevented the accident.

Conclusions

Typically, fire events occur as a result of multiple direct and contributing causes. In most of the events reviewed in this section, the direct causes were single occurrences, but waste characteristics played a common role. Careful implementation of 10 CFR 61.56 waste characteristics requirements and waste acceptance criteria should eliminate most accident scenarios involving waste reactivity. When considering fire prevention, and the prevention of direct causal events that lead to fire events, both fundamental fire science concepts and good housekeeping practices and safety culture should be considered, implemented, and maintained.

G.6.2 Physical Protection During Operations

GTCC waste streams are likely to be quite varied, based on the DOE's GTCC FEIS and potential wastes from reprocessing activities, including the variety of wasteforms (e.g., sealed sources, activated metals, vitrified waste, and grouted waste). Some of the GTCC waste

streams contain SNM and other radionuclides (i.e., as specified in Appendix A, "Category 1 and Category 2 Radioactive Materials," to 10 CFR Part 37, "Physical Protection of Category 1 and Category 2 Quantities of Radioactive Material") in quantities that could require implementation of the physical protection measures specified in 10 CFR Part 37 and 10 CFR Part 73, "Physical Protection of Plants and Materials."

The regulations in 10 CFR Part 37 impose security requirements for the possession and use of Category 1 and Category 2 quantities of radioactive material. These regulations establish the objectives and minimum requirements that licensees must meet to protect against theft or diversion. These requirements are intended to protect the public against the unauthorized use of a Category 1 or Category 2 quantity of radioactive material by reducing the risk of theft or diversion of the material. The NRC considers Category 1 and Category 2 quantities of radioactive material to be risk significant, and these quantities refer specifically to 16 radioactive materials (14 single radionuclides and 2 combinations). These materials are americium-241, americium-241/beryllium, californium-252, curium-244, Co-60, Cs-137, gadolinium-153, iridium-192, plutonium (Pu)-238, Pu-239/beryllium, promethium-147, radium-226, selenium-75, Sr-90, thulium-170, and ytterbium-169. Table 1 of Appendix A to 10 CFR Part 37 provides the thresholds for these categories of radioactive material. The regulations in 10 CFR Part 73 impose security requirements for the possession and use of quantities of SNM of strategic significance.

The regulations at 10 CFR 61.16(a) require specific information regarding physical security. The physical security measures apply to radioactive waste prior to disposal and are dependent on the characteristics and quantities of waste present at the facility. The regulations in 10 CFR 61.16 state the following:

> **§ 61.16 Other Information**
>
> Depending upon the nature of the wastes to be disposed of, and the design and proposed operation of the land disposal facility, additional information may be requested by the Commission including the following:
>
> (a) Physical security measures, if appropriate.
>
> Any application to receive and possess special nuclear material in quantities subject to the requirements of part 73 of this chapter must demonstrate how the physical security requirements of part 73 will be met. In determining whether receipt and possession will be subject to the physical protection requirements of part 73, the licensee is not required to consider the quantity of special nuclear material that has been disposed of.

It is expected that the majority of radioactive waste requiring physical protection will be due to Category 1 and 2 quantities of radionuclides, under 10 CFR Part 37. This guidance is focused on physical protection requirements for waste that meets the exemption under 10 CFR 37.11(c). Additionally, an evaluation of the 10 CFR Part 73 requirements for SNM will need to be done to demonstrate compliance with 10 CFR Part 73. It is not expected that facilities will possess SNM in quantities greater than those of low strategic significance (prior to disposal). Significantly more protective measures could be required depending on the quantity of material present prior to disposal.

This section summarizes some key aspects of physical protection of waste that contains Category 1 and Category 2 quantities of radioactive materials that are exempted from most of the security requirements in 10 CFR Part 37. More detailed information and guidance can be found in NUREG-2155, "Implementation Guidance for 10 CFR Part 37 Physical Protection of Category 1 and Category 2 Quantities of Radioactive Material," Revision 2, issued March 2022 (NRC, 2022), and NUREG-2166, "Physical Security Best Practices for the Protection of Risk-Significant Material," issued May 2014 (NRC, 2014).

G.6.2.1 Physical Protection of Radioactive Waste

NRC regulations are intended to ensure protective measures are commensurate with the potential consequences of malevolent acts that include both theft and diversion. A principal concern of 10 CFR Part 37 is the potential theft or diversion of risk-significant quantities of radioactive materials that could lead to their use in a radiological dispersal device or a radiological exposure device. This arises, in part, due to the widespread use of radioactive materials in the United States by industrial, medical, and academic institutions.

The NRC has recognized the significant difference between certain radioactive waste of very low concentrations and radioactive materials that are highly concentrated or that do not need substantial work to be used for nefarious purposes. Regulations in 10 CFR Part 37 exempt certain radioactive waste from a majority of the 10 CFR Part 37 physical protections requirements:

> Paragraph (c) has been added to § 37.11 to address radioactive waste materials. The provision does require that some security measures be applied to the waste, but the majority of the 10 CFR part 37 requirements would not apply. Measures include the use of continuous physical barriers, alarmed locked gates or doors, and assessment and response of unauthorized entry."
> (78 FR 16942; March 19, 2013)

The exemption provided to radioactive waste at 10 CFR 37.11(c) is not applicable to all radioactive waste, as certain types of waste, such as sealed sources, ion-exchange resins, or activated metal waste of limited volume, have characteristics (e.g., higher concentration of certain radionuclides) that could be attractive for theft or diversion. Thus, 10 CFR 37.11(c) states, "[E]xcept that any radioactive waste that contains discrete sources, ion-exchange resins, or activated material that weighs less than 2,000 kg (4,409 lbs) is not exempt from the requirements of this part." The 2,000-kilogram (kg) (4,400-lb) threshold applies only to activated metals.

Licensees disposing of waste that requires physical protection under 10 CFR Part 37 will, at a minimum, need to implement the following requirements listed at 10 CFR 37.11(c) to secure the radioactive waste:

(1) Use continuous physical barriers that allow access to the radioactive waste only through established access control points;

(2) Use a locked door or gate with monitored alarm at the access control point;

(3) Assess and respond to each actual or attempted unauthorized access to determine whether an actual or attempted theft, sabotage, or diversion occurred; and

(4) Immediately notify the local law enforcement agency (LLEA) and request an armed response from the LLEA upon determination that there was an actual or attempted theft, sabotage, or diversion of the radioactive waste that contains category 1 or category 2 quantities of radioactive material.

A licensee's physical protection program will need to characterize the facility operations, identify the targets for theft and diversion (e.g., Category 1 or Category 2 quantities of radioactive material), and define the threat posed. Radioactive waste is a somewhat unique material from the perspective of physical protection in that, for most radioactive waste, the valuable radionuclides are typically removed to the maximum extent possible. In contrast, non-waste facilities using and possessing radioactive materials are typically concentrating them for a specific use. The physical protection needs to be commensurate with the threat. For example, 10 CFR 37.11(c) radioactive waste with diffuse Category 1 or Category 2 quantities, as opposed to discrete sources, is exempt from the requirements in 10 CFR Part 37, Subpart B, "Background Investigations and Access Control Program"; Subpart C, "Physical Protection Requirements During Use"; and Subpart D, "Physical Protection in Transit." These wastes, which contain items such as contaminated clothing, gloves, soil, or other waste with low specific activity, are instead subject to the requirements in 10 CFR 37.11(c)(1–4) in Subpart A, "General Provisions," of 10 CFR Part 37. NUREG-2155 and NUREG-2166 include further guidance for demonstrating compliance with 10 CFR Part 37. Information on compliance with 10 CFR Part 73 requirements can be found at Regulatory Guide 5.59, "Standard Format and Content for a Licensee Physical Security Plan for the Protection of Special Nuclear Material of Moderate or Low Strategic Significance," Revision 1, issued February 1983 (NRC, 1983a).

G.6.2.1.1 Continuous Physical Barrier and Access Control

The regulations at 10 CFR 37.11(c)(1) require the use of continuous physical barriers that allow access to the radioactive waste only through established access control points. A physical barrier is a natural or man-made structure or formation sufficient for the isolation of Category 1 or Category 2 quantities of radioactive material. A continuous physical barrier provides for the isolation of Category 1 and Category 2 quantities of radioactive materials and allows access to the security zone only through established access control points.

Access control plays an important part in securing risk-significant radioactive material and should involve the use of the appropriate physical access controls and adherence to administrative processes (e.g., the organization's security policies and procedures). The physical access controls should regulate the movement of people or material into or out of the secured area and should detect the actual or attempted access of unauthorized personnel. Licensees should be alert for covert actions in preparation for an attempted theft, sabotage, or diversion. Such covert actions could include, for example, the theft and copying of keys to locked security zones or the gaining of access to security codes for copying such codes onto duplicate key cards or for turning off the alarm system. Therefore, licensees should monitor, assess, and respond to actual or attempted unauthorized access to keys, security cards, codes (e.g., access and alarm codes), or other means that could be used to gain unauthorized access to the risk-significant radioactive material.

NUREG-2155 and Section 4.1.1 and Section 4.1.5 in NUREG-2166 contain further information.

G.6.2.1.2 Locks and Monitoring at Access Control Points

The regulations at 10 CFR 37.11(c)(2) require the use of a locked door or gate with monitored alarm at the access control point. Locks and locking systems are important elements for securing risk-significant radioactive material. Locks are generally used to assist in controlling access to areas, facilities, and materials through doors, gates, container lids, and similar material or personnel access points. Although some locks are difficult to pick or manipulate, no lock can claim to be "manipulation proof." Therefore, the licensee should use locks in conjunction with other security measures, such as intrusion detection devices. In all applications, the licensee should ensure that the lock delay time and capability closely match the penetration resistance of the rest of the secured barrier (e.g., balanced protection). For example, the lock should not be significantly stronger or weaker than the attached door, wall, or fence.

The risk-significant best security practice is to avoid relying on keys to control access to the security zone. Key control becomes more difficult as time passes (especially for large institutions) as more duplicate keys are made for newly approved individuals. The same is true for the control of lock combinations over time. The licensee should, instead, use key cards, cipher locks, or some other type of electronic access control device, where feasible. With these devices, the licensee can delete the access code after an individual no longer requires access to the secured area or when a key card is lost or stolen.

To meet the requirement in 10 CFR 37.11(c)(2) for a monitored alarm at the access control point, the licensee should establish and maintain the capability to continuously monitor and detect without delay all unauthorized entries at the access control point. The intrusion detection system is an integral part of the physical protection program. Intrusion detection is defined as the detection of a person or vehicle attempting to gain unauthorized access into an area that is being protected. Detection of an intruder should take place as early as possible (before or at the delay barriers) to allow the necessary time to stop the adversary from achieving his or her final objective (e.g., theft or diversion of Category 1 or Category 2 quantities of radioactive material). Generally, intrusion detection uses sensors that, once activated, send a signal to the monitoring station, or activates an alarm, or both.

NUREG-2155 and Section 4.1.4 and Section 4.2 in NUREG-2166 contain further information.

G.6.2.1.3 Assessment of Unauthorized Access

Following detection of an actual or attempted unauthorized access, the licensee must assess whether an actual or attempted theft, sabotage, or diversion occurred (10 CFR 37.11(c)(3)). Assessment, which completes the detection function, is a critical component of detection and is equally important to the initiation of response. Assessment provides a means to determine the necessary actions (responses) needed to mitigate situations that pose a challenge to physical security. The key element within the assessment process is identification. Identification assists the response force in selecting appropriate responses to address security-related and potential threat situations resulting from the detected activity. Identification also provides the ability to determine the absence of a threat resulting from detected activity, such as a nuisance alarm caused by wildlife or debris. Assessment techniques should also identify the stimulus that caused the alarm quickly before it was no longer present. The identification of the stimulus

enables the initiation of timely response consistent with the goals and objectives of the physical protection program and protective strategy. Video camera coverage of each sensor zone or direct visual checks can provide alarm assessments. Performing alarm assessments using video is considered a best practice for assessing alarms.

NUREG-2155 and Section 4.3 in NUREG-2166 contain further information.

G.6.2.1.4 Response and Communications with Local Law Enforcement Agencies

Responding to an assessed alarm is the final function of the physical protection program. The regulations at 10 CFR 37.11(c)(4) require licensees to immediately notify and request an armed response from the LLEA upon determination that there was an actual or attempted theft, sabotage, or diversion of the radioactive waste that contains Category 1 or Category 2 quantities of radioactive material. In 10 CFR 37.5, "Definitions," the NRC defines an LLEA as a public or private organization that has been approved by a Federal, State, or local government to carry firearms and make arrests and that is authorized and has the capability to provide an armed response in the jurisdiction in which the licensed Category 1 or Category 2 quantity of radioactive material is used, stored, or transported.

The response force or LLEA could be located onsite or offsite. However, if the LLEA is offsite, the licensee may need better detection and delay systems to account for the additional time necessary for an LLEA response.

NUREG-2155 and Section 4.3.2 in NUREG-2166 contain further information.

G.6.2.2 Exemption from Physical Protection Requirements in 10 CFR Part 73

The regulations in 10 CFR Part 73 provide an exemption from the physical protection requirements at 10 CFR 73.67, "Licensee fixed site and in-transit requirements for the physical protection of special nuclear material of moderate and low strategic significance," for radioactive waste containing SNM of low strategic significance licensed under 10 CFR Part 61 that meets certain characteristics that limit its attractiveness as a target for theft and diversion. In particular, 10 CFR 73.67(b)(1)(iv) requires the radioactive waste containing SNM is (a) to be in quantities of low strategic significance prior to disposal, (b) not mechanically separable from the other radioactive waste material, and (c) in a concentration of SNM that is no more than 0.01 percent of the mass of the other waste material.

Manifest information will be critical for ensuring that the quantity of SNM present at the site, prior to disposal, is in quantities of low strategic significance and represents waste containing SNM that is not readily separable.[6] A licensee will need to estimate the mass (in grams) of SNM in a waste container and consider sources of uncertainty in determining the mass present at the facility prior to disposal. More importantly, the shipping licensee's manifest information should be very clear with respect to the waste material source and characteristics to ensure the disposal site operator can verify that the SNM is not mechanically separable.

[6] "Readily separable" means chemical or mechanical separation of SNM from non-SNM waste using commercially available processes. Readily separable does not include the development of new specialized processes for the separation of SNM.

G.6.2.3 Physical Protection after Disposal

The physical protection requirements at a 10 CFR Part 61 facility are applicable to the radioactive waste prior to disposal. Once a waste container has been emplaced in a disposal unit (e.g., concrete vault, trench), it is considered disposed; however, a considerable amount of time can pass before final closure activities are completed at a facility. A licensee that possesses radioactive waste that is required to have physical protection, prior to disposal, will specify a security zone for the use and possession of the radioactive material. It is expected that, prior to final closure of a disposal unit, the licensee will include disposal units containing physically protected waste material within a security zone with characteristics comparable to those used to protect the material prior to disposal. After emplacement of the waste but prior to closure, the licensee may employ certain design measures (e.g., temporary or initial cover designs, sealing of concrete vaults) that will increase the physical protection of the waste and that could be considered in the security plan for the emplaced waste.

G.6.3 Criticality Controls

GTCC waste streams are quite varied (DOE, 2016a,b); some may contain SNM in significant concentrations of fissile material. These GTCC wastes can present a concern associated with nuclear criticality during handling and after disposal. The regulation at 10 CFR 61.16(b) requires the prevention of a criticality accident during operations:

> (1) Any application to receive and possess special nuclear material in quantities that would be subject to the requirements of §70.24, "Criticality accident requirements" of this chapter must demonstrate how the requirements of that section will be met, unless the applicant requests an exemption pursuant to §70.24(d) of this chapter. In determining whether receipt and possession would be subject to the requirements of §70.24, the applicant is not required to consider the quantity of special nuclear material that has been disposed of or radioactive waste containing fissile material that meets the exemption requirements specified at §71.15(c) of this chapter.
>
> (2) Any application to receive and possess special nuclear material must describe proposed procedures for avoiding accidental criticality, which address both storage of special nuclear material prior to disposal and waste emplacement for disposal. The procedure is not required to address radioactive waste containing fissile material that meets the exemption requirements specified at §71.15(c) of this chapter.
>
> (3) Any application to dispose of Greater-Than-Class C radioactive waste containing special nuclear material in quantities that exceed the limits set forth in § 70.24, "Criticality accident requirements," of this chapter must identify the disposal unit and facility design features that limit reconcentration of fissile material following disposal to ensure that the performance objectives of subpart C of this part will be met during the compliance period.

This guidance is intended to provide (1) approaches for determining the types of material for which criticality accidents need to be considered during operations, including emplacement, at a near-surface disposal facility, and (2) approaches for preventing criticality for those situations in which there is a potential for a criticality accident. Disposal site designs currently in use vary

from engineered systems with concrete vaults to landfill-style embankments. Emplacement depths range from around 6 to 14 m (20 to 45 ft). The controls and design features used to prevent criticality will be specific to the facility and waste to be possessed and disposed.

Section 3 in NUREG/CR-6626, "Emplacement Guidance for Criticality Safety in Low-Level-Waste Disposal," issued July 1999 (NRC, 1999a), provides useful perspectives on criticality safety concerns relevant to LLW disposal facilities, including the following:

> Spontaneous nuclear fission occurs naturally in a very small percentage of radioactive decays of SNM atoms. When fission occurs, neutrons are emitted along with the nuclear fragments (e.g., cesium and strontium atoms). These neutrons may be absorbed by a non-SNM nucleus, may be absorbed by a[n] SNM nucleus and cause fission, or may be lost from the system through leakage or some other mechanism. In natural uranium (enrichment of 0.71 wt % uranium-235) that is present in most soil, the neutrons produced during spontaneous fission are typically absorbed by a non-SNM nucleus.
>
> Unlike spontaneous nuclear fission, criticality is a chain reaction of fission events where large numbers of neutrons are produced. Criticality occurs when, on average, one neutron produced during a fission goes on to produce one other fission. Since more than one neutron may be produced during each fission event, some neutrons can be lost to absorption or leakage while still possibly leaving enough neutrons to produce additional fissions. In other words, criticality requires a balance between neutron production and neutron loss. Criticality safety relies on minimizing neutron production and maximizing neutron losses.
>
> ***
>
> Even though a criticality can intentionally be produced with a relatively small mass of SNM under ideal conditions, it is more difficult for a criticality to occur in more diffuse material. However, it is not inconceivable for a criticality accident to occur at an LLW disposal facility. For example, if containers with SNM of sufficient density, which are not critical individually, were stacked in a large array, such as in a disposal cell, a criticality could occur. …In general, the most important factors that affect the criticality safety of an LLW disposal site are the following:
>
> 1. the quantity, isotope, enrichment, and distribution of the SNM;
> 2. the presence of moderating materials, and their distribution; and
> 3. the presence of neutron-absorbing materials, and their distribution.

G.6.3.1 Consideration of Criticality Accidents during Operations

The quantity and concentration of SNM present determines whether the material is subcritical and thus does not require criticality controls during operations, including emplacement. As a first step, a licensee should determine whether the material is subcritical due to the quantity of material present (Item 1 below) or the concentration of SNM (Item 2 below):

(1) Mass quantities shown to be subcritical:

> The requirements of 10 CFR Part 71, "Packaging and Transportation of Radioactive Material," ensure criticality safety for the transportation of packages containing SNM.

For disposal facilities licensed by NRC Agreement States, the amount of SNM in packages is further constrained by the mass limits in 10 CFR Part 150, "Exemptions and Continued Regulatory Authority in Agreement States and in Offshore Waters Under Section 274." The amount of SNM that can be licensed by an Agreement State is defined in 10 CFR Part 150 as 350 g (0.77 lb) of U-235, 200 g (0.44 lb) of U-233, and 200 g (0.44 lb) of plutonium (1 g = 0.0022 lb). If mixtures of SNM isotopes are present in the waste, the "sum of fractions" rule applies. These package limit quantities of SNM have been shown to be subcritical (NRC, 1999a).

(2) Concentrations shown to be subcritical:

Extensive analyses in NUREG/CR-7239, "Review of Exemptions and General Licenses for Fissile Material in 10 CFR 71," issued January 2018 (NRC, 2018b), support the fissile material exemptions at 10 CFR 71.15, "Exemption from classification as fissile material." NUREG/CR-7239 considered a variety of scenarios and accident conditions in analyzing the safety margin provided by the low-concentration exemption at 10 CFR 71.15(c) that requires at least 2,000 g (4.4 lb) of solid nonfissile material for every gram of fissile material for exempting material from classification as fissile material. The accident conditions considered included fire, water immersion, reconfiguration into a worst-case geometry, and the combining of material from multiple packages. The exemption is adopted at 10 CFR 61.16(b)(2) when considering the quantity of material that would be subject to the requirements of 10 CFR 70.24, "Criticality accident requirements."

Waste material not meeting either of the above two criteria needs further evaluation to determine whether criticality controls are necessary for storing and emplacing waste containers (e.g., the number of waste containers to be stacked and placed next to each other).

G.6.3.1.1 Guidance for Minimizing Criticality for Stored and Emplaced Waste

A licensee emplacing containerized waste material containing SNM not meeting the criteria specified in Section G.6.3.1 will need to consider whether the placement configuration of the containers either during storage or emplacement in the disposal unit requires additional controls for criticality safety. NUREG/CR-6626 provides a graded approach for determining the need for additional criticality review:

The purpose of this guidance is to provide a way that LLW disposal facility licensees could demonstrate that SNM waste at emplacement will not cause a nuclear criticality accident. In addition, the guidance can be used by regulators in Agreement States that license LLW disposal facilities. The guidance presents a graded approach to allow flexibility and is based on several assumptions. Users of the guidance should review the compatibility of these assumptions with the characteristics of the waste and disposal site. In addition, site-specific criticality safety analyses based on other assumptions could be provided by a licensee to demonstrate that SNM waste at emplacement would not cause a criticality accident. (NRC, 1999a)

Section 6 of NUREG/CR 6626 provides three graded approaches that are applicable to SNM-contaminated LLW that meets the following general conditions:

1. The waste does not contain more than 0.1 weight percent (wt %) of beryllium, graphite, or deuterium.

2. The SNM is homogeneously distributed in the waste or meets the uniformity criteria as described in Section 5.1.5 of NUREG/CR-6626.

3. The mass of plutonium-241 does not exceed the mass of plutonium-240.

NUREG/CR-6626 includes further details on applying the three graded approaches. An example is provided below for the First Graded Approach provided in NUREG/CR-6626. The First Graded Approach is the most conservative approach of the three approaches in NUREG/CR-6626. This approach is comprised of an areal density determination or an average enrichment determination; either may be used to assess subcriticality conditions.

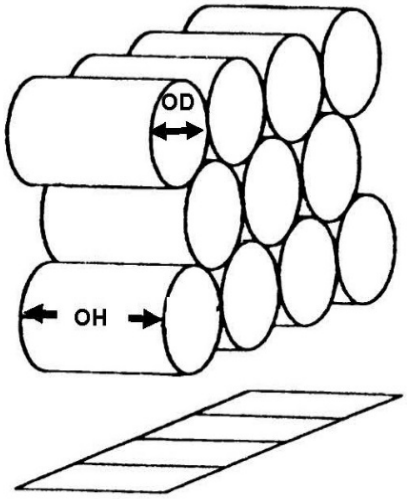
Areal Density Determination

The areal density is the mass of SNM per unit area of a disposal site as if the SNM were projected downward onto a horizontal surface. For example, a disposal unit that contains 0.14 g per m^3 (5 g per ft^3) and is 3 m (10 ft) deep would have an areal density of 4.2 g per m^2 (50 g per ft^2). The advantage of using areal density as a measure of the SNM in a disposal site is that vertical settling or compaction will not change the areal density value (NRC, 1999a). Only lateral movement would impact the amount of SNM in a column of material. Figures G-8 and G-9 provide potential configurations of SNM containing waste in storage or emplacement and the approach for determining areal density.

Operational limits established for the areal density for the First Graded Approach are based on conservative assumptions, such as hypothetical cylinders filled with SNM solution placed in an infinite planar array, each cylinder limited to a subcritical SNM mass (i.e., 350 g (0.77 lb) for U-235, 250 g (0.55 lb) for U-233, and 225 g (0.50 lb) for Pu-239), and a safety margin of 2.3 to account for accidental double batching and uncertainties. The configuration producing the minimum critical areal density served as the starting point for the limits in the First Graded Approach. The critical areal densities were reduced by 20 percent to produce subcritical limits, and then by another 10 percent to produce the operational limits. The areal density operational limits from NUREG/CR-6626 are reproduced in Table G-14.

Determination of the areal density for comparison to the operational limits is easily done for a single 55-gal drum, which can be assumed to have a base area of approximately 2.9 ft^2. If this drum contains 350 g U-235 (enriched to no more than 10 percent), its areal density is determined by dividing the mass of SNM (350 g) by the area (2.9 ft^2), which results in a value of 121 g/ft^2. This value exceeds the operational limit for U-235 in Table G-14 of 94 g/ft^2 and, therefore, does not provide for subcriticality conditions under the areal density method. However, in a closely packed triangular-pitched array of drums, 12 ft^2 of base area can contain three 55-gal drums. Therefore, if one drum contains 350 g U-235, and the remaining two drums contain 200 g U-235 each, the average areal density over the 12 ft^2 area is 62.5 g/ft^2 (((350 + 200 + 200)/12 ft^2) = 62.5 g/ft^2), which meets the limit in Table G-14 for maintaining subcritical conditions.

Figures G-8 and G-9 show a similar calculation for additional containers and two different drum orientations that are determined to be below the operational limit for areal density (i.e., a subcritical configuration).



Given 3 layers of 55-gallon drums
(stacked horizontally)

Containing 145 g U^{235} per drum
and no bk beryllium or carbon

Areal density determined by

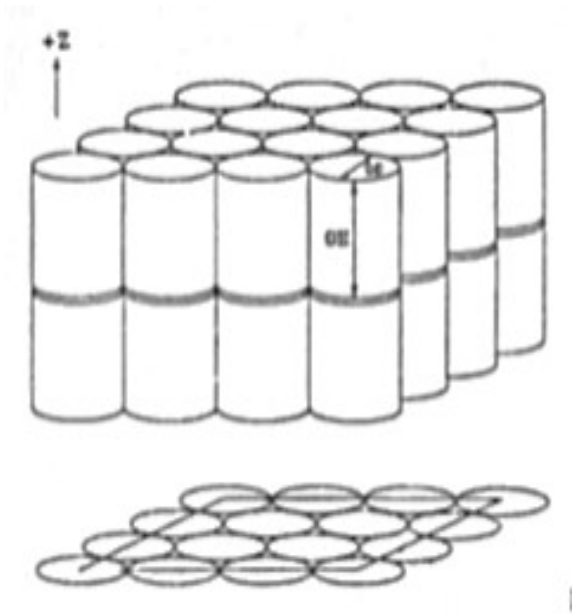
(M x N)/(OD x (OH)
where

M = fissile isotope mass (grams per drum)
N = number of layers of drums
OD = drum outside diameter (1.9 ft)
OH = drum outside height (r.9 ft)

(3 x 145)/(1.9 x 2.9) =

78.9 g U^{235} per ft^2

Figure G-8 Potential Configurations of Horizontally Stacked Canisters Containing SNM (based on Figure 2.1 in NUREG-6284 (NRC, 1995)



Given 2 layers of 55-gallon drums (stacked vertically)

Containing 125 g U^{235} per drum and no bulk beryllium or carbon

Areal density determined by

(M x N)/(0.866 x OD)
where
M = fissile isotope mass (grams per drum)
N = number of layers of drums
OD = drum outside diameter (1.9 ft)

(2 x 125)/(0.866 x 1.9) =

151.9 g U^{235} per ft^2

Figure G-9 Potential Configurations of Vertically Stacked Canisters Containing SNM (based on Figure 2.2 in NUREG-6284)

Table G-14 Areal Density Operational Limits (from Table 7.1 in NUREG/CR-6626)

SNM Radionuclide(s)	Areal Density (g/ft^2)*
100 wt % enriched uranium	94 (U-235)
10 wt % enriched uranium	174 (U-235)
U-233	82 (U-233)
Pu-239	52 (Pu-239)
Isotopic mixture of plutonium (76 wt % Pu-239, 12 wt % Pu-240, and 12 wt % Pu-241)	51 (Pu-239)

* The areal density in kg/m^2 can be obtained by multiplying the g/ft^2 values by 0.010763.

Subcritical Enrichment Levels (based on Section 7.1.1.2 in NUREG/CR-6626)

Enrichment is a ratio of the weight of U-235 or U-233 to total uranium and is typically expressed as a percent. This enrichment (wt %) can be calculated using information on the manifest. The activity of U-238 in MBq, if present, would be recorded on the manifest (NRC Form 541, Item 15). The activity can be converted to a mass of the isotope by dividing by the specific activity, or the mass of SNM isotopes can be obtained directly from the manifest (NRC Form 541, Item 1). The enrichment can then be calculated using the masses of the required isotopes.

The First Graded Approach in NUREG/CR-6626 also provides that SNM isotopes at very low enrichments may be disposed of without regard for concentration or areal density of the disposed waste. As stated in NUREG/CR-6626:

> ….waste contaminated with uranium that has not been enriched in uranium-235 above the natural abundance of approximately 0.71% may be placed in a disposal site with no concentration limits. Under conditions normally found in a a disposal environment, uranium that is below 0.96 wt% uranium-235, with the rest being uranium-238, cannot be made critical.

NUREG/CR-6626 specifies that waste containing SNM at or below the following enrichments may be placed in a disposal site without concern for a nuclear criticality accident:[7]

- U-235 = 0.96 wt %
- U-233 = 0.66 wt %

For example, consider a 55-gal drum containing 350 kg (772 lb) of waste contaminated with 100 g of U-233 at an enrichment of 1.5 wt %. This drum would then contain over 6.6 kg (15 lb) of U-238 (i.e., 98.5 percent of the mass is U-238). The quantities of U-233 and U-238 can be taken from the NRC Form 541 information, as described above. If three adjacent drums in the emplacement each contain 350 kg (770 lb) of waste contaminated with 100 g (0.22 lb) of U-233 at an enrichment of 0.4 wt %, each of these drums would therefore contain 25 kg (55 lb) of U-238. The total mass of U-233 in these four drums would be 400 g (0.88 lb), and the total mass of U-238 would be 81.6 kg (180 lb), making the average U-233 enrichment 0.49 wt %.

$$\frac{400 \text{ g U-233}}{400 \text{ g U-233} + 81{,}600 \text{ g U-238}} \times 100\% = 0.49 \text{ wt \%}$$

The total mass of waste in these four drums would be 1,400 kg (3,100 lb). Therefore, these four drums averaged together meet the subcritical enrichment criteria for the First Graded Approach.

The guidance in NUREG/CR-6626 is applicable to a range of disposal configurations; there may be conditions at a particular disposal site that warrant a site-specific evaluation. Geological or geochemical conditions at a site may make it unique such that a specific evaluation would be

[7] The uranium enrichment of waste may be averaged over no more than 1,500 kg (3,300 lb) of waste to meet these criteria.

worthwhile. Highly engineered design features that provide long-term protection of the waste from environmental degradation may allow for higher allowable SNM concentrations in the waste, as many specific wasteforms contain sufficient neutron-absorbing materials to make nuclear criticality extremely unlikely. For SNM waste that cannot meet the graded approach limits, a site-specific analysis or reliance on not exceeding the mass limits in 10 CFR Part 150 for a particular vault or disposal unit may be necessary for demonstrating subcritical conditions.

G.6.3.2 Reconcentration of Special Nuclear Material after Permanent Closure

As discussed in Section 3 of NUREG/CR-6626, it is reasonable to assume that the container will degrade and, at some point, may no longer be able to physically contain the waste following disposal. When the containers are breached, the waste could either settle and fill the bottom of a concrete vault or settle within the trench. Limiting void space can help mitigate the potential for differential settlement of material. In these situations, the waste could become slightly more dense, but the areal density, or grams of SNM per square foot, and the concentration of the SNM within the waste would be essentially unchanged. Reliance on an areal limit would not be impacted by differential settlement; however, a non-areal concentration approach may need to consider the impact of such settlement. The impact of differential settlement is generally considered to be less of a concern than an infiltrating water scenario.

One scenario that could result in redistribution of SNM within a disposal system is migration of SNM from transport with infiltrating water at a near-surface disposal facility. Water would infiltrate the disposal unit and SNM, particularly uranium, may be altered chemically (oxidized or reduced) and may move from its original disposal position. The concentration of SNM in the waste would change. The SNM in solution with the infiltrating water could move horizontally, vertically, or both and could be redeposited in another location. This process is called "reconcentration." Physical barriers to movement, such as vaults, can reduce the amount of SNM that is available for migration and reconcentration. However, such barriers can also act as collection points for mobile SNM. The larger the volume over which SNM collection is considered, the larger the potential increase in the local concentration of SNM. The distribution of SNM over the entire disposal site would need to be considered to evaluate whether large masses of SNM could be reconcentrated, thereby creating the potential for a nuclear criticality accident.

Two previous studies evaluate the potential impacts of reconcentration: NUREG/CR-6505, "The Potential for Criticality Following Disposal of Uranium at Low-Level-Waste Facilities," Volumes 1 and 2, issued June 1997 and June 1999, respectively (NRC, 1997; 1999b). Sections 4.3 and 4.4 of NUREG/CR-6626 discuss the scope and results of these studies. Based on these studies, the NRC concluded that the reconcentration process is unlikely and would take tens of thousands of years to reconcentrate the uranium. Note that simplifying assumptions were used to reach these conclusions and that large uncertainties remain. The Commission expressed concern (NRC, 2000) about the realism of a postulated criticality event based on reconcentration of SNM at the time the agency developed NUREG/CR-6626. The previous assessment did not consider quantities of SNM in GTCC waste, which are greater. The Commission advised the staff to consider the recommendations from these studies for the need to limit unusual moderators in LLW are discussed, as discussed in Section 3.3 in NUREG/CR-6626.

Two processes need to happen for reconcentration to occur: the first is the mobilization of radionuclides (SNM) in water, and the second is the subsequent immobilization of the previously

mobilized radionuclides (SNM). If the SNM is not immobilized, its concentration does not increase and there is no increased likelihood of a criticality accident. Certain compounds of uranium are highly soluble such as uranyl fluoride, uranyl nitrate, and uranyl potassium carbonate. Because uranium is expensive to enrich and soluble uranium is easily recoverable, it is likely that generators would reprocess this type of waste and recover the soluble uranium, therefore, substantial quantities of soluble uranium are not anticipated to be present in LLW. If soluble uranium compounds were present in LLW, the disposal facility operator might not be able to determine their presence from available documents. NRC Form 541 and Form 741, "Nuclear Material Transaction Report," require that only prevalent chemical forms of the waste be reported. This information may or may not include the SNM chemical form. Therefore, some uncertainty exists on the presence of soluble uranium. Highly soluble uranium would also be unlikely to be immobilized (the second process for reconcentration).

When large quantities of SNM (e.g., tens to hundreds of kilograms) are planned for disposal in a near-surface facility, especially if disposal would be in a single disposal unit, it will be important to consider the following:

- unusual moderating materials that may be present (Section G.6.3.2.1)
- good practices for preventing criticality from reconcentration (Section G.6.3.2.2)
- performing calculations for estimating the potential for criticality in deciding on the appropriate disposal unit design and configuration of the SNM within the disposal unit (Section G.6.3.2.3)

G.6.3.2.1 Co-emplacement of Special Nuclear Material with Unusual Moderating Materials

Water is both a good moderator (a material that slows neutrons) and a poor absorber (a material that does not capture neutrons). It is common in criticality safety evaluations to assume that water may be present with the SNM. However, other materials, such as beryllium, graphite, and heavy water (D_2O), are less efficient neutron moderators than water but have lower neutron-absorption characteristics than water. Therefore, a system of SNM with such moderators can become critical at a lower concentration (but not a lower mass) than it would with only water present. These moderating materials are used at many nuclear facilities and may be present with the SNM in the waste material. Guidance in NUREG/CR-6626 and relied upon in this document is not applicable if the waste contains more than trace amounts (0.1 wt %) of these materials. Furthermore, it is a good practice to segregate LLW containing SNM from LLW containing beryllium, graphite, or heavy water.

G.6.3.2.2 Good Practices for Emplacement Relevant to Special Nuclear Material

Several factors affect SNM migration or concentration. These include the amount of water entering the disposal unit, the chemistry of the waste and water, and the presence of preferential pathways, such as drains and sumps. To minimize the potential for migration, and to maximize the probability of detecting migration before it becomes significant, the following good practices should be incorporated:

- Infiltration of water at a near-surface disposal facility should be minimized to lengthen the life of the disposal containers and to limit the volume of water available to move the

uranium from its disposal location. Unsaturated conditions greatly reduce opportunities for migration and concentration.

- Highly soluble chemical forms of SNM should be minimized in a disposal site. Examples of soluble compounds include uranyl nitrate and uranyl fluoride, which are common chemical forms of uranium in waste from fuel processing facilities.

- Facility designs should minimize preferential pathways that would tend to concentrate or focus effluents. The use of sumps or other means of sampling the water below the LLW emplacement will provide a means of detecting migration before it becomes significant. However, the geometry and size of such collection points should be restricted to avoid potential critical geometries.

- Near-field disposal facility LLW emplacements of LLW should minimize zones with strong reducing potential. Uranium compounds tend to become more soluble in oxidizing conditions, and precipitate under reducing conditions. Rainwater tends to be oxidizing thus it dissolves uranium and transports it through a disposal site. Organic materials and iron tend to be reducing, so uranium in solution may precipitate when it encounters these materials, forming a reconcentration zone that could cause a nuclear criticality accident. Limiting infiltration will also help keep the SNM from oxidizing and dissolving.

- Where operationally feasible, the areal density of SNM should be as low as is practical; that is, SNM waste containers should not be stacked and should be placed as far from other SNM containers as is practical. Consideration should be given to the limited area associated with certain borehole disposal designs that could also be shown to limit criticality concerns.

- Where operationally feasible, the enrichment of SNM should be reduced; that is, depleted uranium, natural uranium, or natural thorium should be placed adjacent to SNM waste.

- Void space between SNM containers should be backfilled with sand or grout to minimize post-disposal settling.

G.6.3.2.3 Evaluation of Criticality Due to Reconcentration of Special Nuclear Material

As reported in NUREG/CR-6505, Volume 1 (NRC, 1997), the NRC conducted a study to determine whether fissile uranium in LLW facilities can be concentrated by hydrogeochemical processes to permit nuclear criticality. This investigation presented the first attempt to study the potential for nuclear criticality at LLW facilities using both quantitative hydrogeochemical processes and nuclear criticality safety calculations. Because of the numerous combinations of parameters that could be considered in nuclear criticality evaluations, the study used bounding and simplifying assumptions. Likewise, the hydrogeochemical simulations used simplifying assumptions because of the numerous combinations of parameters that can affect the mobilization of U-235. Should a determination be made that the potential for criticality warrants further evaluation, the approaches and methods documented in NUREG/CR-6505 Vol. 2 (NRC, 1999b), provide a good starting point.

G.7 Enhanced Controls, Limitations, and Specialized Designs

For some types of GTCC waste, enhanced controls, limitations, and specialized designs may be needed to ensure safe disposal in an LLW disposal facility. Some GTCC waste, such as waste containing long-lived alpha-emitting transuranic radionuclides in concentrations greater than 370 kBq/g (10,000 nCi/g), is generally not suitable for near-surface disposal, but may be acceptable for near-surface disposal on a case-by-case basis if approved by the Commission.
Most near-surface LLW disposal in the United States has used some form of engineered trenches, sometimes combined with reinforced concrete vaults or canisters. The depth of disposal has ranged from approximately 2 m (7 ft) to 40 m (130 ft) below the land surface. The NRC established the 370 kBq/g (10,000 nCi/g) limit for long-lived alpha-emitting transuranic radionuclides for near-surface disposal facilities based on assessing potential doses to inadvertent intruders who may drill into that waste at some time in the future. Because the waste is long lived, it is likely to remain hazardous for longer than many near-surface intruder barriers may remain effective. However, each radionuclide is different in terms of the hazard and half-life and, thus, the NRC staff acknowledges that specialized designs and controls may be implemented that could allow for the near-surface disposal of some types of GTCC waste above the 370 kBq/g (10,000 nCi/g) limit for long-lived alpha-emitting transuranic radionuclides. For example, controls that could effectively prevent inadvertent intrusion for 1,000 years are not very significant with respect to the decay of Pu-239, which has a 24,100-year half-life, but that timeframe is quite significant with respect to the decay of Pu-238, with an 88-year half-life.

A licensee should make use of a pre-licensing consultation to meet with the NRC staff to discuss the waste characteristics, proposed engineering design, and other controls that may be used to ensure safe disposal of the waste. For example, extensive coordination was completed for disposal of a unique waste component (reactor pressure vessel) at an LLW disposal facility as well as a high-activity sealed source in a common irradiator model at the US Ecology, Hanford (WDOH, 1998; Radiation Source Protection and Security Task Force Report, 2018). Pre-licensing consultation can achieve alignment that disposal of the proposed waste is allowed under the appropriate regulatory requirements. It can also be used to achieve alignment of the expected technical basis needed to demonstrate that the proposal is safe.

Deeper disposal can be used when uncertainties are unmanageable for near-surface disposal. Under 10 CFR 61.55(b)(1)(iii), a licensee has the option to demonstrate that any wastes that are LLW and meet the waste characteristics requirements can be safely disposed in the near surface. However, the NRC is the independent body tasked with determining that the submitted information demonstrates that the requirements have been met. Significant uncertainties can make the process of achieving alignment between the operator and regulator time consuming and expensive, perhaps cost prohibitive. Specialized engineered designs applicable to both near-surface and deeper land disposal include advanced wasteforms, disposal configuration, and barriers and materials used. Specialized designs may be necessary to mitigate uncertainties associated with the near-surface disposal of materials that pose higher risks for longer, compared to traditional LLW. Deeper disposal in a specialized land disposal facility is required for land disposal of GTCC waste with long-lived transuranic radionuclides exceeding 370 kBq/g (10,000 nCi/g). Although not required for other waste streams, deeper disposal can also be used to achieve the required performance or to limit uncertainties for disposal of wastes with characteristics that may pose performance challenges in typical near-surface disposal facilities (e.g., waste with significant heat loading or significant inventories of long-lived mobile radionuclides).

A licensee should consider advanced wasteforms for wastes that pose significant challenges for either near-surface or deeper disposal. Advanced wasteforms are those demonstrated to have long-term durability in the expected range of future disposal system chemical environments. Whereas ordinary Portland cement is commonly used in LLW disposal in the United States, other wasteforms are much less common. These may include borosilicate and lead phosphate glasses, ceramics, glass-ceramics, advanced cementitious wasteforms (e.g., grouts) such as tertiary blends with chemically reducing components, Synroc, and metallic wasteforms. It may be useful to apply the concept of thermodynamic diversity, which is to use combinations of compatible materials that have good performance in different extreme environments. Performance of advanced wasteforms must be demonstrated with adequate testing and analysis, as described in Section 3.2.4 of the main document. Advanced wasteforms will have benefits with respect to demonstrating compliance with 10 CFR 61.41; 10 CFR 61.42, "Protection of individuals from inadvertent intrusion"; and 10 CFR 61.43, "Protection of individuals from operations." Advanced wasteforms will result in the following:

- much slower release rates of radioactivity to water and air
- maintenance of waste as recognizable from natural materials for a longer time
- decreased dispersibility of the waste, especially under operational accident scenarios

Borehole disposal may be a viable concept for certain types of GTCC waste. Borehole disposal is the use of relatively small diameter pipes (wells) or caissons (about 15 cm to 2 m (6 in to 7 ft)) at depths ranging from tens of meters to a few thousand meters. Boreholes accomplish the desired goals of achieving greater depths without excessive costs while maintaining a small footprint. A small footprint (i.e., compared to a near-surface trench design) functions to limit the probability that an inadvertent intruder on a disposal site will contact waste, simply because of the smaller target area a borehole, or even multiple boreholes, presents to a vertical water well. For example, as discussed in IAEA (2017), the likelihood a water well placed 1 per 1000 m^2 (1 per 0.25 acres) would intersect a disposal facility with a cross-sectional area of about 0.05 m^2 (0.54 ft^2) is about 1 in 20,000, based on simple geometric considerations. In addition, as discussed in Section G.5.5, maintaining a small footprint can also make it more practical to use advanced engineered intrusion barrier designs that might be cost prohibitive over a larger area.

The IAEA has developed a program for borehole disposal of radioactive waste to facilitate safety and security of nuclear materials, especially disused sealed radioactive sources. Many smaller countries do not have a large nuclear infrastructure but use and eventually need to dispose of small volumes of radioactive waste (IAEA, 2009). A conceptual diagram of a borehole disposal facility is shown in Figure G-10 (IAEA, 2017). The design is narrow-diameter boreholes where steel canisters containing conditioned waste are stacked. A robust steel deflector plate is placed on top of the borehole, and the top portion of the borehole is removed and backfilled with native natural materials to make the exact locations of the boreholes not readily identifiable to an adversary (10 CFR Part 61 requires the use of permanent markers for the disposal facility; see Section 9.1.2 of the main report).

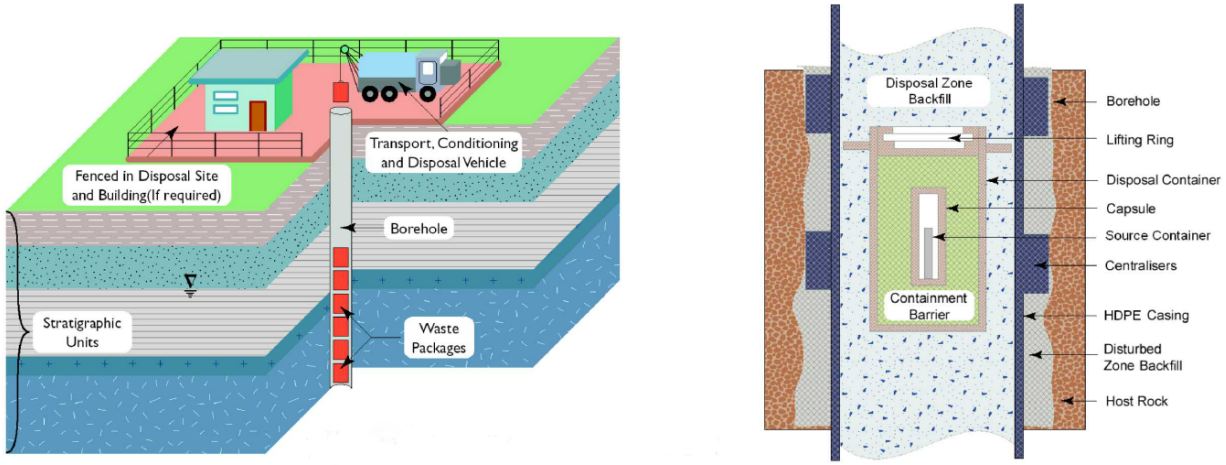


Figure G-10 Conceptualization of the Borehole Disposal Concept (IAEA, 2017)

The steels used will likely be stainless and the conditioning materials are typically cements. Borehole disposal has been implemented in South Africa and has been licensed and is progressing towards implementation in Malaysia. Borehole disposal is being assessed in Ghana and is progressing towards licensing. Based on the initial successes of these programs, other countries are considering borehole disposal.

The DOE used disposal in large diameter near-surface boreholes at the Greater-Confinement-Disposal area of the Nevada Nuclear Security Site (Cochran et al., 2001). Four boreholes were used to place special case TRU waste a minimum of 21 m (69 ft) below the land surface. The boreholes are approximately 3 m (10 ft) in diameter and extend to a depth of 37 m (120 ft). The total mass of the waste was approximately 60,000 kg (130,000 lb) and the average concentration of transuranic radionuclides exceeded 185 kBq/g (5,000 nCi/g).

For water resources, protection against inadvertent intrusion can be provided when the depth of potable water an intruder might drill to is significantly less than the depth at which the waste is emplaced, because an intruder would have no reason to drill a well deeper than the usable potable water zone. Placing waste below usable potable water (e.g., within brackish water zones beneath a potable aquifer) would make it unlikely that a domestic water well would reach the depth of emplaced waste. Several deep saline aquifers overlain by mechanically robust, very low conductivity caprock (e.g., those in the Illinois, Michigan, Williston, Newark, Gettysburg, and Culpepper Basins) may provide natural features suitable for isolating GTCC waste in deep boreholes.

Not all locations or disposal concepts will be able to take advantage of placing waste below any potable water, because the depth to saline water may be great enough to make emplacement at that depth cost-prohibitive. However, for sites and designs where waste can be emplaced below the potable aquifer, this configuration offers a durable means of protection because, once established, it does not depend on engineered features to maintain the isolation of waste from the water resource. For example, a 1 kilometer deep borehole could dispose of waste in the bottom 500 meters allowing a 500 meters above the waste as a means to preclude human intrusion, in combination with other engineered and natural features. Available information [see Figure 12 from Stanton et al. (2017)] indicates that in many areas in the conterminous United States, a 1-km deep borehole with waste emplacement 500 meters (1640 feet) or more below

the ground surface could place the waste in deep saline water, below potable water resources that an inadvertent intruder could attempt to access.

The safe land disposal of GTCC waste requires consideration of the specific physical, chemical, and radiological characteristic of the wastes to be disposed of to determine the most appropriate combination of enhanced controls, limits, and specialized designs to ensure safe disposal. Because waste streams vary widely in thermal output, mobility of radionuclides, and long-lived radiological hazards, licensees must evaluate whether near-surface disposal or, deeper specialized designs are most appropriate to ensure safe disposal of the waste streams to be disposed. Advanced wasteforms, specialized disposal geometries, and favorable hydrogeologic conditions can substantially reduce uncertainties and enhance long-term waste isolation. Early and ongoing consultation with the NRC staff is essential to ensure that proposed approaches align with regulatory requirements and provide a sound technical basis for demonstrating disposal will protect offsite members of the public and inadvertent intruders throughout the compliance period.

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ATTACHMENT G-1
POTENTIAL THERMAL IMPACTS FROM GREATER-THAN-CLASS C WASTE DISPOSAL

The U.S. Nuclear Regulatory Commission (NRC) staff evaluated the disposal of the Greater-Than-Class-C (GTCC) waste in the U.S. Department of Energy (DOE)'s DOE/EIS-0375, "Final Environmental Impact Statement for Disposal of Greater-Than-Class C (GTCC) Low-Level Waste and GTCC-Like Waste," issued January 2016 (DOE GTCC FEIS) at a single disposal site. This attachment describes thermal calculations that were performed to understand the key considerations when evaluating potential thermal impacts from the disposal of heat-generating waste.

The DOE GTCC FEIS proposed three representative approaches: (1) trench disposal, (2) vault disposal, and (3) large borehole disposal. Currently, all commercial low-level waste disposal in the United States uses trench disposal concepts. The DOE considered two representative disposal containers for contact-handled (CH) waste (208-liter (L) (55-gallon (gal))) drums and standard waste boxes (SWBs)). Small cesium (Cs)-137 irradiators were assumed to be placed in drums, and large Cs-137 irradiators were assumed to be handled separately as self-contained packages. The DOE assumed that most remote-handled (RH) waste would be packaged in drums, with activated metal packaged in unshielded stainless steel activated metal canisters (AMCs).[8]

Table G-1-1 summarizes the total count of waste containers that have been generated or are projected to be generated from existing facilities (Group 1) and waste that might be generated from future activities (Group 2). Table G-1-2 summarizes the required number of disposal units (trenches, vault cells, or boreholes) needed to accommodate the estimated number of waste containers, based on the packing densities assumed by the DOE (2016b, Appendix D.3). The CH waste streams are assumed to be segregated into separate cells, with each cell containing one type of waste package. However, two RH streams can be combined within trenches and vault cells because the drums and AMCs have sufficiently compatible dimensions.

A schematic representation of the vault disposal design (Table G-1-1) illustrates basic characteristics of the different hypothetical disposal configurations. A single vault consists of 11 disposal cells in a linear array, with concrete exterior walls, floor, and roof. Each disposal cell holds either CH or RH waste. A RH vault cell can hold 72 concrete cylinders vertically stood on end in an 8 by 9 hexagonal pattern, with each cylinder holding four drums, three AMCs, or one RH canister. Gaps between cylinders are assumed to be filled with pea gravel. The vault is covered by approximately 5 meters (m) (16.4 feet (ft)) of earthen materials, potentially with several layers including a geomembrane flow barrier. The waste disposal interval is approximately 5 to 10 m (16.4 to 33 ft) below the ground surface.

The trench design is like the vault disposal approach, except that the trenches have no concrete walls or floor, and the RH trenches only have three rows of reinforced concrete cylinders across the trench width. Trenches have a reinforced concrete lid below the earthen backfill. The borehole design uses a rectangular grid of boreholes, with each assumed to have just one of the five waste categories in Table G-1-1. Each waste container type has a different packing

[8] The containers are not activated metals, rather they are packages that will contain activated metals and are referred to as activated metal canisters.

options. An earthen cover of 30 m (100 ft) is assumed to be present over a reinforced concrete lid above the topmost waste container; otherwise, the waste containers are surrounded by unconsolidated material (clean sand backfill and the native earth). The waste disposal interval is approximately 30 to 40 m (100 to 130 ft) below the ground surface.

Table G-1-1 Estimates of Total Waste Container Counts from Table B-11 of the GTCC FEIS, Combining Groups 1 and 2, for the Land Disposal Option (DOE, 2016b)

Waste type	Container Type	Waste Volume (m^3)	Number of Containers
RH	AMC	2,007	5,515
RH	55-gal drum	3,986	19,196
CH	Sealed Cs-137 irradiators	1,019	1,435
CH	SWB	2,748	1,471
Contact handled	55-gal drum	1,887	9,082

Table G-1-2 Summary of Waste Package Allocations Based on Packing Densities for each Land Disposal Option (DOE, 2016b, Appendix D.3)

		Trench	Vault cell	Borehole						
Waste	Container	Cnt*	N	F	Cnt*	N	F	Cnt*	N	F
RH	AMC	906	6.1	0.21	216	25.5	0.20	36	153.2	0.17
RH	55-gal	1208	15.9	0.55	288	66.7	0.53	56	342.8	0.38
CH	Sealed	1680	0.9	0.03	300	4.8	0.04	20	71.8	0.08
CH	SWB	500	2.9	0.10	100	14.7	0.12	8	183.9	0.20
CH	55-gal	3000	3.0	0.11	630	14.4	0.11	56	162.2	0.18

*Cnt: maximum number of waste containers to fill the disposal unit
N: number of disposal units required
F: fraction of disposal units used

G-1.1 Thermal Responses to Generated Waste Heat

A simplified analytical approach illustrates some of the essential characteristics of the thermal response of the system and is likely to be sufficient in most cases. In practice, a numerical model that accounts for the thermal properties of each of the materials could be used to determine the detailed temperature field around AMCs. A vault containing AMCs is a heated mixture of AMCs; reinforced concrete cylinders; possibly some RH waste packages; concrete walls, floor, and ceiling; and granular fill, surrounded by earthen materials. A trench containing AMCs omits the concrete walls and floor, but retains the concrete cylinders, AMCs, and possibly some RH waste packages. A borehole containing AMCs includes just the containers and granular fill with a concrete lid.

In all three systems, the ground surface remains in approximate thermal equilibrium with the atmosphere, slightly warmed by energy from the vault. In a conduction-dominated system, dissipation of a fixed thermal flux is proportional to the thermal conductivity and the temperature gradient. At steady state, overall release from the vault is controlled by the thermal conductivity and temperature gradient in the surrounding earthen material. Transfer of thermal energy from the AMCs to the exterior of the vault is controlled by the bulk thermal conductivity and temperature gradient within the vault. The bulk thermal conductivity inside the disposal cell is expected to be substantially larger than outside, because the activated metal in the AMCs is predominantly stainless steel, with a much larger thermal conductivity than granular media.

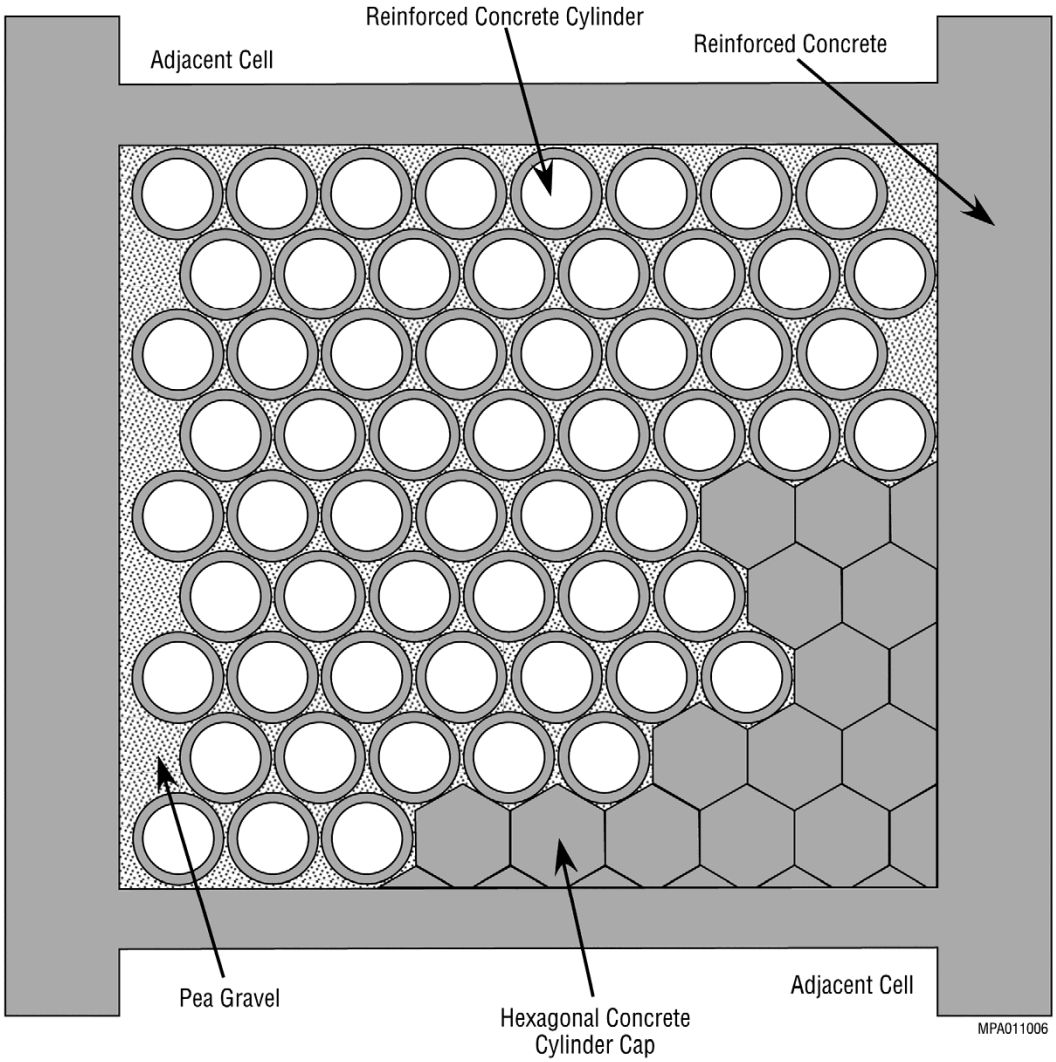


Figure G-1-1 Top View of a Vault Cell for Disposal of Remote-Handled Waste (DOE, 2016b, Figure D-11)

The temperature change in an infinite homogeneous domain resulting from a point source with exponentially decaying power is shown as follows:

$$v = v_0 \exp(-u^2)\, \mathcal{R}[\mathrm{w}(f + iu)]$$

$$w(z) = \exp(-z^2)\, \mathrm{erfc}(-iz)$$

$$v_0 = \left(\frac{1}{4\pi r}\right)\left(\frac{q_0}{\lambda}\right)$$

$$q(t) = q_0 \exp\left(-\frac{t}{\tau}\right)$$

$$u^2 = \frac{r^2}{4\kappa t} = \frac{r^2}{4\kappa\tau f} = \frac{1}{f}\left(\frac{r}{R}\right)^2$$

$$f = \frac{t}{\tau}$$

$$= \frac{\lambda}{\rho c}$$

$$\tau = \frac{T_{1/2}}{\ln(2)}$$

Where:

v	change in temperature from background due to the source (K)
q_0	initial rate of energy supply at time of emplacement (J/T)
r	distance from the source (L)
t	time after emplacement (T)
$T_{1/2}$	source power half-life (T)
ρ	material density (M/L^3)
c	material specific heat (J/M-K)
λ	material thermal conductivity (J/T-M-K)
κ	thermal diffusivity (L^2/T)

The v_0 group is the steady state solution for a point source with continual release at the initial source rate, the R parameter is a characteristic distance based on thermal diffusivity and source decay rate, and the f parameter describes the time relative to the source decay rate. The $\mathrm{w}(z)$ function is called the Faddeeva function (Weideman, 1994). The steady state solution is a geometric term and a scale factor consisting of the ratio between source power and thermal conductivity (i.e., the steady temperature is proportional to the source power and inversely proportional to thermal conductivity) (Carslaw and Jaeger, 1959). The transient response to steady state for a constant source rate (infinite $T_{1/2}$) is shown as follows:

$$\frac{v}{v_0} = \mathrm{erfc}(u)$$

A quasi-steady approximation assumes that the transient response to change in power dissipates very quickly compared to the time-dependent change in power $q(t)$ (very large R), leading to the following:

$$\frac{v}{v_0} = \frac{q(t)}{q_0} = \exp(-f) = \exp\left[-\frac{\ln(2)t}{T_{1/2}}\right]$$

Figure G-1-2 shows the transient temperature deviation from the ambient temperature, normalized by the deviation at the shortest time and distance in the figure. The horizontal axis is time normalized by the source half-life, and the vertical axis is radial distance normalized by a characteristic distance relative to the source half-life. There are three regions for the transient solution: (1) quasi equilibrium (small distance and time after emplacement), (2) propagation (large distance and small time after emplacement), and (3) residual spreading (more than a few half-lives after emplacement, after the bulk of the source energy has already been released). The quasi-steady approximation uniformly overestimates the temperature deviation (Figure G-1-3). Although the relative error may be very large, the absolute difference in temperature between the two solutions is generally less than 0.1 percent of peak v (i.e., evaluated for early times and short distances).

The v field resulting from a volume containing a heat source can be calculated by integrating the point-source solution over the volume. The integral can be approximated by subdividing the volume into a set of *N* smaller subvolumes, each with a point source, and summing the contributions from each subvolume. The power for each subvolume, q_{0i}, is the total power in the source volume multiplied by ω_i, the fraction of the total volume occupied by the subvolume:

$$q_{0i} = q_0\omega_i$$

$$v = \sum_{i=1}^{N} v_i$$

With this approach, a stack of AMCs is represented by a set of *N* point sources, and the v field in and near a disposal cell results from the superposition of the point sources for each nearby AMC. A zero-change boundary (the ground surface) is represented with mirror point sources that have the same source power but opposite sign. In practice, it is useful to precalculate a lookup table for v around the stack centroid, accounting for all *N* point sources and the mirror images, to reduce the need for repeated summations.

Table G-1-3 shows the thermal properties of representative materials and Table G-1-4 provides the effective radius for approximate thermal equilibrium after one half-life. Values for porous media significantly depend on the bulk density and water content, because of the contrasting properties of air, water, and solid grains in the mixture. In general, thermal conductivity and diffusivity increase with bulk density; for a fixed dry density, these also increase with water content. Concrete has a much wider range of values than shown in Table G-1-3 (across the combinations of porosity and composition). Using representative properties of 5 to 50 m^2/yr for thermal diffusivity to estimate the effective radius of influence after one half-life yields 10 to 32 m (33 to 110 ft) for cobalt (Co)-60 and 45 to 140 m (150 to 460 ft) for nickel (Ni)-63. This implies that the steady-state solution using q_0 should be relatively accurate for the local influences of

AMCs within a disposal cell and perhaps the closer parts of the adjacent cells after 5 years, but the small contributions from more distal cells will be overestimated.

At a 5 to 10 m (16.4 to 33 ft) AMC depth for the trench and vault scenarios (mirror source distance is 10 to 20 m (33 to 66 ft)), for most porous media the thermal field will be at or very close to equilibrium with the ground surface boundary within the Co-60 half-life. The much deeper borehole case (mirror source distance is 60 to 80 m or 200 to 260 ft) would take most of a century to fully equilibrate to a steady source, so peak temperature changes may not reflect the surface boundary condition.

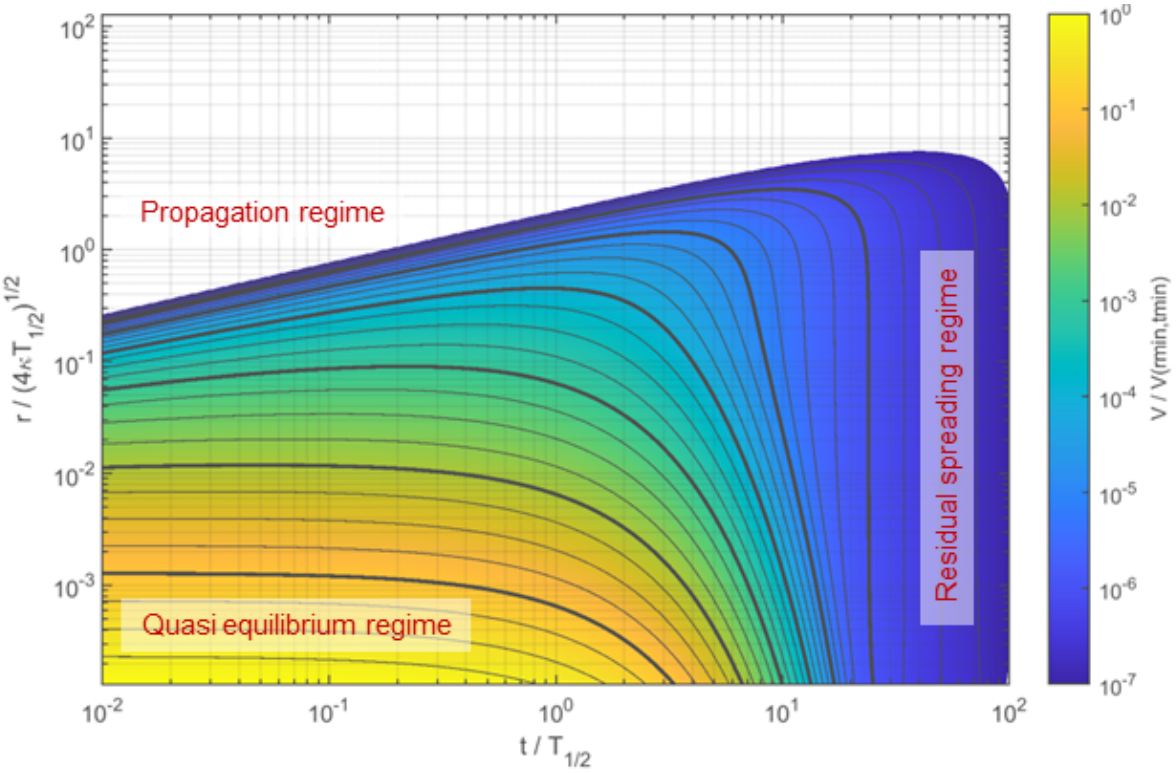


Figure G-1-2 Transient Deviation from the Ambient Temperature for the Analytic Solution[9]

[9] Time starting at emplacement, normalized by the deviation at the smallest time and radius in the figure. The time and radius axes are normalized by a characteristic time and length, respectively.

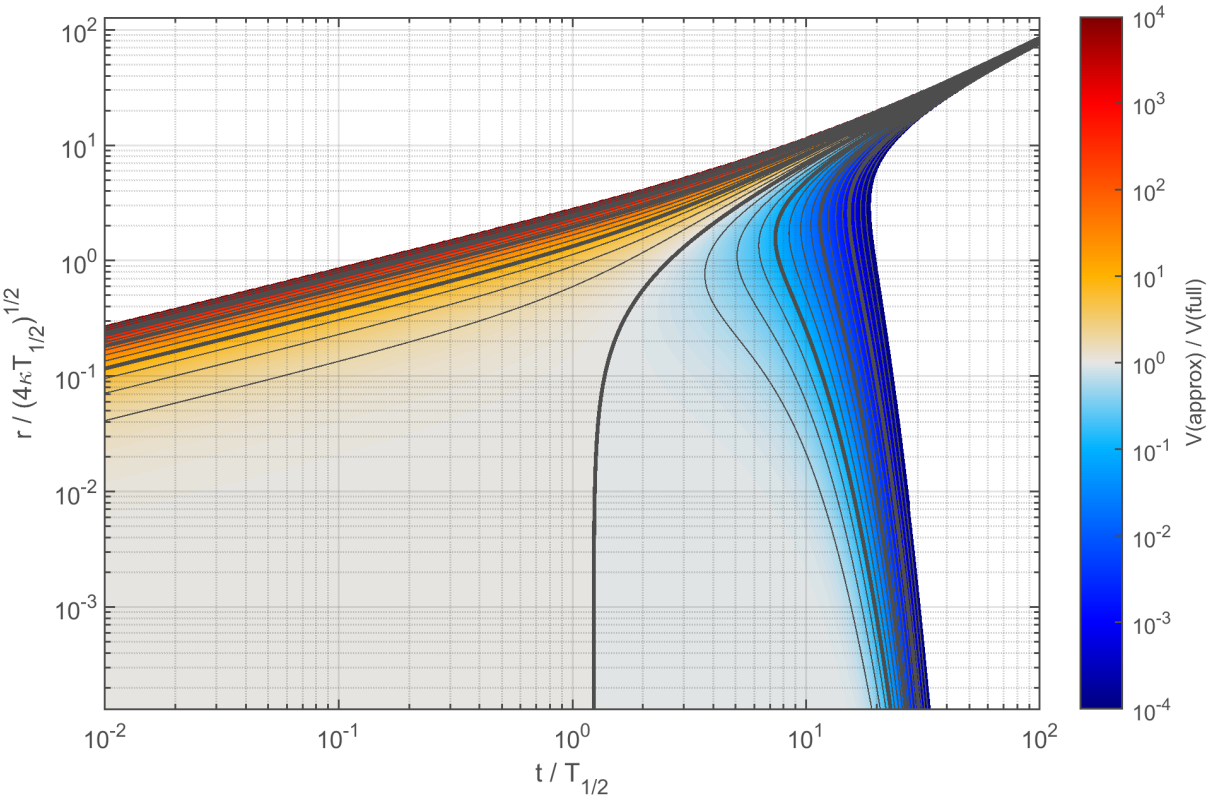


Figure G-1-3 The Ratio Between the Quasi-steady Approximation and the Full Analytic Solution

Table G-1-3 Thermal Properties of Representative Materials

Material	Note	Thermal conductivity (W/m-K)	Density (kg/m^3)	Thermal capacity (J/kg-K)	Thermal diffusivity	
					(×10^{-7} m^2/s)	(m^2/yr)
Water		0.602	1000	4184	1.44	4.5
Air		0.0243	1.3	1004	186	588
Stainless steel	316	16.3	8027	502	40.5	128
Carbon steel	0.5% C	54	7833	456	151	477
	1.5% C	36	7753	486	95.5	302
FEBEX bentonite clay	Dry	0.571	1600	802	4.45	14
	Wet	1.26	2010	802	7.82	25
Sand	Dry	0.25	1500	830	2.0	6.3
Lightweight / fibered earth	Min	0.09	300	600	1.9	6
	Max	0.47	1200	1500	5.5	17
Solid / massive earth	Min	0.5	1400	600	3.8	12
	Max	1.7	2200	1100	7.7	24
Normal concrete	Min	1.6	2520	920	6.9	22
	Max	3.2	2059	1160	13.4	42

G-1.2 Thermal Responses for Disposal Strategies

This analysis assumes that the disposal cells are placed on different rectangular grids for the trench, vault, and borehole disposal strategies, and the AMCs are packed differently within the disposal cells for each strategy. The three disposal strategies have different implications with respect to thermal dissipation. Thermal dissipation is increased by (1) increasing the disposal cell surface area to volume ratio and (2) increasing the separation between disposal cells, and the need for thermal dissipation is reduced by reducing the number of AMCs in a disposal cell. The vault approach has the densest AMC packing with respect to both disposal cell factors, while the trench and borehole approaches have relatively slender packing strategies with greater surface area to volume ratios.

Implications of the thermal management schemes can be understood by examining two representative endpoint packing options. The packed option divides the disposal cell rectangle into quarters and places the AMCs compactly in one of the quarters. The wide option spreads the AMCs approximately uniformly across all four quarters, within the disposal cell packing constraints. In trenches and vaults, AMC stacks are placed every fourth location along rows and the starting point is offset between rows. The borehole packing strategy completely fills each borehole with one type of container, so the same spreading strategy is by borehole instead of AMC stack.

The disposal strategies are compared using a geometric factor ϕ based on the steady-state temperature distribution for a constant source rate:

$$v = \left(\frac{q_0}{\lambda}\right)\sum_{i=1}^{M}\sum_{j=1}^{N}\left(\frac{\epsilon_j}{4\pi r_{ij}}\right) = \left(\frac{\sigma_0 V}{\lambda}\right)\sum_{i=1}^{M}\sum_{j=1}^{N}\left(\frac{\omega_j}{4\pi r_{ij}}\right)$$

$$\phi = v\left(\frac{\lambda}{\sigma_0}\right) = V\sum_{i=1}^{M}\sum_{j=1}^{N}\left(\frac{\omega_j}{4\pi r_{ij}}\right)$$

$$q_0 = \sigma_0 V$$

where σ_0 is the power density inside the source (J/T-L^3) and V is the source volume (L^3). The temperature deviation is calculated for different emplacement conditions by multiplying ϕ (m^2) by the power density inside the AMC (W/m^3) and dividing by thermal conductivity (W/m-K).

Table G-1-4 Effective Radius for Approximate Thermal Equilibrium after One Half-life

Thermal diffusivity κ (m^2/yr)	Effective radius $\left(4\kappa T_{1/2}\right)^{1/2}$ (m)		
	Co-60 $T_{1/2}$ = 5.26 yr	Cs-137 $T_{1/2}$ = 30 yr	Ni-63 $T_{1/2}$ = 100.1 yr
1	4.6	11.0	20.0
2	6.5	15.5	28.3
5	10.3	24.5	44.7
10	14.5	34.6	63.3
20	20.5	49.0	89.5
50	32.4	77.5	141.5
100	45.9	109.5	200.1
200	64.9	154.9	283.0
500	102.6	244.9	447.4

Six different examples of disposal facilities and packing strategies were examined. The six examples combining disposal facility with packing strategy reveal similarities and differences in the geometric factor when plotted in plan view on a common spatial scale and color range (Figure G-1-4). The horizontal plane passes through the centroid of each AMC stack. The vault facility has the densest packing, but temperatures remain relatively low between the disposal cells. The borehole facility has relatively warm conditions between the boreholes, compared to the other facilities, even though the boreholes are spaced relatively widely, because the top depth for the AMC stacks is 30 m (100 ft), while for other facilities the top depth is 5 m (16.4 ft). The vertical temperature gradient from the ground surface boundary condition to the AMCs is approximately the same in all three cases. Extending the gradient to a greater depth increases the average AMC temperature. However, this behavior will not fully develop for waste where the heat is generated by Co-60, because the transient time needed to develop the steady state is much longer than the Co-60 half-life.

Figure G-1-5 is a closeup plan view of the trench option for a plane passing through the centroid of the middle AMC in the three-AMC stack (stacks are represented by magenta circles). The top plot represents ϕ and the bottom plot is the horizontal gradient of ϕ. The peak value for ϕ in is approximately 1.9 m^2 (20 ft^2).

As an example, assume a licensee wants to evaluate AMC's that are 75 percent filled with waste containing Co-60. The initial power density for the AMC contents is 875 W/m^3 from Table G-1. If the waste was aged for 42 years (eight half-lives) before emplacement, the emplacement power density for the AMC contents is 2.56 W/m^3 (calculated as $0.75 \times 875 / 2^8$ W/m^3). Assuming that the representative medium is soil with a thermal conductivity of 0.4 W/m-K, the peak temperature increase above ambient is (1.9 m^2) × (2.56 W/m^3) / (0.4 W/m-K) = 12.2 kelvin (K). Assuming the annual temperature near the disposal facility is approximately 18 degrees C (°C), then the peak temperature would be approximately 30 °C after accounting for the ambient temperature. With a lower thermal conductivity of 0.25 W/m-K, the peak temperature would be approximately 42 °C. Aging has a strong impact on the peak temperature generated by Co-60 decay. Ten years less aging (two half-lives) would mean that the emplacement power density is four times greater, yielding a peak temperature of approximately 67 °C.

Table G-1-5 summarizes the peak values for each of the disposal and AMC spacing options shown in Figures G-1-5 through G-1-10. The vault approach, with the smaller surface area to volume ratio, generates considerably greater peak temperatures and corresponding gradients than the trench approach under the packed spacing option, while the borehole approach generates somewhat smaller peak values. The change in peak values from the packed-to-wide spacing is similar for the two hexagonal options, which increase the minimum AMC spacing by the same factor but do not change the disposal cell spacing. However, the change from packed to wide spacing has a smaller impact in the borehole option, which increases the spacing between disposal cells without changing the minimum AMC spacing.

Table G-1-5 Summary of Peak Values for Example Disposal and AMC Spacing Options

Case	Disposal scheme	Spacing	Source	Peak ϕ (m^2)	Peak ϕ' (m)	ϕ / ϕ(1)	ϕ' / ϕ'(1)	ϕ(wide) / ϕ(pack)	ϕ'(wide) / ϕ'(pack)
1	Trench	Packed	Figure 5	1.91	0.32	1	1		
2	Trench	Wide		0.54	0.14	0.28	0.44	0.28	0.44
3	Vault	Packed	Figure 7	3.07	0.56	1.61	1.75		
4	Vault	Wide	Figure 8	0.8	0.22	0.42	0.69	0.26	0.39
5	Borehole	Packed		1.49	0.26	0.78	0.81		
6	Borehole	Wide	Figure	0.86	0.22	0.45	0.69	0.58	0.85

The analytical approach does not consider a mixture of thermal conductivity values. With the disposal cell approach, the interior of the disposal cell will typically have a larger effective thermal conductivity than the earthen exterior, because of the conductive metal in the disposed waste. Cement-based materials (e.g., reinforced concrete pipe, cement walls, grout) may also have larger thermal conductivity than the surrounding exterior, or they may be insulators, depending on composition and density. Temperatures for the vault disposal option are likely most affected by material contrasts, because of the relatively long conduction path length to the exterior for the innermost AMCs; the AMCs are close to the exterior for trench and (especially) the borehole options. The wide-spacing option reduces the density of AMCs in the trenches and vaults by placing the AMC cylinders among non-heat-generating RH cylinders. Reducing the AMC packing density to approximately one in four of the concrete cylinders reduces the generated heat within the cell by a factor of 4, but temperature gradients inside the cell may not drop by a factor of 4. This is because the effective thermal conductivity also changes by replacing the highly conductive activated metal waste with some other waste.

The following is an example of how a licensee could use the figures in this attachment. Consider Figure G-1-7 (the packed vault case). Assume that (1) the earthen material has a thermal conductivity of 0.4 W/m-K, (2) the medium inside the vault has an effective thermal conductivity of 5 W/m-K, and (3) the emplacement power density inside the AMCs is 10 W/m^3. With the earthen material, the peak temperature increase is (3.2 m^2) × (10 W/m^3) / (0.4 W/m-K) = 80 °C, while at the wall it is (2 m^2) × (20 W/m^3) / (0.4 W/m-K) = 50 °C. With the effective medium, the temperature drop from peak to wall is (30 °C) × (0.4 W/m-K) / (5 W/m-K) = 2.4 °C, so that the peak interior temperature increase is 52.4 °C instead of 80 °C.

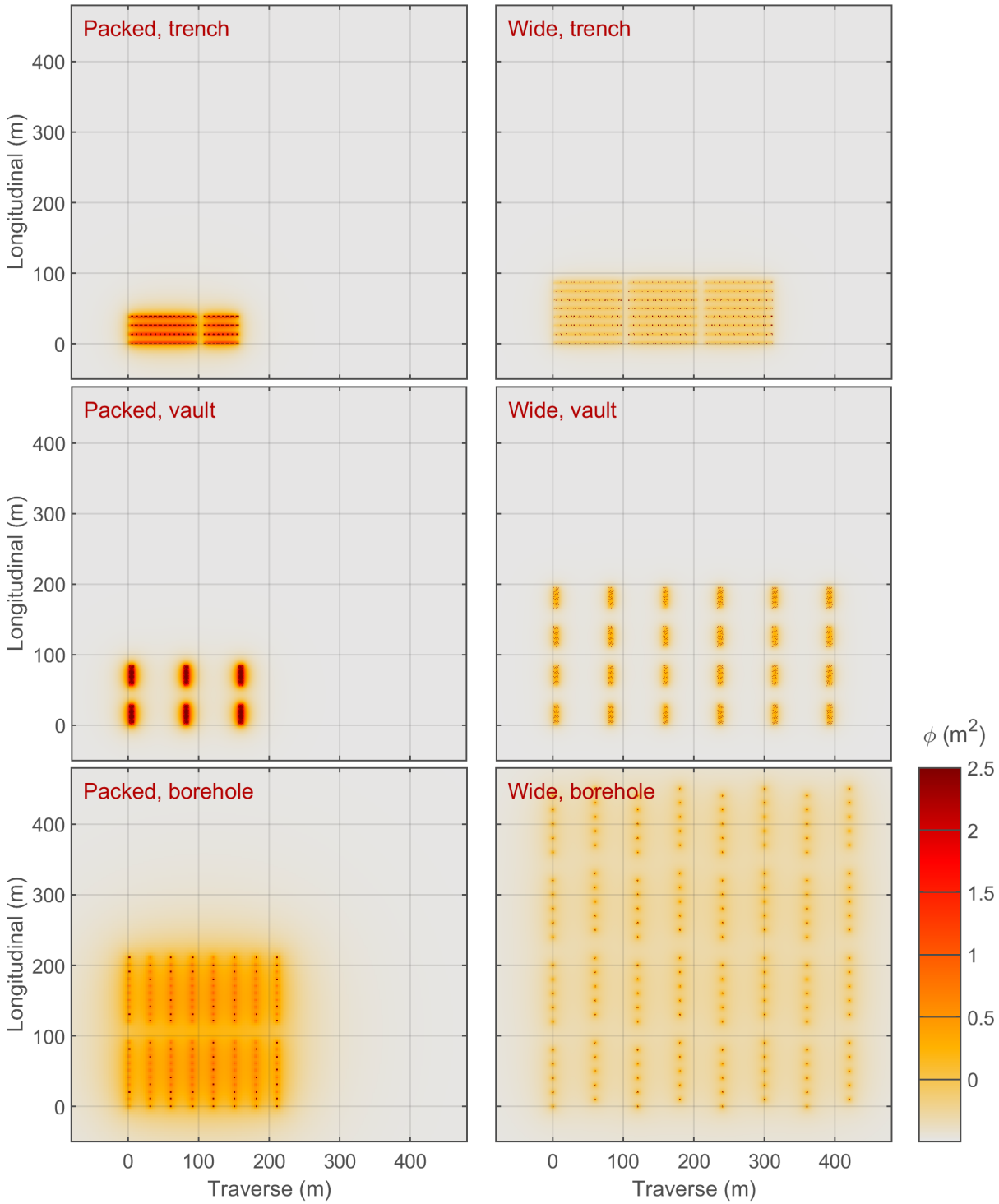
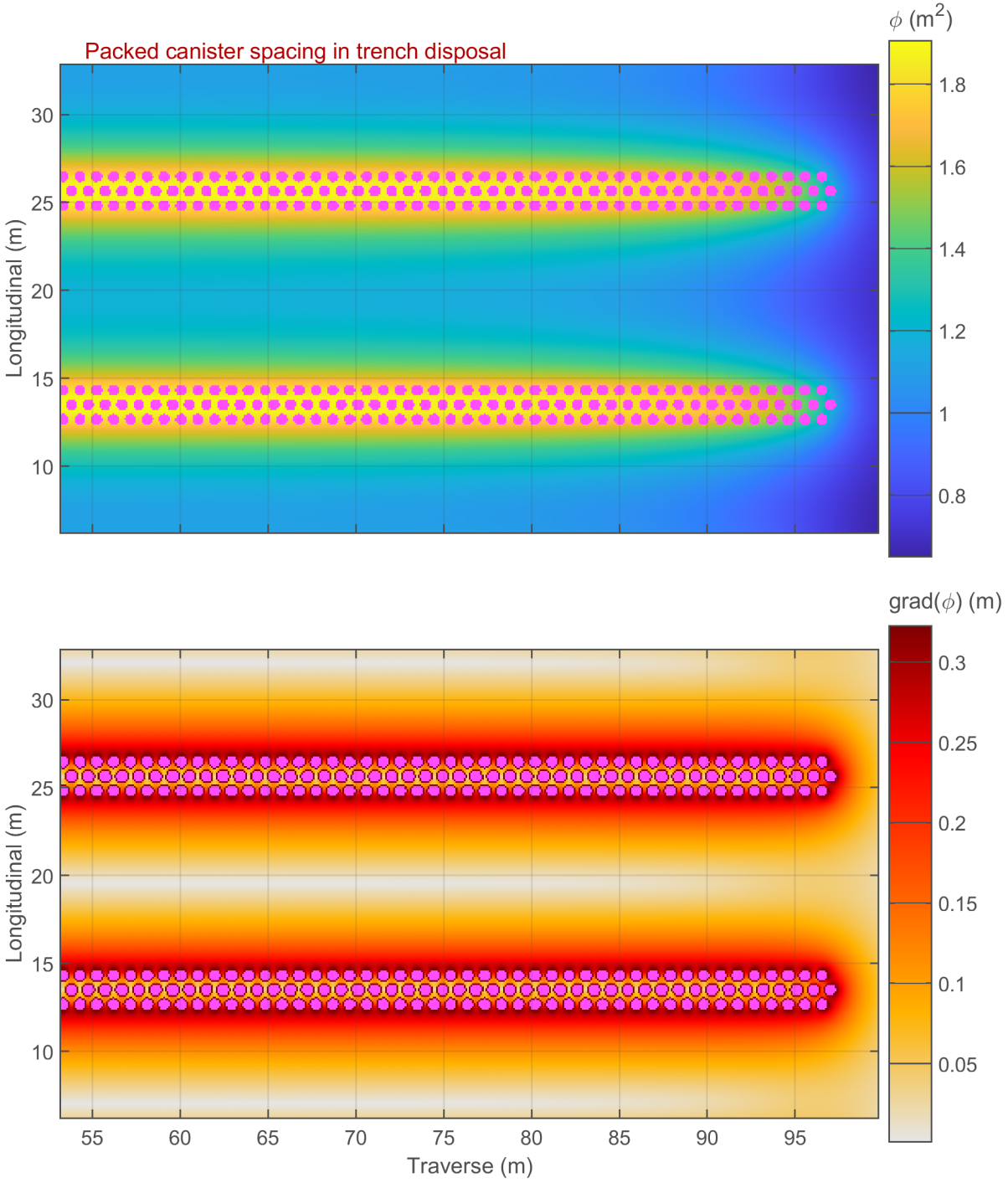


Figure G-1-4 Plan View of the Geometric Factor ϕ for Steady-state Temperature Deviation at the Vertical Center of the AMC Stacks for the Six Examples. The temperature deviation is recovered by multiplying ϕ by the power density inside the AMC and dividing by thermal conductivity

1



2

3 **Figure G-1-5 Plan View of the Geometric Factor ϕ (top) and its Horizontal Gradient**
4 **(bottom) for a Tightly Packed Trench Emplacement Configuration**

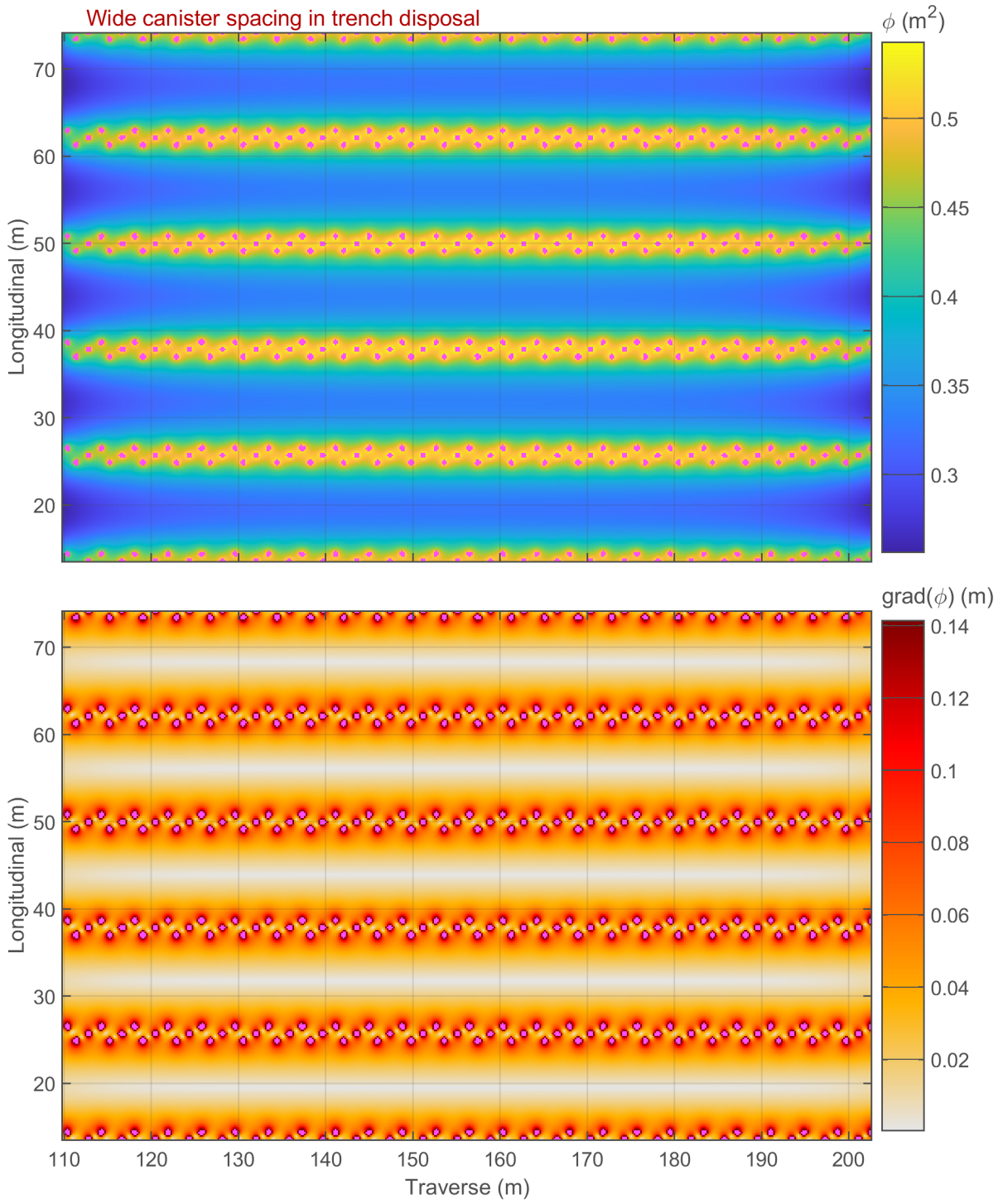


Figure G-1-6 Plan View of the Geometric Factor ϕ (top) and its Horizontal Gradient (bottom) for a Loosely Packed Trench Emplacement Configuration

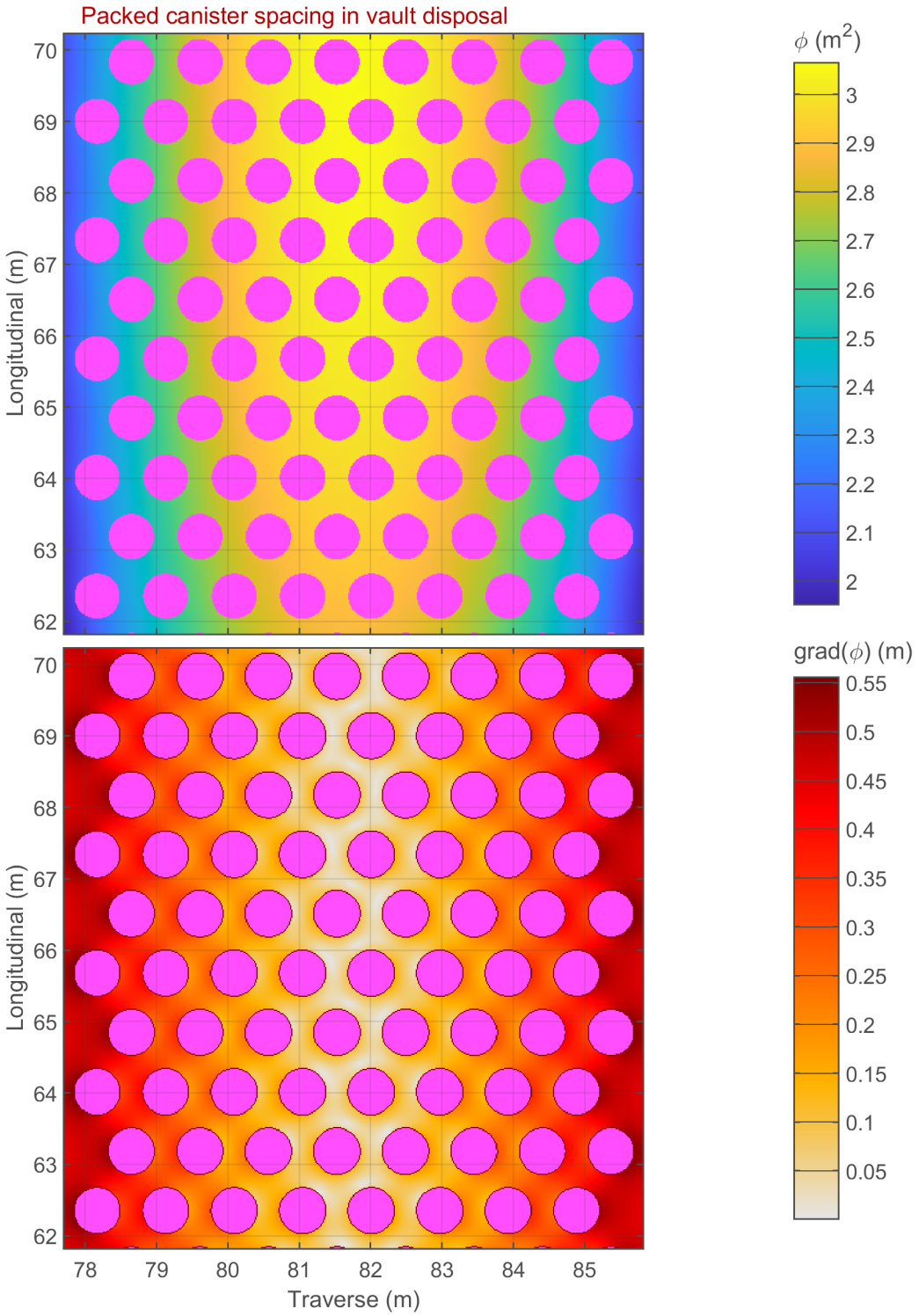


Figure G-1-7 Plan View of the Geometric Factor ϕ (top) and its Horizontal Gradient (bottom) for a Tightly Packed Vault Emplacement Configuration

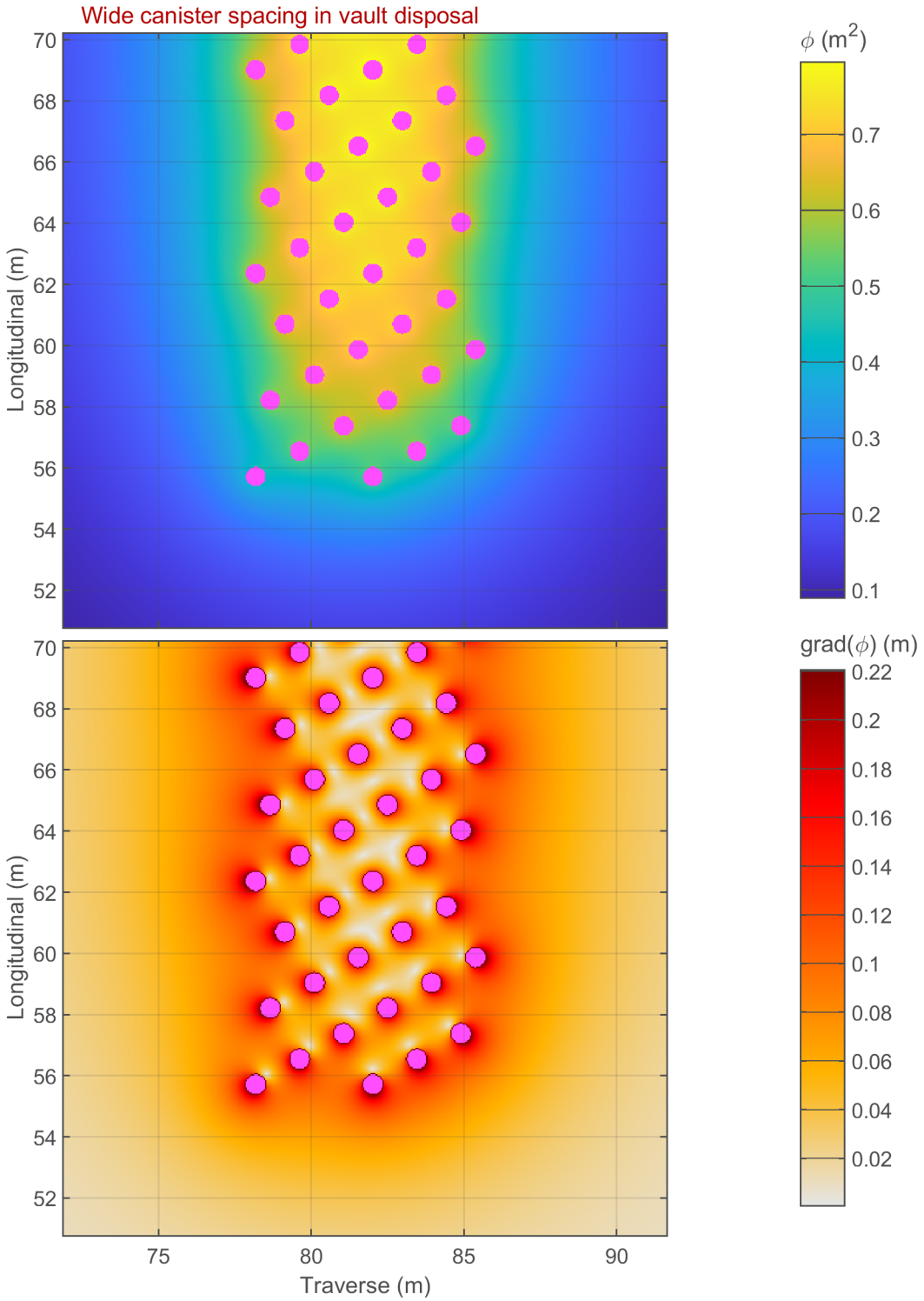


Figure G-1-8 Plan View of the Geometric Factor ϕ (top) and its Horizontal Gradient (bottom) for a Loosely Packed Vault Emplacement Configuration

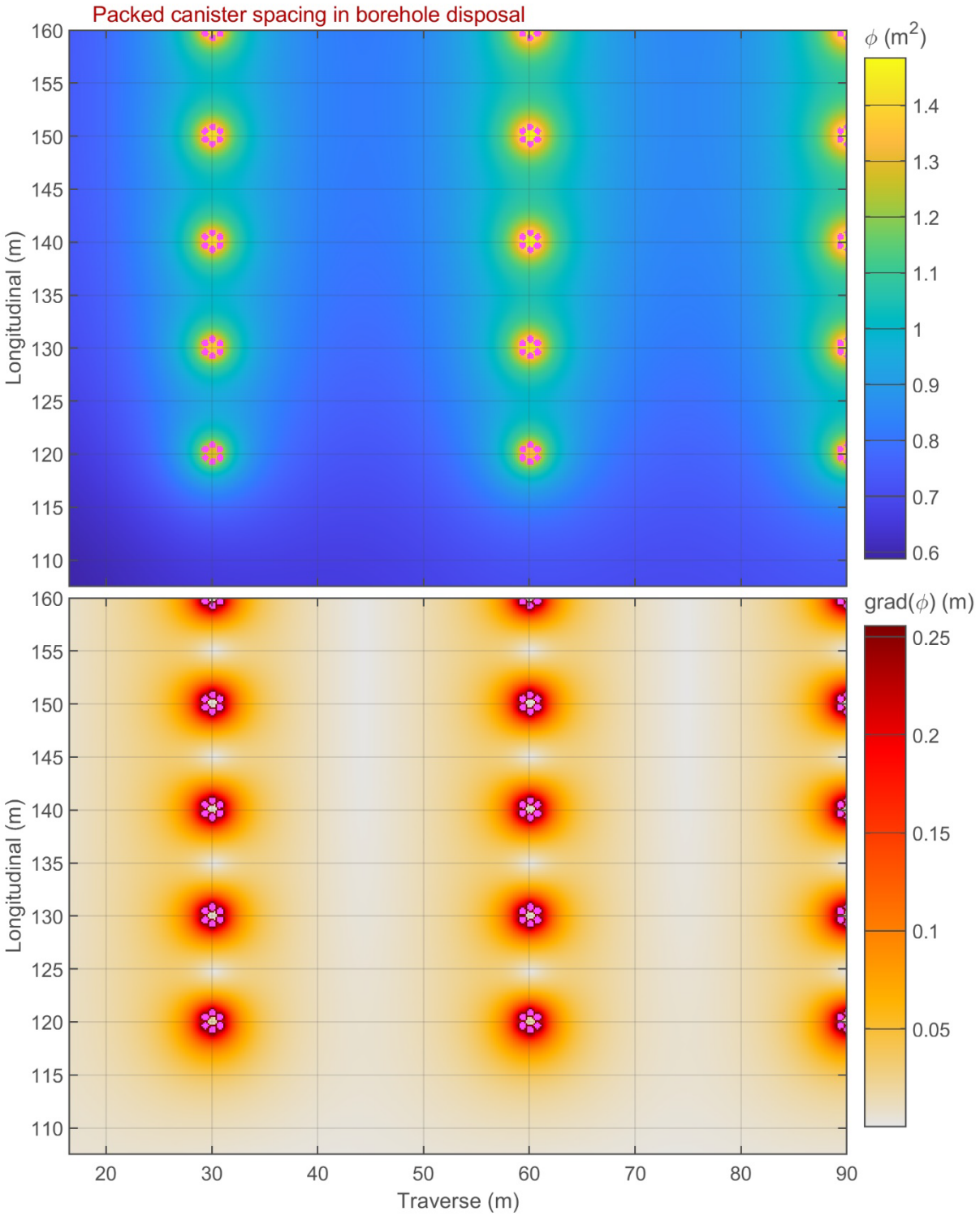


Figure G-1-9 Plan View of the Geometric Factor ϕ (top) and its Horizontal Gradient (bottom) for a Tightly Packed Borehole Emplacement Configuration

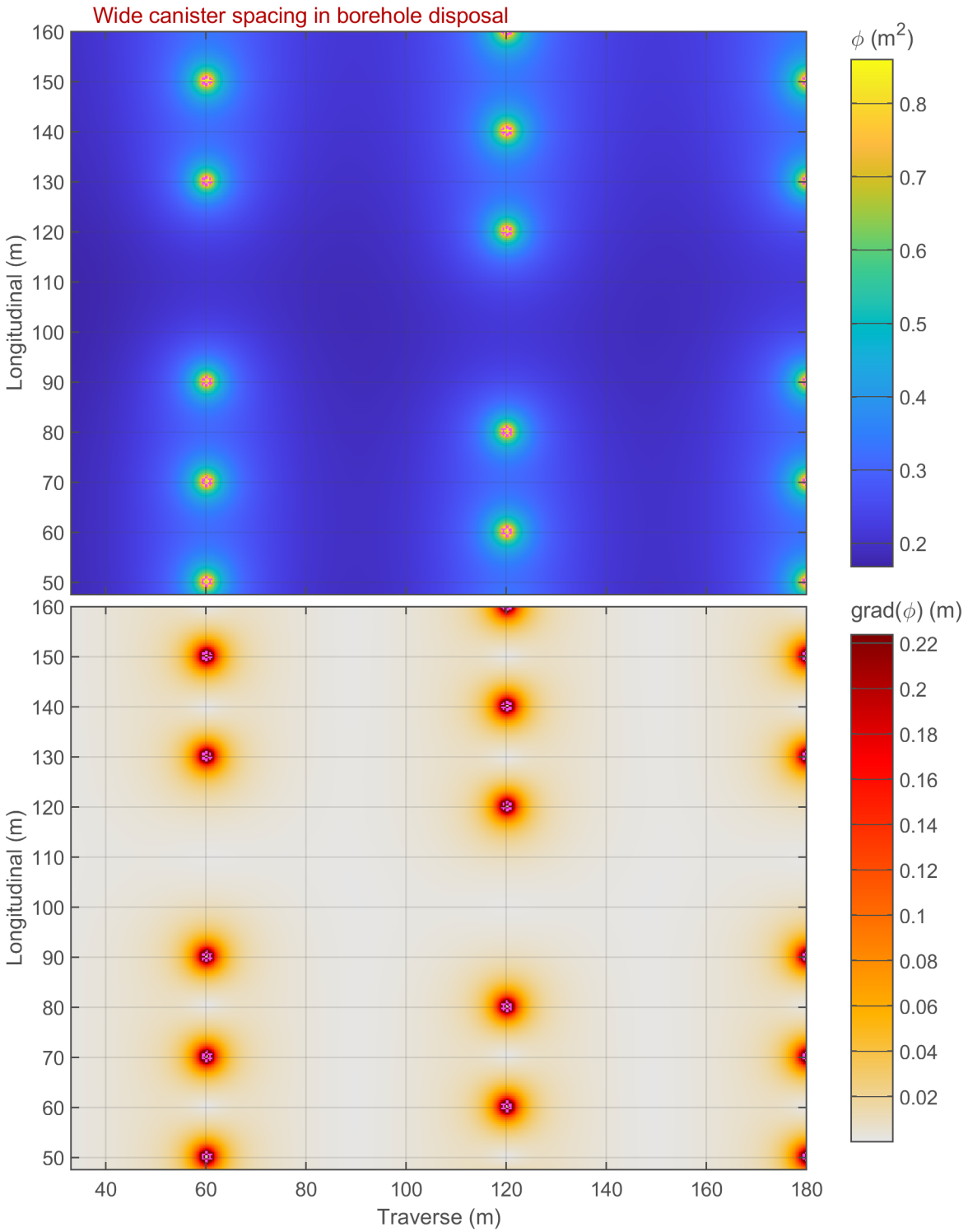


Figure G-1-10 **4Plan View of the Geometric Factor ϕ (top) and its Horizontal Gradient (bottom) for a Loosely Packed Borehole Emplacement Configuration**

ATTACHMENT G-2
HUMAN INTRUSION BARRIERS DEPLOYED AT COMMERCIAL LOW-LEVEL WASTE NEAR-SURFACE DISPOSAL FACILITIES

As of 2026, the United States has four operating commercial low-level waste (LLW) disposal facilities located in Richland, Washington (1965-present), Barnwell, South Carolina (1971-present), Clive, Utah (1991-present), and Andrews, Texas (2012-present). This attachment briefly describes these four facilities and discusses how some of the final disposal cover layers and concrete waste vaults may serve as barriers to inadvertent intrusion by domestic water well drilling.

G-2.1 US Ecology Richland Disposal Facility

The US Ecology LLW disposal facility on the Hanford Reservation near Richland, Washington, accepts Class A, B, and C LLW. The Richland facility was licensed in 1965 by the State of Washington, an Atomic Energy Commission (AEC) Agreement State. The facility currently has no human intrusion barriers. However, in the "Final Environmental Impact Statement: Volume I Commercial Low-Level Radioactive Waste Disposal Site Richland, Washington," (WDOH, 2004), four final cover design alternatives are identified for the trenches that comprise the commercial LLW facility: (1) Cover Design-No Action, consisting of 2.4 to 3.4 meters (m) (8 to 11 feet (ft)) of site soils, (2) US Ecology Cover, consisting of a 4.9 m (16 ft) thick multilayer cover that includes a 91 centimeter (cm) (36 inch (in)) surface silt loam layer and a low-permeability bentonite layer, (3) Homogeneous Cover, consisting of a 1.8 m (6 ft) thick soil cover with a 152 cm (60 in) surface silt loam layer, and (4) the Enhanced Cover, consisting of a 152 cm (60 in) silt loam layer and a low-permeability barrier consisting of (a) a 31 cm (12 in) asphalt barrier, or (b) a geotextile clay barrier, or (c) a 31 cm (12 in) bentonite barrier. WDOH (2004) states that a cover that is 5 m (16.4 ft) thick or more is "…expected to be effective at preventing direct contact by an inadvertent intruder." Decisions for the final cover design have been made and the facility is expected to close in the 2050s.

G-2.2 Energy*Solutions* Barnwell Disposal Facility

The Energy*Solutions* Barnwell Disposal Facility near Barnwell, South Carolina accepts Class A, B, and C LLW. The Barnwell facility was licensed in 1971 by the State of South Carolina, an AEC Agreement State. The total thickness of the Barnwell cover, as designed, is approximately 1.5 m (5 ft), and none of the layers that comprise the cover would appear to be effective at deterring human intrusion. In 1989, the then-operator of the Barnwell facility, ChemNuclear Systems, developed a concrete disposal vault for Class C waste that the South Carolina Department of Health and Environmental Control approved as an engineered intruder barrier. This will allow the Barnwell facility to continue to dispose of Class C waste in trenches without a closure cover that is at least 5 m (16.4 ft) thick. The concrete vaults ensure that the closure cover is well supported and has increased structural stability. Three types of concrete vaults are used: (1) 3 m long by 3 m wide by 3.4 m tall (10 ft × 10 ft × 11 ft) rectangular vaults for metal boxes and drums, (2) 2.4 m diameter by 2.7 m tall (8 ft × 9 ft) cylindrical vaults, and (3) 5.2 m long by 1.5 m wide by 1.4 m tall (17 ft × 5 ft × 4.5 ft) vaults for slit trenches.[10]

[10] https://www.scdhec.gov/sites/default/files/docs/HomeAndEnvironment/Docs/commercial_low_level.pdf (accessed 2/10/2020)

According to Baird et al. (2007), the lids of the vaults (also called canisters) used at the Barnwell facility are 41 cm (16 in) thick, constructed with 27.6 megapascals (4,000 pounds per square inch) concrete with two #5 (5/8 in) reinforcing layers placed on 15.2 cm (6 in) centers in both horizontal directions. The lids of the vaults appear to be placed slightly above the grade of the disposal trenches, suggesting that the 1.5 m (5 ft) closure cover rests directly on the lids or on the low-permeability soil and dry sand placed in the vertical gaps between the vaults that are greater than 30.5 cm (12 in). The South Carolina Department of Health and Environmental Control approved the vaults as engineered intruder barriers because of the 41 cm (16 in) reinforced concrete lids.

G-2.3 Energy*Solutions* Clive Disposal Facility

The Energy*Solutions* Clive Disposal Facility near Grantsville, Utah, was granted a license in 1991 by the State of Utah, a U.S. Nuclear Regulatory Commission (NRC) Agreement State. The Clive facility only accepts Class A LLW, so it is not required to have a cover that is at least 5 m (16.4 ft) thick or a human intrusion barrier that is effective for 500 years under Title 10 of the *Code of Federal Regulations* (10 CFR) 61.52(a)(2). The total thickness of the currently planned Clive cover is approximately 1.7 m (5.5 ft), and the 46 cm (18 in) thick erosion layer composed of cobbles less than 11.4 cm (4.5 in) in diameter may discourage some drillers from starting a water well.

G-2.4 Waste Control Specialists Texas Compact Waste Facility

The Waste Control Specialists (WCS) Texas Compact Waste Facility (CWF) is licensed to accept Class A, B, and C LLW for disposal at a near-surface disposal facility near Andrews, Texas. The CWF was licensed in 2012 by the State of Texas, which is an NRC Agreement State.

The facility currently has no human intrusion barriers. However, the final cover design for the CWF ranges in thickness from 8.7 to 13.7 m (25 to 45 ft) thick. In addition, the WCS performance assessment (WCS, 2007) showed the maximum dose from inadvertent intrusion is less than 0.07 millisievert per year (7 millirem per year). WCS (2007) noted that the 0.9 m (3 ft) thick bio-intrusion barrier composed of cobbles, the 0.3 m (1 ft) thick reinforced shotcrete layer, and the modular concrete canisters (MCCs) used for in-cell disposal will provide additional protection against inadvertent intruders. Cylindrical MCCs are approximately 1.8 m (6 ft) in diameter, 3 m (10 ft) tall, and they have a wall thickness of 0.3 m (1 ft).[11] Information in van Vliet (2012) indicates that the concrete wall, bottom, and lid of the MCC will be reinforced with double mats of #4 and #6 epoxy-coated rebar. It is unknown whether the capacity of the reinforced MCC lids to deter domestic water well drilling has been tested or demonstrated.

In WCS (2007), the licensee stated, on page 8-55, that the total cover system will be between 7.6 and 13.7 m (25 and 45 ft) thick. According to WCS (2007), the licensee took credit for (1) the overall thickness of the cover (7.6 to 13.7 m (25 to 45 ft)) as a deterrent to human intrusion by excavation, and (2) the layering of the cover system (especially the cobble bio-barrier and the red bed fill) being recognized by a water well driller as layers of an engineered cover system, which should be a deterrent to continued drilling. License conditions have been used to ensure that more hazardous waste is placed deeper within disposal cells to reduce the likelihood of contact by an inadvertent intruder.

[11] https://www.tceq.texas.gov/assets/public/comm_exec/pubs/sfr/104-16.pdf (accessed 2/10/2020)

APPENDIX H
DETERMINATION OF SIGNIFICANT QUANTITIES OF LONG-LIVED RADIONUCLIDES

The performance objectives in Title 10 of the *Code of Federal Regulations* (10 CFR) 61.41, "Protection of the general population from releases of radioactivity," and 10 CFR 61.42, "Protection of individuals from inadvertent intrusion," specify a period of time over which protection of the public and an inadvertent intruder must be achieved, or the "compliance period." The definitions in 10 CFR 61.2, "Definitions," include the following:

> *Compliance period* means the time from the completion of site closure to 1,000 years after site closure for disposal sites that do not contain significant quantities of long-lived radionuclides. For disposal sites that contain significant quantities of long-lived radionuclides, the compliance period ends 10,000 years after closure of the disposal site.
>
> *Significant quantities* of long-lived radionuclides means an amount (volume or mass) and concentration accepted for disposal after the [30 DAYS AFTER DATE OF PUBLICATION OF THE FINAL RULE IN THE *FEDERAL REGISTER*] that could, if released, result in the performance objectives of subpart C of this part not being met.
>
> *Long-lived radionuclide* means radionuclides where: (1) more than 10 percent of the initial activity of the radionuclide remains after 1,000 years; (2) the peak activity from progeny occurs after 1,000 years; or (3) more than 10 percent of the peak activity of the radionuclide (including progeny) that occurs within 1,000 years remains after 1,000 years.

To determine which compliance period is appropriate for use at each disposal site and to determine whether an existing licensee meets the criteria in 10 CFR 61.1(b), the quantity of long-lived radionuclides must be assessed. A significant quantity, expressed in units of mass or radioactivity (e.g., curies (Ci) or becquerels (Bq)) is generally the product of amount (volume or mass) and concentration. The significance is determined by the propensity of the material to cause the 10 CFR 61.41 and 10 CFR 61.42 performance objectives not to be met. The following sections provide guidance on how licensees may determine and how reviewers should evaluate what constitutes a "significant" quantity of long-lived radionuclides, and how this calculation enables selection of the appropriate compliance period to be used in the licensee's technical analyses as well as determines whether licensees meet the criteria in 10 CFR 61.1(b).

As of 2026, all operating commercial low-level radioactive waste (LLW) disposal facilities are in States where Agreement States regulate waste disposal. Based on compatibility requirements, the Agreement States have the option to be more restrictive than U.S. Nuclear Regulatory Commission (NRC) regulations in terms of establishing a compliance period. If a 10,000-year or longer compliance period is adopted for the disposal of all LLW, then the guidance provided in the remainder of this appendix does not need to be implemented. A licensee must consult with the appropriate Agreement State regulator and implement the regulations in the State where the facility is located. Existing licensees that meet the criteria in 10 CFR 61.1(b) are not required to implement several of the revised licensing requirements for LLW disposal included in the

Integrated Low-Level Radioactive Waste Disposal (ILLRWD) rule (see Figure H-4 for a more detailed process to determine compliance period for depleted uranium (DU) disposal). For those licensees that are required to implement the revised requirements, the process is to start with simpler approaches and progress to more detailed analyses for determining significant quantities of long-lived radionuclides.

H.1 Background

LLW is typically a mixture of both short- and long-lived radionuclides. For an offsite member of the public, much of the radiological risk is from the long-lived radionuclides because the short-lived radionuclides will decay in place in most modern disposal facilities. For a member of the public who inadvertently uses the site after the institutional control period, much of the radiological risk is from the short-lived radionuclides. The approach adopted by the NRC in 10 CFR Part 61, "Licensing Requirements for Land Disposal of Radioactive Waste," is a risk-informed requirement to account for differences in waste, disposal facilities, and disposal sites. The requirement is to demonstrate that public health and safety are protected from the disposal of all types of LLW. This guidance accounts for the differences between long-lived and short-lived waste. DU is long-lived and may be both concentrated and have significant volume. The radiological decay of the uranium into progeny radionuclides with different properties (e.g., radon) creates unique challenges for assessing the disposal of significant quantities of DU. Some Greater-Than-Class-C (GTCC) waste streams may be comprised of mostly short-lived radionuclides, whereas others may be long-lived or include a mixture. The technical analyses performed should be consistent with the characteristics of the waste, accounting for the interaction of the waste with the disposal system.

Radiological risk is a product of both the amounts and the concentrations of radionuclides in waste as well as the properties of the radionuclides and how that radioactivity is contained in a disposal system. Additional considerations include who will be exposed and when that exposure will occur. Some radionuclides are much more mobile in the environment than others. The time between the release of radioactivity within a disposal system and exposure of a receptor can be delayed by a variety of phenomena, such as containment by engineered barriers and sorption in the geosphere. These delays can be hundreds or thousands of years or longer.

Disposal of LLW will only be permitted at a location where it can be demonstrated that the regulatory requirements will be satisfied. The regulations in 10 CFR 61.50, "Disposal site suitability requirements for land disposal," describe the characteristics that a site must possess. Some disposal sites may be inherently better suited to contain radioactivity or certain types of waste. Complexity arises because of the variety of wastes and disposal site environments. For instance, a site may not have potable groundwater. Should the compliance period for the disposal of long-lived waste be the same for a site with potable water as for one that currently does not have potable water? Other usage of water should be accounted for, but consumption of contaminated water is generally one of the largest contributors to dose. To account for the variability in wastes and disposal site environments, the approach recommended in this appendix is to start with screening calculations and, if necessary, to progress to more detailed analyses to establish whether a site will receive significant quantities of long-lived waste. A practical approach is needed to perform this iterative process. If the screening evaluation becomes complex or detailed, then it is likely that the compliance calculations will have already been performed and a longer compliance period should be used. If the screening is a simple calculation, then it may be appropriate to use a 1,000-year compliance period.

The NRC staff recommends the following steps to determine whether the quantities of long-lived waste disposed will be significant:

1. Use generic concentration[1] screening values.
2. Perform simplified dose screening analyses.
3. Conduct site-specific dose analyses.

Figure H-1 is an overview of the process to determine timeframes for the analyses as described in this appendix, although if the waste is DU, Figure H-4 is better suited for determining the compliance period for that waste. At any point in this process a licensee can elect to use a longer compliance period to demonstrate that the performance objectives will be met. Because all active commercial LLW disposal facilities are in NRC Agreement States (as of 2026), the Agreement State regulators perform the reviews and issue the licensing decisions for waste disposal, except for GTCC waste disposal, which will be licensed by the NRC. For disposal of Class A, B, and C wastes licensed by an Agreement State, the regulatory staff should review the basis for the compliance period used by the licensee in its analyses. The review will be commensurate with the detail for each evaluation. Evaluations relying on significant assumptions or involving more complex disposal sites (e.g., intricate water pathways in the unsaturated and saturated zones) may require more detailed review. Selection of a compliance period may consider many different factors that are technical, policy, and socioeconomic.

In accordance with 10 CFR 61.1(b):

> (1) Licensees need not comply with the requirements in §§ 61.10(c), 61.13(a) through 61.13(e),61.24(l), 61.41(a) and (b), 61.42(a) and (b), 61.50(a) and (b), and 61.58, if the following criteria are met:
>
> > (i) the land disposal facility license was originally issued before [30 DAYS AFTER DATE OF PUBLICATION OF THE FINAL RULE IN THE *FEDERAL REGISTER*]; and
> >
> > (ii) the licensee does not accept Greater-Than-Class C waste or a significant quantity of long-lived radionuclides. For purposes of this paragraph, an amount greater than or equal to 10 metric tons of depleted uranium is considered a significant quantity of long-lived radionuclides.
>
> (2) Licensees who meet the criteria of § 61.1(b)(1)(i) and (ii) and who choose not to comply with the requirements in §§61.10(c), 61.13(a) through (e), 61.24(l), 61.41(a) and (b), 61.42(a) and (b), 61.50(a) and (b), and 61.58 must comply with §§ 61.13(f), 61.41(c), 61.42(c), and 61.50(c).

Therefore, licensees that meet the criteria in 10 CFR 61.1(b)(1) may elect to continue to comply with 10 CFR 61.13(f), 10 CFR 61.41(c), 10 CFR 61.42(c), and 10 CFR 61.50(c). These sections duplicate the original 10 CFR Part 61 regulations prior to the ILLRWD rule. These licensees may also continue to use the timeframes in their analyses deemed acceptable by the appropriate regulator prior to the current revisions to 10 CFR Part 61. LLW disposal facilities

[1] Risk is the product of mass or volume and concentration. The concentration screening values have an assumed volume of waste, and a licensee may modify the assumed values for its evaluation.

that begin operations after revision of 10 CFR Part 61 must meet the revised requirements regardless of the quantities of long-lived radionuclides planned for disposal.

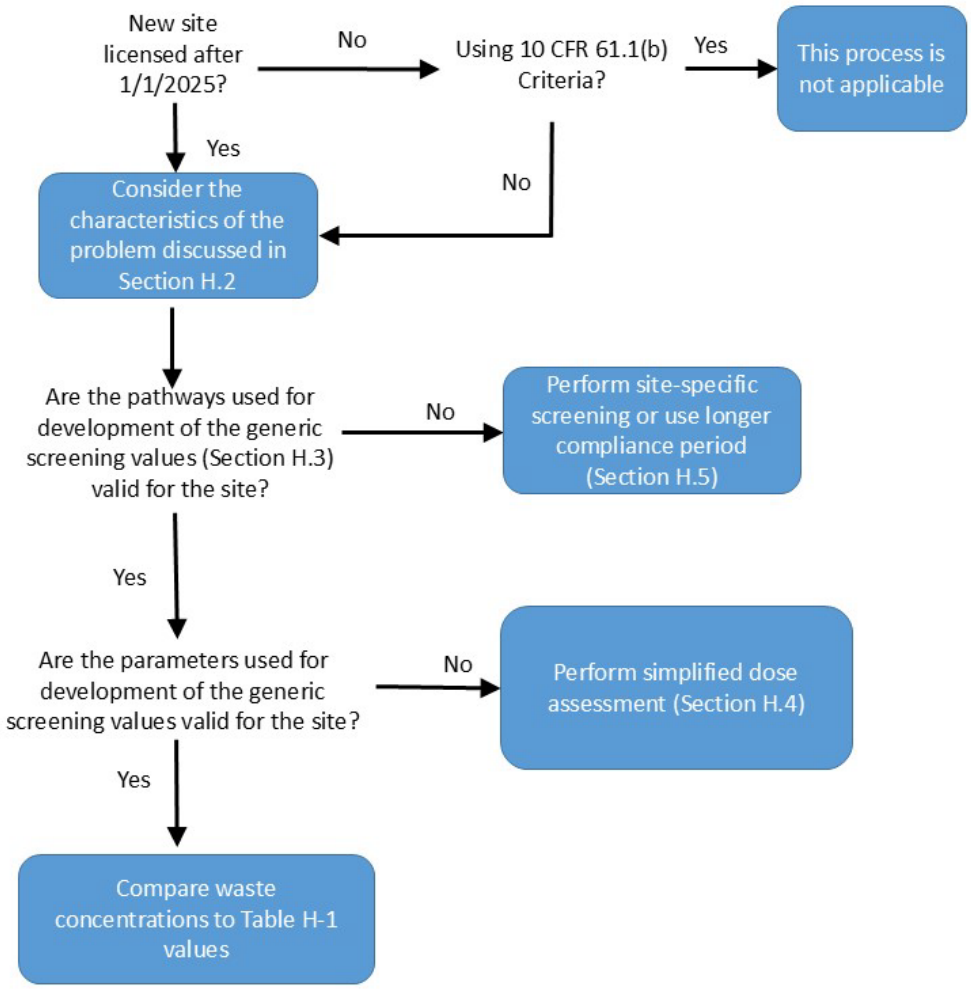


Figure H-1 Process for Selecting the Compliance Period (Significant Quantity of Long-lived Radionuclides Evaluation)

The following sections provide general guidance on how to determine site-specific amounts that could be considered a significant quantity of long-lived radionuclides at a given disposal site, as an alternative to the 10 metric tons (MT) of DU provided in 10 CFR 61.1(b)(1)(ii) and for other long-lived waste. This site-specific significant quantity amount should be treated as the threshold above which waste disposals should be evaluated using technical analyses with a 10,000-year compliance period to determine compliance with 10 CFR 61.41(a) and 10 CFR 61.42(a). The goal is to perform an analysis with an appropriate compliance period for the waste that is being (or is proposed to be) disposed. The approach outlined in this appendix is sufficiently flexible to account for different site characteristics and different types of long-lived waste that may be generated because of future nuclear energy development. To minimize regulatory burden, a screening process is provided. The licensee must have sufficient basis for the analyses performed. The NRC staff should review the technical basis provided to ensure the data used were appropriate and that the assumptions made in the development of generic screening values are consistent with the specific site being evaluated.

The purpose of completing the steps outlined in Figure H-1 is to determine whether the proposed disposal site will accept significant quantities of long-lived radionuclides or DU such that a longer (i.e., 10,000-year) compliance period should be used in the technical analyses rather than the 1,000-year compliance period suitable for short-lived radionuclides and limited quantities of long-lived radionuclides.

H.2 Site-Specific Characteristics to Consider

The NRC staff recommends that licensees consider, and reviewers evaluate, the following site-specific factors when performing their evaluation to determine what quantities of long-lived radionuclides would be considered significant:

- disposal volume
- waste concentrations
- wasteform and geochemical environment
- disposal depth
- engineered barriers
- precipitation
- groundwater

The site-specific characteristics determine the impacts from releases of radioactivity to the environment. For long-lived waste without significant ingrowth of risk-significant daughter radionuclides, the impacts to potential inadvertent intruders (under 10 CFR 61.42) generally decrease with time after the institutional control period. However, for long-lived waste the impacts to a member of the public under 10 CFR 61.41, which are generally from transport in groundwater, can be delayed because of the implementation of engineered and natural barriers. Figure H-2 shows a cross-sectional view of a waste disposal site and outlines some of the features and processes that influence radiological impacts.

Though the timing of impacts for long-lived waste may be significantly delayed, the magnitude of the impacts may remain unchanged. For short-lived waste, the impact of delays can be very significant due to radiological decay. Delays allow short-lived radionuclides to decay before being transported closer to the hypothetical receptors.

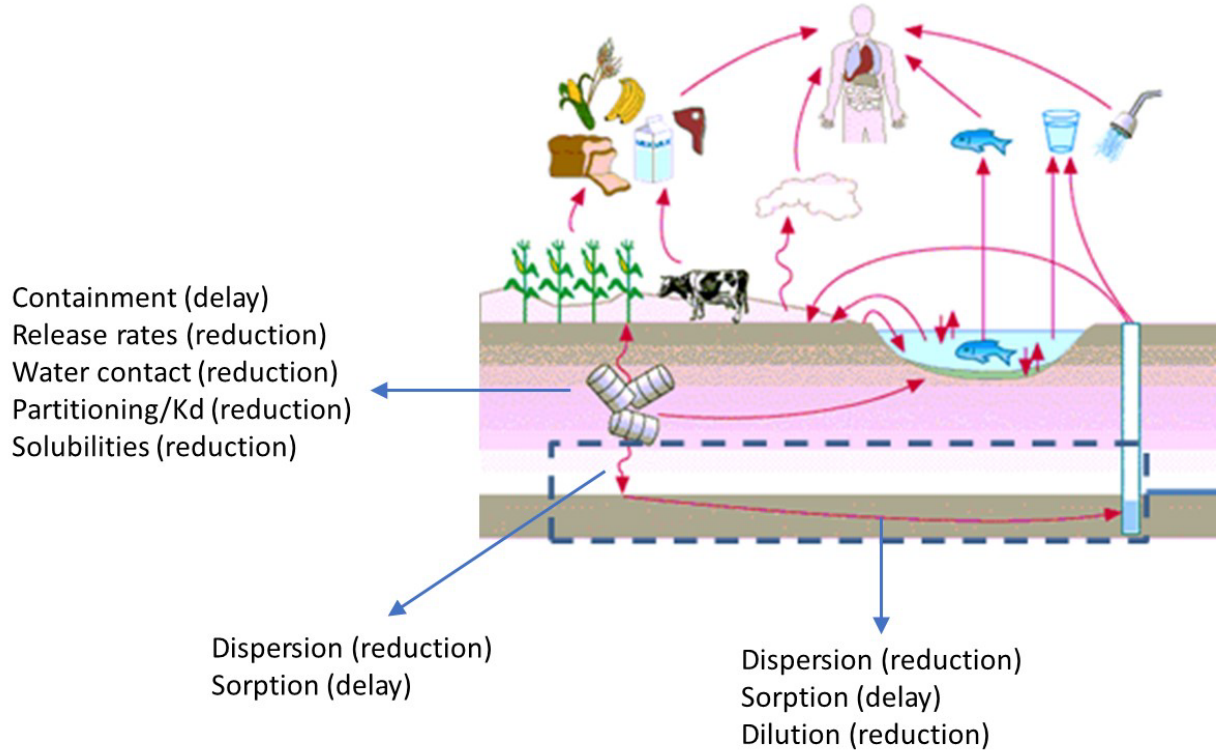


Figure H-2 Examples of Processes Influencing the Impacts from LLW Disposal

For long-lived waste, the more important features and processes to account for in a screening process are those that result in reductions rather than delays. The purpose of the evaluation of site characteristics is to identify those features, events, and processes likely to impact the timing and magnitude of potential doses. As previously noted in this guidance, some of the factors to consider, such as climate and groundwater, may change over time. Simplified screening analyses (see Section H.4) may use reasonably pessimistic assumptions to account for changes over time and avoid unnecessary complexity. Delays in radiological impacts are good but reduction in the magnitude of eventual impacts is better.

H.2.1 Disposal Volume

The volume of waste should be based only on the waste streams containing long-lived radionuclides and not the total volume of waste disposed in the facility. For small disposal volumes, impacts to groundwater will be generally proportional to the volume of waste. When a receptor pumps groundwater for usage, a groundwater capture radius is formed around the well at the water table of the underlying aquifer. The size of that capture radius is dependent on the characteristics of the withdrawal as well as the properties of the aquifer. The capture radius can be larger than the plume, which results in groundwater impacts being proportional to disposal volume, up to a certain amount. When the plume is significantly larger than the capture radius, the results are no longer sensitive to disposal volume. A conservative assumption is that all releases from the disposed waste are captured by a well withdrawing water for use; however, that assumption may not be appropriate for disposal of large amounts of waste. Impacts to air or releases to the land surface would be similarly proportional to disposal volume. Disposal volume will not generally impact the timing of releases.

H.2.2 Waste Concentrations

The radiological impacts from the disposal of LLW are in many cases dependent on the concentration of long-lived radionuclides in the waste, but in some cases they may be independent. Potential impacts to inadvertent intruders are generally directly proportional to waste concentrations, but the impacts to offsite receptors may or may not be directly proportional to waste concentrations, depending on the radionuclide(s) and disposal system. For instance, the release of a radionuclide may be solubility-limited such that the impacts are independent of waste concentration as long as the waste concentration is above the solubility limit. Dose impacts will be generally proportional to waste concentrations with some exceptions, but waste concentrations will not generally impact the timing of releases.

H.2.3 Disposal Depth

The depth of disposal of waste is important for the disposal of long-lived radionuclides because increasing depth does the following:

- decreases the likelihood that an inadvertent intruder will contact the waste
- decreases the likelihood that surficial processes will disturb the waste or engineered barriers
- decreases the likelihood that biota will contact the waste
- decreases the impacts from some gaseous radionuclides such as radon

Depth of disposal is a key element to reduce uncertainties associated with the long-term assessment of impacts from the disposal of GTCC waste containing long-lived radionuclides and DU. In a screening assessment to determine whether the proposed disposal constitutes a significant quantity of long-lived radionuclides, the influence of depth of disposal can be difficult to quantify. For DU and GTCC waste containing long-lived radionuclides, depth of disposal is a key design consideration. If the waste will be disposed deeper, then the amount that constitutes a “significant quantity” would likely be greater as compared to values associated with shallower disposal. This is because deeper disposal decreases the potential risk to the intruder who excavates a basement and builds a house above the disposed waste. An intruder who drills a well transports less waste to the land surface than an intruder who digs a house foundation into the waste. Increasing the waste disposal depth also decreases the likelihood of disruption of the waste from other processes.

For DU disposal, a greater disposal depth decreases the flux of radon to the receptor due to greater attenuation by the geology of the disposal site. The required depth of 5 meters (m) (16.4 feet (ft)) from the land surface is larger than the traditional value of 3 m (10 ft) used to distinguish between an intruder-construction scenario and an intruder-drilling scenario because the decay of uranium results in production of radon. For most sites, the extra 2 m (6 ft) of subsurface material between the waste and the potential excavation would be needed to reduce radon emissions. At arid or semi-arid sites (i.e., those with less than 20 centimeters (cm) (8 inches (in) per year annual rainfall), more subsurface material (greater than an extra 2 m (6 ft)) may be necessary to effectively reduce doses from radon. The required depth for DU disposal is to be determined from the minimum of the preexisting land surface or the surface of

the land after development of the site, and not from the top surface of the cover at closure. This is to help ensure that the waste will be protected from long-term surficial processes.

H.2.4 Engineered Barriers

Engineered barriers are used in LLW disposal to delay or reduce the release of radioactivity. Barriers that are used include, but are not limited to, containers, overpacks, backfill, vaults, intruder barriers, liners, drains, sumps, engineered covers, and wasteforms or other stabilization materials.

For the disposal of long-lived waste, a licensee should start by listing the engineered barriers that will be used and their prospective performance. The licensee's determination that the quantity of long-lived waste is significant should generally not consider engineered barriers that delay when releases to the environment may occur. This is because barriers that delay impacts (while very important for short-lived waste) will not decrease impacts associated with long-lived waste disposal. For example, a licensee proposes to put waste in a steel container that the licensee assumes will last more than 1,000 years. In a screening analysis, there would be no calculated impacts from the waste in the next 1,000 years. The licensee may conclude that the quantity of long-lived waste will not impact a member of the public and that a longer compliance period is not necessary; however, the NRC staff may not agree with this conclusion. The waste itself is not inherently low risk; rather, the engineered barriers are being used to delay or reduce the risk.

For engineered barriers used to reduce releases, the barriers may be relied upon to limit the impacts from the release of long-lived radionuclides. For these types of barriers, using a 1,000-year compliance period may be sufficient, but the technical basis for achieving long-term reductions in releases beyond 1,000 years must be provided. If assurance of the long-term reduction in releases cannot be given, then the technical analyses should use the 10,000-year compliance period followed by the performance period to evaluate what happens when the barrier no longer limits releases.

H.2.5 Wasteform

To reduce the impacts from the disposal of long-lived radionuclides, robust wasteforms may be used. A robust wasteform may have multiple performance benefits, including, but not limited to, reducing the contact of water with the radioactivity, reducing the rate of diffusion of radioactivity from the waste, controlling the chemical environment for release, and inhibiting the transport of released radioactivity in the disposal system. The performance of a wasteform may contribute significantly to performance of the disposal system.

For most long-lived radioactive waste, the magnitude of the reduction in release rates that a wasteform must provide is generally insensitive to the analyses' timeframes. When performing screening analyses, it is useful to evaluate potential doses with normal and significantly degraded wasteform performance to identify the contribution of the wasteforms.

H.2.6 Precipitation

The climate at a site is an important consideration when determining whether significant quantities of long-lived radionuclides will be disposed, with precipitation being the most important component of the climate. Releases to groundwater will be strongly influenced by the

amount of water that reaches the waste and the transport of released radioactivity out of the disposal units. The amount of water that infiltrates the land surface and eventually recharges the aquifer is driven in part by the amount of precipitation. In general, the quantity of long-lived radionuclides that may be significant at a site with high precipitation may be less than at a site with low precipitation. For most long-lived radionuclides, the most significant impacts generally occur to water pathways rather than air pathways.

For the disposal of DU, impacts may occur to water pathways or air pathways. Though uranium in oxide form is rather insoluble, given the disposal of significant concentrations and quantities, uranium can be leached to infiltrating water to produce significant impacts if the dilution in the aquifer underlying the disposal facility is not large enough. DU decays and eventually produces radon gas. The transport of radon gas in the subsurface is strongly dependent on the moisture present in the system. Under high moisture conditions, the diffusion of radon gas is slower, and it decays during transport. Under low moisture conditions the diffusion of radon gas is more rapid, and it can reach the accessible environment in greater concentrations.

H.2.7 Groundwater

The presence of groundwater, how it is used, and the hydrological characteristics of an aquifer are key characteristics to consider when evaluating the disposal of long-lived radionuclides. A disposal site may have potable groundwater or groundwater that is used for other purposes. In general, if a disposal site does not have potable groundwater and the site is expected to not have potable water in the future, then the quantity and concentration of long-lived radionuclides that are considered significant will be larger than at a site that does have potable groundwater. Usage of water for purposes other than drinking should be considered. It is possible that other pathways may be the most significant release pathway (e.g., radon gas from the disposal of DU), but the groundwater pathway is generally the most significant pathway for the disposal of long-lived radionuclides because the waste is buried to make it relatively inaccessible to the environment.

H.2.8 Multiple Sources

Some disposal facilities may have multiple disposal units (sources) that may impact a receptor. These sources may or may not be located close to one another at the disposal facility. These impacts may or may not overlap in time and space. Licensees may use conservative screening calculations where they assume the impact from different sources all impact the same receptor. If the results are acceptable, the complexity associated with assessing multiple sources would not require further evaluation. If the impacts (plumes) are not expected to overlap because of different hydrologic transport pathways, different wasteform performance, or different geochemical transport characteristics, licensees may examine each source individually to determine if the waste represents a significant quantity of long-lived radionuclides. The most likely case is that there will be some overlap of impacts from different sources to the assumed receptors. In this case, the licensee should account for overlapping impacts and consider the uncertainty in the estimation of the timing and location of potential impacts.

H.2.9 Other

Numerous other factors may play a role in the determination of significant quantities of long-lived radionuclides and should be considered as necessary.

These include but are not limited to the following:

- Site environmental conditions—The geologic materials and the geochemical environment can significantly impact the timing and magnitude of radionuclides that may reach a hypothetical receptor.

- Local and regional practices—The practices of hypothetical receptors, such as land use, location of populations from the facility, recreational activities, consumption rates, and time of exposure will have a direct impact on the determination of significant quantities and concentrations of long-lived radionuclides.

- Disposal practices and methods—Disposal practices can impact the determination of significant quantities and concentrations of long-lived radionuclides. For example, placing DU at the bottom of disposal cells at an arid or semi-arid site will help reduce the long-term impacts from radon.

If these other factors are sufficiently complex or require significant analyses to address, then screening analyses may not be appropriate, and the licensee should instead proceed to technical analyses using a 10,000-year compliance period to determine compliance with 10 CFR 61.41(a) and 10 CFR 61.42(a).

H.3 Generic Screening Values

Generic screening values can be useful to make the regulatory process more efficient for licensees and regulators. As outlined in Section H.2, many factors can impact the determination of significant quantities and concentrations of long-lived radionuclides. Generic screening values are generally based on a set of conservative or pessimistic, but not worst case, assumptions. Table H-1 compares different concentrations that the NRC staff considered for use as generic screening values. The values provided in the middle column are the Class A concentrations given in 10 CFR 61.55, “Waste classification.” These values were derived with consideration of inadvertent intruders (10 CFR 61.42) rather than release to groundwater pathways that are consistent with technical analyses performed for the 10 CFR 61.41 performance objective. The NRC provides generic screening values in Appendix H to NUREG-1757, Volume 2, “Consolidated Decommissioning Guidance: Characterization, Survey, and Determination of Radiological Criteria—Final Report,” Revision 2, issued July 2022 (NRC, 2022), for use in decommissioning. These values are for surface soils (not for buried waste) and are shown for select radionuclides in the second column from the right in Table H-1. The NUREG-1757 values are considerably more restrictive because the radioactivity is assumed to be present in surface soils. For buried waste, the radionuclides first need to be released and transported before they can be contacted or consumed by a receptor. The release and transport processes result in considerable dilution and dispersion in most systems. The right-most column in Table H-1 provides values calculated in this appendix using the approach, assumptions, and values found after the table. The values generally fall between the values derived for surface soils (NUREG-1757, Volume 2, Revision 2) and those derived for inadvertent intruder protection in 10 CFR Part 61.

Table H-1 Concentrations of Long-Lived Radionuclides Considered for Use as Generic Screening Values

Radionuclide	Half-life (years)	10 CFR Part 61 Class A (Ci/m^3)[a]	NUREG-1757 Appendix H (Ci/m^3)[b]	Calculated in This Appendix (Ci/m^3)
C-14	5,700	0.8	$1.2x10^{-5}$	0.48
Ni-59	$1.01x10^{5}$	22*	0.0055	430
Se-79	$2.95x10^{5}$	-	-	1.0
Tc-99	$2.11x10^{5}$	0.3	$1.9x10^{-5}$	0.1
Sn-126	$2.30x10^{5}$	-	-	4.6
I-129	$1.57x10^{7}$	0.008	$5.0x10^{-7}$	0.0036
Pb-210	22.26	-	$9.0x10^{-7}$	0.17
Am-241	432.2	0.01	$2.1x10^{-6}$	0.25
Np-237	$2.14x10^{6}$	0.01	-	0.0002
Pu-239	24,110	0.01	$2.3x10^{-6}$	0.025
Pu-240	6,564	0.01	-	0.025
Pu-242	$3.75x10^{5}$	0.01	-	0.027
U-234	$2.45x10^{5}$	-	$1.3x10^{-5}$	0.032
U-238	$4.47x10^{9}$	-	$1.4x10^{-5}$	0.033
Ra-226	1,600	-	$7.0x10^{-7}$	0.067

* In activated metal.

a Concentrations given in nanocuries (nCi) per gram (g) converted assuming a density of 1,000 kilograms (kg) per cubic meter (m^3).

b Concentrations given in picocuries (pCi)/g converted assuming a density of 1,000 kg/m^3.

The NRC staff provides the generic screening values in Table H-1 for determining whether the quantity of long-lived radionuclides proposed for disposal is significant, and therefore the appropriate analyses timeframes to apply. The values in the table should not be used to accept or decline the disposal of long-lived waste; rather, they can be used to determine whether a 1,000- or 10,000-year compliance period is appropriate.

The values in the table were calculated (as described below and in Attachment H-1) with simple, first-order methods that will generally apply to many sites. If licensees can demonstrate that these simple, first-order methods and the general assumptions discussed below are applicable to the potential disposal site, they may compare the concentration of long-lived radionuclides in their waste to the right-hand column of Table H-1 and calculate a sum of fractions as is done for waste classification (10 CFR 61.55).

If the resulting value is greater than 1, then the waste would be considered to contain significant quantities of long-lived radionuclides. Attachment H-1 gives the example of how the value in the right-hand column of Table H-1 was calculated for plutonium-239, as well as how a licensee may derive different values using a simplified dose screening assessment. If a licensee were to derive site-specific values, it should consider potential impacts to offsite members of the public (10 CFR 61.41), as well as to inadvertent intruders (10 CFR 61.42). Any waste that contains concentrations of long-lived radionuclides greater than the values found in Table 1 of 10 CFR 61.55 (applying the sum of fractions) would be considered to contain significant quantities of long-lived radionuclides, regardless of the estimation of potential groundwater impacts.

The NRC staff calculated the values in the right-hand column of Table H-1, accounting for the major physical and chemical processes that influence the concentrations of radionuclides to which a receptor may be exposed from releases to groundwater. The most important factors are the rate of water contacting the waste (e.g., infiltration), the partitioning of radioactivity between the waste and water contacting the waste (e.g., distribution coefficient or K_d), and the amount of dilution the occurs between the flux of water flowing through the waste and the water flowing in the underlying aquifer.

Release of radioactivity from the waste will be influenced by when the waste is contacted by water (or can migrate in gaseous form) and how much radioactivity is dissolved in the water after contact. This infiltration is usually a site-specific factor that can be influenced by many features and processes. Intact waste packages, robust wasteforms, and engineered barriers such as cover systems may all contribute to reducing the amount of water that contacts the waste. Over long periods of time, the effectiveness of man-made systems will decrease until the amount of water contacting the waste is driven by the natural characteristics of the site. For screening analyses, resistive barriers should not be credited because they will eventually degrade and fail and will not provide measurable performance. Long-term infiltration values of 15-25 percent of annual precipitation are usually appropriate for screening analyses. For a hypothetical site with 25 cm/year (yr) (10 in/yr) precipitation, this is approximately 5 cm/yr (2 in/yr) of infiltration.

The presence of containers and variability in natural materials results in degradation of physical barriers over time but not at the same rate. In arid or semi-arid locations, even carbon steel, which is generally not viewed as being an effective long-term engineered barrier, may provide some performance benefit. These features and processes result in a spreading of releases over time such that not all radioactivity is released from the facility at the same instance. A reasonable value to implement in the screening process is consideration of a 30-year period for releases, or a fractional release rate (FRR) from the facility of 0.033/yr (NOTE: this parameter is shown as FRR_container in Attachment H-1). Most waste facilities will operate for longer than 30 years, such that the waste and packages will be emplaced at different times and have different periods of exposure to the disposal environment that could lead to degradation and release.

When water contacts the waste, not all the radioactivity in the waste partitions into the water. The durability of the wasteforms and the geochemical environment of the disposal system will dictate the partitioning of radioactivity of the waste to the contacting water. Complex geochemical modeling is sometimes used, but more commonly a simpler approach with a distribution coefficient (K_d) is used. The distribution coefficient expresses the ratio of the mass of contaminant per mass of solid to the mass of contaminant per volume of solution. It has units of volume/mass.

Figure H-3 shows the results of simulation of the FRR for a single unit of waste applying different infiltration values and distribution coefficients (NOTE: this parameter is shown as GeochemFactor in Attachment H-1). The response to different infiltration values is relatively linear. Higher values of infiltration result in higher FRRs because the water associated with the void space in the waste package is exchanged more quickly. The response to different distribution coefficients is non-linear and inversely proportional.

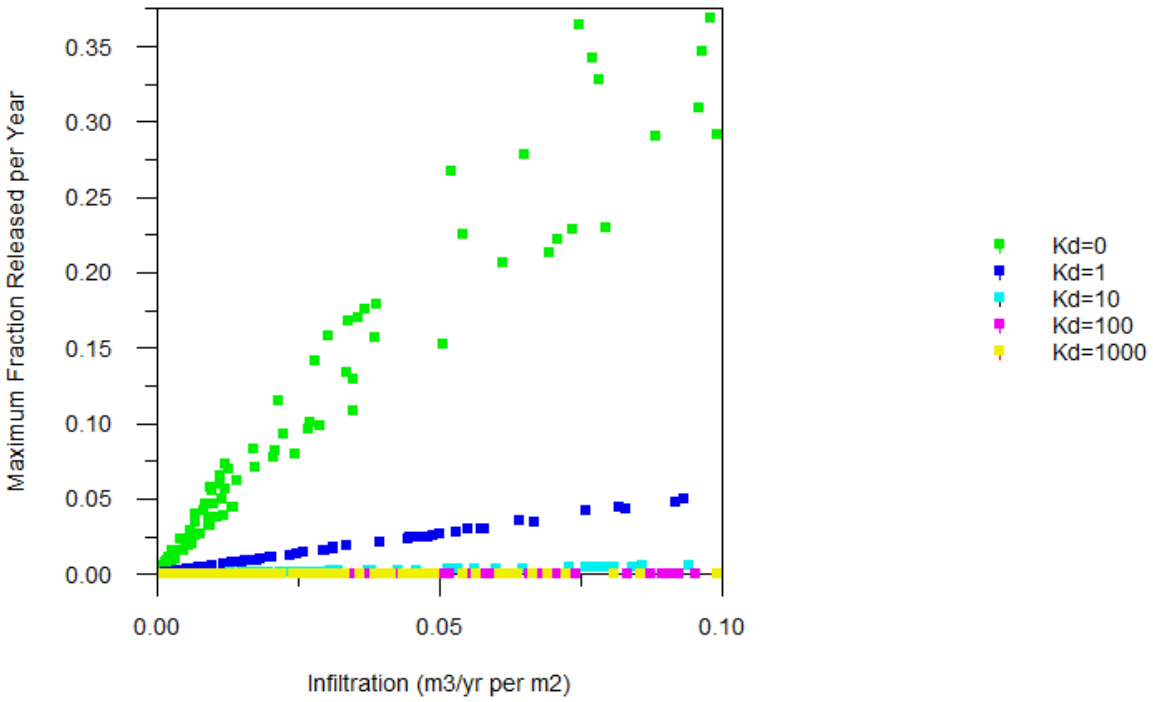


Figure H-3 Simulated Fractional Release Rates as a Function of Infiltration Rate and Distribution Coefficient

At an infiltration value of 5 cm/yr (2 in/yr), the median FRR for distribution coefficients of 0, 1, 10, 100, and 1,000 milliliters (mL)/g were 0.18, 0.027, 0.003, 0.0003, and 0.00003, respectively. Different values for different combinations of infiltration values and distribution coefficients can be obtained from Figure H-3 because values precise to an order of magnitude are sufficient for screening.

The appropriate distribution coefficients to assign for a screening calculation are more difficult to determine because distribution coefficients are dependent on many factors. The approach used for this evaluation was to examine the ranges of values provided in the commonly cited compilation in Sheppard and Thibault (1990).

This reference provides geometric mean values and ranges for a variety of elements in different geologic materials. Because the reference is based on observations and measurements, the values provided also reflect different geochemical conditions. The NRC staff used the geometric mean values for sand for each element considered, reduced to the next unit value in log space. For example, the geometric mean value for uranium in sand was given as 35 mL/g. This value was reduced to 10 mL/g. The values assigned to C, Ni, Se, Tc, Sn, I, Pb, Am, Np, Pu, U, and Ra were 1, 100, 10, 0, 100, 1, 100, 1000, 1, 100, 10, and 100 mL/g, respectively.

The amount of dilution that occurs within the aquifer is a site-specific factor that can be very different from site to site. The approach to calculate the dilution factor is to estimate the amount of infiltration flowing through the plan view area of the disposal facility per year. That volume of infiltration then enters the aquifer where the flux of radionuclides is mixed with uncontaminated water flowing in the aquifer.

The following assumptions were made:

- The volume of long-lived waste disposed is 100,000 m^3, with an average waste thickness of 5 m (16 ft).
- The long-term infiltration is 5 cm/yr (2 in./yr)— appropriate for a semi-arid site.
- The average flow rate of water in the aquifer is 0.1 m/day (100 ft/yr), and the thickness of the plume at the receptor location is 5 m (16.4 ft).
- The area of waste disposal is roughly square, and flow in the aquifer is perpendicular to one dimension.

The resultant dilution factor in the aquifer is approximately 25. Because the results are relatively linear with changes in the dilution factor, a licensee can modify the values found in Table H-1 for its site-specific dilution factor using a simplified dose screening assessment.

After radioactivity is released and transported in the aquifer, it was assumed that the contaminated water would be consumed by an adult receptor at a rate of 2 liters (L)/day. This consumption rate corresponds to a value of 68 ounces, and the receptor spends no time away from the exposure point. Other ingestion pathways could be included, but it is not necessary if pessimistic values for water ingestion are used. As a good rule of thumb for screening analyses, inclusion of other ingestion pathways besides water results in about twice the dose from a generic mixture of radionuclides. Actual values depend on the radionuclides present, what is consumed, and the partitioning of the radionuclides between water, soil, and the consumed products. The simulation used dose conversion factors from Federal Guidance Report No. 12, “External Exposure to Radionuclides in Air, Water and Soil,” issued September 1993 (EPA, 1993). Other dose conversion factors could be used, as the effect is linear with changes in the coefficients. The dose conversion factors of progeny may be substantially different than that of the parent radionuclide. When evaluating decay chains, a licensee should use caution to ensure that the dose conversion factors applied are appropriate based on the estimated concentrations of radionuclides impacting the receptor.

When a licensee can demonstrate that the potential disposal site is consistent with the assumptions, the values in Table H-1 may be used without modification. The most likely case is that a licensee’s disposal site may have some differences. The licensee may use the values in Table H-1 as a starting point to then calculate alternative values consistent with its site-specific information and design. For example, a licensee wishes to dispose of waste containing Tc-99 at 3 Ci/m^3 at a disposal site with 5 millimeters (mm)/yr of infiltration (over all future climate conditions) rather than the 5 cm/yr value used for development of Table H-1. Other site characteristics are generally comparable. The licensee would use Figure H-3 to select an FRR (GeochemFactor) suitable for the lower infiltration value for a radionuclide that is non-sorbing (0 mL/g K_d), that is, approximately 0.02. Therefore, a suitable Tc-99 concentration would be 1 Ci/m^3 x (0.18/0.02) or 9 Ci/m^3. The concentration of Tc-99 proposed for disposal (3 Ci/m^3) would not trigger the need to evaluate the 10,000-year compliance period for significant quantities of long-lived waste. Attachment H-1 describes this example in further detail.

The results in the righthand column of Table H-1 are generally consistent with the Class C values provided in 10 CFR Part 61. A notable exception is the value calculated for Np-237, which is lower because of the generally higher rate of transport compared to most transuranic

radionuclides. Values for additional radionuclides that may be important in long-term performance assessments are provided. Given a description of the methodology, a licensee could calculate values for other radionuclides not provided in the table. The values calculated for uranium indicate that disposal of concentrated DU should be subject to evaluation with a longer compliance period unless the quantity of material disposed is limited.

The values provided in the table do not include the impact of daughters that may be produced during radioactive decay. For some radionuclides, the impact of a daughter radionuclide can be of greater significance than that of the parent. For example, radon gas is produced in the decay of uranium. At arid sites, radon emission to the air pathway is the most significant contributor to dose rather than ingestion of water containing uranium. Though Am-241 has a relatively high screening concentration relative to other transuranics, it decays into Np-237, which has a relatively low screening concentration. Because of different rates of transport in different geochemical and geologic environments, daughter radionuclides are best considered using simplified approaches such as identifying the potential peak concentrations of daughters based on the decay rates of parents and making screening decisions using those peak concentrations.

H.4 Simplified Dose Assessment Screening

After considering the factors in Section H.2, and if the generic screening values provided in Section H.3 are lower than the proposed waste concentrations or not appropriate given the site-specific conditions, the NRC staff recommends licensees complete a site-specific screening or simplified dose assessment to determine what quantity and concentration of long-lived radionuclides would be considered a significant quantity under their specific disposal conditions and would, therefore, require a 10,000-year compliance period for the technical analyses.

H.4.1 Factors to Consider for a Simplified Dose Assessment

Whereas certain assumptions were necessary for the generic screening performed in Section H.3, those assumptions may not be applicable at a specific site. When developing the screening values, a licensee should determine that the assumptions are consistent with the licensee's proposal. Likewise, a regulator should verify that a licensee's proposal is consistent with use of the screening values. Site-specific screening or a simplified dose assessment may be necessary.

The first consideration is the primary pathway for release and exposure to a member of the public. For most sites, because the waste is buried in an engineered system, the most likely release pathway is in groundwater. However, at some very arid sites, the dominant release pathway may be in air, through biota, or from releases due to surficial geological processes. At some very arid sites, the releases to the groundwater pathway may be negligible or there may be no viable uses of groundwater now and in the future. A groundwater-based screening process would not be appropriate for these sites.

For sites where releases to groundwater are the primary exposure pathway, the values of parameters used to estimate the reduction in concentrations of radioactivity from the waste to the water in the aquifer may be substantially different than those used for generic screening.

Factors that a licensee could consider include, but are not limited to, the following:

- The effective dilution factor of fluxes of radioactivity from the vadose zone to the aquifer can be significantly smaller or larger depending on the infiltration and groundwater flow velocities.

- Infiltration and groundwater flow velocities are sensitive to precipitation (climate). For screening purposes, the licensee may want to identify pessimistic values that account for future long-term climate states.

- The wasteforms and waste packages or overpacks may be more robust such that an FRR (FRR_container) smaller than 0.033/yr is appropriate. Licensees should be sure not to take double credit with geochemical aspects of release.

- The long-term infiltration and geochemical environments for release may be different than those assumed in the generic screening analyses. Figure H-3 or the equivalent may be used to select different values.

- Different assumptions about the conversion of groundwater concentrations of radioactivity to dose may be appropriate.

A licensee's technical basis should be sufficient to justify the results of the screening process to its regulator. If there are significant uncertainties that cannot be addressed or would take substantial effort to address, then the licensee should use the longer compliance period in the technical analyses supporting its safety case.

As mentioned previously, the primary difference in the revised 10 CFR Part 61 requirements is the addition of the inadvertent intruder assessment as a requirement to ensure protection of the intruder for 10 CFR 61.42 for radionuclides that are not contained in the waste classification tables (Tables 1 and 2 of 10 CFR 61.55) or for which the concentrations exceed the values in the waste classification tables (GTCC waste) but that may be disposed deeper and with more robust intruder barriers. Assessment of the potential impacts to a member of the public under 10 CFR 61.41 is required. The compliance period must be aligned with the characteristics of the waste being disposed. Although uranium isotopes are not included in Tables 1 and 2 of 10 CFR 61.55, they should be included in a licensee's performance assessment of its facility if those isotopes are present in the disposed waste. Consideration of groundwater (or other pathway) impacts is encouraged for determining the compliance period. At a humid site, the impacts to an offsite member of the public through groundwater may be significantly more limiting than the impacts to an inadvertent intruder.

H.4.2 Simplified Dose Assessment Calculations

This section discusses simplified dose assessment calculations for determining whether a proposed disposal of DU and other long-lived radionuclides represents a significant quantity. Section H.5 discusses a site-specific screening process. The main distinction between the two analyses is that the simplified dose assessment may rely on mostly generic sources of information and use more conservatism, whereas the site-specific screening will rely on actual site data and may take credit for features, events, and processes that are generally not credited when performing a simplified dose assessment.

A key driver of variability in the groundwater pathway doses associated with disposal of long-lived radionuclides is typically the variability in hydrogeological characteristics. Most disposal sites have a viable water pathway. The disposal of long-lived radionuclides can result in estimated drinking water doses that exceed 0.25 millisievert (mSv)/yr (25 millirem (mrem)/yr), sometimes by a significant margin. Depending on the hydrogeology, these water pathway impacts may occur within the first 1,000 years or significantly later. The NRC staff previously determined that the long-lived radionuclides I-129, Tc-99, Cl-36, and C-14 are the most problematic because of their relatively high mobility in the environment (NRC, 1982b, page 5-43). Uranium can also be problematic if the quantity of uranium-bearing waste is large. The solubility of uranium in oxide forms is moderately low. Depending on the form of uranium, the specific surface area may be large. If the surface area is large, such as if the material is in powder form, and if there is a moderate or high natural recharge rate, the flux of uranium to the aquifer underlying the site may be significant. Various site-specific conditions can influence the impacts from uranium disposal at a site. Licensees should not assume that disposal of uranium-containing waste is inherently compliant with the performance objectives.

With respect to the demonstration of compliance with 10 CFR 61.41, the quantities of long-lived radionuclides that may be acceptable for disposal may vary considerably at different sites. The following will impact the magnitude of estimated impacts:

- the dilution or dispersion of radioactive releases to an aquifer at a site with a potable groundwater pathway
- ingrowth of progeny that increases the radiotoxicity of the waste
- wasteform(s)
- geochemistry of the site
- erosion rates and engineered barrier performance
- the rate of release of gaseous radionuclides (e.g., from the ingrowth of radon)

Site-specific concentration and quantity values for long-lived radionuclides that are acceptable to evaluate using a 1,000-year compliance period can be generated. To determine whether a quantity of long-lived radionuclides is significant, a licensee may use screening analyses. The screening analyses will likely be iterative because the initial results may identify key features to consider in greater depth or identify features, events, and processes that are of lesser significance. Regardless of whether a site does or does not contain significant quantities of long-lived radionuclides, every licensee will have to provide a performance assessment that evaluates an analysis timeframe of at least 1,000 years based on the definition of compliance period in 10 CFR 61.2, unless the licensee meets the criteria in 10 CFR 61.1(b).

Dose-based screening should be reasonably conservative. A reasonably conservative dose screening analysis is not a worst-case evaluation; instead, it is an evaluation that assigns credit to the more robust long-term phenomena. If the dose screening analyses become very complex to determine whether the site should be evaluated with a 1,000-year or 10,000-year compliance

period, the licensee should use the 10,000-year compliance period followed by the performance period. The dose screening analyses should do the following:[2]

- Credit dilution in an aquifer.
- Credit wellbore dilution.
- Use infiltration values comparable to natural recharge rates.
- Account for radioactive decay.
- Credit wasteform performance associated with mass transfer limitations.
- Consider ingrowth of progeny. If radon is a consideration, ensure that the analyses include the buildup of radium from decay of uranium.
- Do not credit differences in expected transport times between radionuclides.
- Eliminate performance credit for those engineered features that are not anticipated to provide performance credit over the complete analysis timeframe (e.g., reduction in infiltration from a surface cover).
- Evaluate erosion if the design does not include engineered barriers that provide erosion protection over the complete analysis timeframe.
- Evaluate impacts to offsite members of the public and inadvertent intruders.

Licensees should document a basis when including (or excluding) features, events, and processes that improve (or decrease) performance.

Development of simplified dose assessment screening calculations should not be a significant burden. The doses may be amenable to calculation in a spreadsheet with analytical expressions. They may be developed from existing analyses that a licensee already has in place if the licensee has an operating facility that it wishes to expand to include new waste streams that contain long-lived radionuclides in different concentrations than previously accepted. However, if the licensee does not have any previous analyses developed, then a new screening analysis would need to be developed. The result of the dose screening analyses should be a projected dose that the licensee can compare to the performance objectives. The licensee can, based on using ratios of its dose results to the dose limits, define what constitutes a significant quantity of long-lived radionuclides for its disposal site. The more limiting value for the two types of receptors (i.e., member of the public and inadvertent intruder) should be used.

H.5 Site-Specific Screening

Licensees may propose site-specific screening analyses to determine what quantity of long-lived radionuclides would be acceptable for disposal using a 1,000-year compliance period. Some disposal sites may have unique pathways of potential release (e.g., biota) and atypical disruptive processes (e.g., moderately high rates of aeolian erosion). The unique characteristics of the disposal facility may make the use of the generic screening values

[2] Each item should only be included if it applies to the site-specific application under consideration.

discussed in Section H.3 unsuitable. The site-specific screening should be clearly conservative and relatively simple, though it is likely to be more complex than the simplified dose assessment discussed in Section H.4. The screening calculations should consider relevant disruptive processes, release pathways, and activities and behaviors of receptors. If the screening process is very complex, it may be more efficient for the licensee to instead assume the disposal amount they are considering constitutes a significant quantity and apply the 10,000-year compliance period. Sites that may experience instability that results in releases to the environment (e.g., erosion) should apply a 10,000-year compliance period because erosion processes are strongly influenced by timeframes due to the coupling with climatic conditions.

H.6 Examples for Depleted Uranium Disposal

Licensees who meet the criteria in 10 CFR § 61.1(b)(1) may elect to continue to comply with 10 CFR 61.13(f), 10 CFR 61.41(c), 10 CFR 61.42(c), and 61.50(c); these sections duplicate the original 10 CFR Part 61 regulations before the ILLRWD rule. These licensees may also use analysis timeframes deemed acceptable by the appropriate regulator according to the original 10 CFR Part 61 regulations. LLW disposal facilities that begin operations after the revision of 10 CFR Part 61 in the ILLRWD rule must meet the current requirements regardless of the quantities of long-lived radionuclides planned for disposal.

The compliance period to use at a site that meets the 10 CFR 61.1(b) criteria should be determined based on the amounts of DU proposed for future disposal. The determination does not need to include waste previously disposed at the site. For example, if the site was licensed before 10 CFR Part 61 was revised and the licensee proposes not to accept more than 10 MT of DU in the future, the site may use a 1,000-year compliance period in the assessment. This licensee would use the total site inventory (including DU disposed) when demonstrating the original 10 CFR Part 61 requirements can be met, along with the disposal requirements used when the initial licensing decision for disposal of the material was made. Section 9.3 of this document discusses potential mitigation measures. If this licensee determines in the future that it wants to accept additional quantities of DU, it should contact its regulator to determine whether different regulatory requirements apply. If the licensee proposes to accept more than 10 MT of DU in the future, the licensee must evaluate, and the regulator must approve, which compliance period will be used at the site.

The NRC staff conducted a technical analysis to understand the impacts of near-surface disposal of large quantities of DU, such as those expected to be generated at uranium enrichment facilities in SECY-08-0147, “Response to Commission Order CLI-05-20 Regarding Depleted Uranium,” dated October 7, 2008 (NRC, 2008). The staff developed a screening model to evaluate the radiological dose and uncertainties associated with near-surface disposal of large quantities of DU at a generic LLW disposal site. The generic disposal site had a broad range of climatic (e.g., humid or arid), hydrological, and geochemical conditions. The model was used to understand the impacts of key variables such as: disposal configurations, performance periods, institutional control periods, wasteforms, site conditions, exposure pathways, and receptor scenarios. Calculations were performed probabilistically to represent the impact of variability and uncertainty on the results. The analysis methodology used in the model was consistent with the technical analysis methodology found in the final environmental impact statement supporting 10 CFR Part 61 (NRC, 1982b). The model framework is based on several key assumptions, which are detailed in Enclosure 1 of SECY-08-0147.

The staff's technical analysis concluded that near-surface disposal (i.e., generally at a depth less than 30 m, as defined in 10 CFR Part 61) may be appropriate for large quantities of DU under certain conditions. However, unfavorable site conditions, such as shallow disposal (i.e., at a depth of less than 3 m) or disposal at humid sites where contaminated groundwater is used such as for consumption, could exceed the performance objectives of 10 CFR Part 61, Subpart C, "Performance Objectives." The staff concluded in its analysis that shallow disposal for large quantities of DU is not likely to be appropriate regardless of site conditions; however, small quantities (approximately 1 – 10 MT) of DU could be disposed of at shallow depths. In the analysis, a significant quantity was determined to be greater than 10 MT. The staff recommends that licensees perform similar analyses, tailored to their site-specific input parameters, and discuss the results with regulators to determine what quantities of DU could be disposed. This analysis would be performed for the additional quantities of DU to be accepted and does not need to include the existing radionuclide inventory at the site.

The screening process to determine a compliance period suitable for DU disposal may be more complex than for other long-lived waste because significant releases can occur to either the air or water pathways, and the ingrowth of daughter radionuclides results in material that is radiologically more hazardous than the parent (uranium) radionuclides over long periods of time. Figure H-4 provides a flow chart acceptable to use for determining the compliance period for DU disposal. As shown in Figure H-4, potential impacts from releases to air must be considered, which is not usually a significant release pathway from most LLW.

A licensee may want to perform a screening analysis for DU disposal based on inventory rather than examining impacts to different dose pathways. The first step in performing a screening analysis based on inventory is to use the concentration limit for uranium developed in the original Draft Environmental Impact Statement (DEIS) for 10 CFR Part 61 (NRC, 1981a). Use of this concentration limit will ensure consistency with the values prescribed for other radionuclides in 10 CFR 61.55. The DEIS for 10 CFR Part 61 contained a concentration limit for uranium of 0.05 microcuries (μCi)/cm^3. Depending on the density of the waste and the assumed mass fractions of U-238, U-234, and U-235, this concentration is comparable to 0.37 kilobecquerel (KBq)/g (10 nCi/g).

For a disposal facility containing 28,000 m^3 (1,000,000 ft^3) of waste, the mass of DU this concentration corresponds to is approximately 3,000 MT of pure DU (in oxide form) distributed over the volume (~1 percent concentration). The volume of waste should be based only on the waste streams associated with DU and not the total volume of waste disposed in the facility. The total mass could vary depending on the mass fractions of the isotopes assumed and the density of the waste. If the waste is not pure DU or the waste has been disposed with backfill, for example, the mass of the waste stream (3,000 MT) may be scaled appropriately, according to its characteristics. For cases with limited quantities or large volumes but low concentrations, inventory-based screening is likely to be sufficient.

The NRC staff stated in the 1981 DEIS that an assessment should include the impacts of radon on air and the impacts of uranium and other elements on groundwater, and that this inclusion was best accomplished with a site-specific analysis. At arid sites, the impacts of radon are generally more significant, whereas at humid sites, the impacts of uranium on groundwater are more significant. There can be considerable variability in observed radon fluxes but over a large number of measurements at different sites, the significance of moisture is clear.

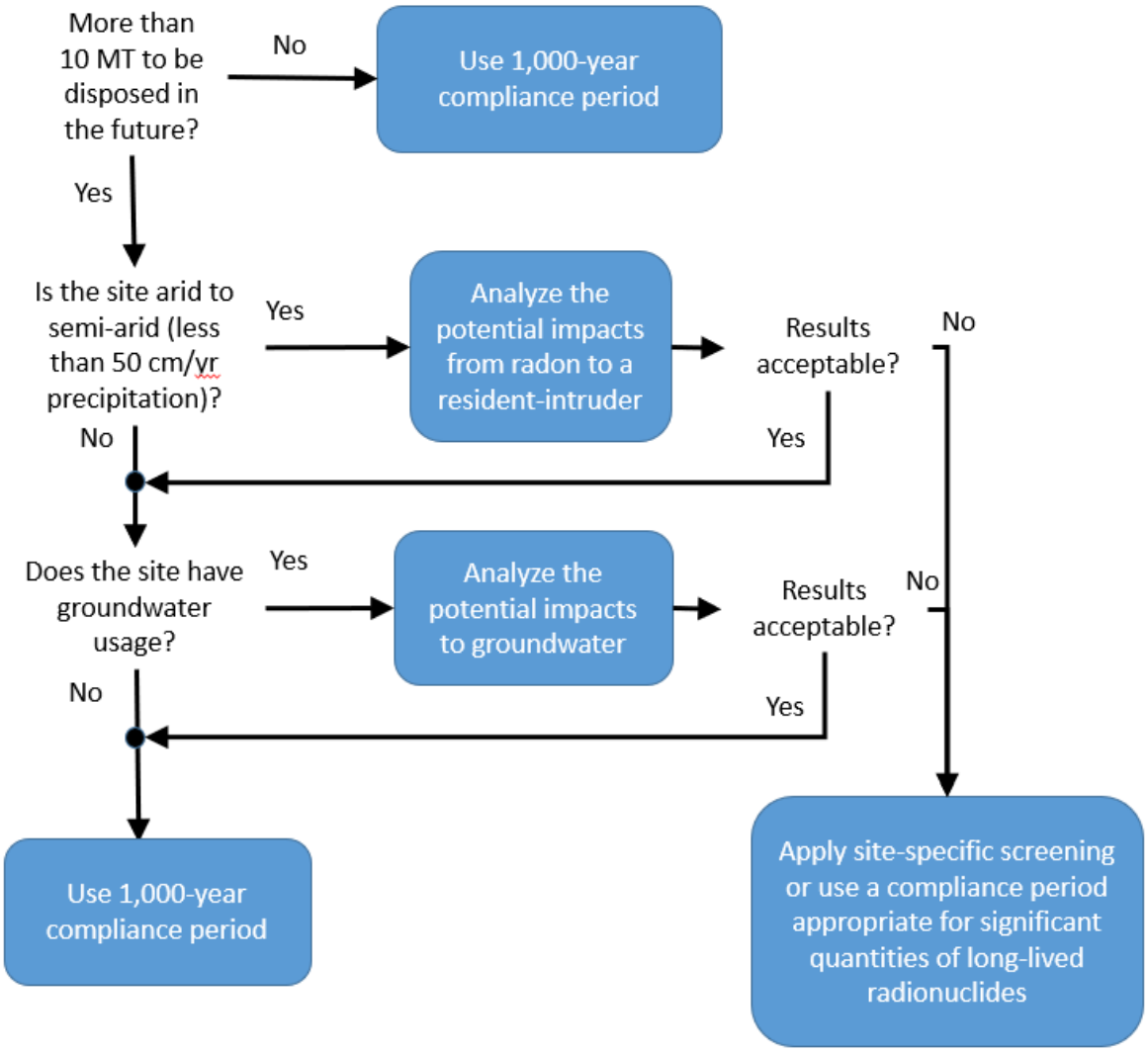


Figure H-4 Recommended Process to Determine Compliance Period for Depleted Uranium Disposal

Depending on the site conditions, the analysis should include a screening assessment of the relevant pathways to determine whether the DU disposal would represent a significant quantity and therefore should be subject to a longer compliance period. If a licensee cannot demonstrate that the impacts from radon or to groundwater can be properly assessed with screening calculations, then a more detailed site-specific analysis, as discussed in Section H.5, is appropriate.

The recommended precipitation value of 50 cm/yr (20 in/yr) in the Figure H-4 screening process should encompass most sites, but a licensee can always estimate potential radon impacts at wetter sites too. Given 5 m (16 ft) of wet to moderately wet soils, effective diffusivity of radon is low, resulting in a substantial reduction in radon fluxes at the surface. Lower thicknesses of cover materials, discrete pathways, and high evapotranspiration would all be characteristics that may cause a licensee to evaluate potential impacts from radon at a disposal facility where precipitation is greater than 50 cm/yr (20 in/yr). Though steps have been provided to ensure that the main technical considerations associated with assessing the risk from uranium have

been accounted for in the inventory-based screening process, the licensee and regulator should ensure that the use of the uranium concentration limit from the NRC DEIS is appropriate for the site. In most cases, many of the assumptions are reasonable; however, there are notable exceptions. For example, the impacts associated with contaminated soil following an intruder disturbance (i.e., the mass loading of soil to air with resultant deposition) are based on humid site conditions.

For sites located in arid or semi-arid environments, the NRC staff expects that the impacts associated with intruder disturbance of soil could be higher; therefore, the uranium concentration limit would be lower. The NRC staff provided the assumptions of the calculations in the following documents, among others:

- NUREG-0782, "Draft Environmental Impact Statement on 10 CFR Part 61 'Licensing Requirements for Land Disposal of Radioactive Waste,'" issued September 1981 (NRC, 1981a)

- NUREG-0945, "Final Environmental Impact Statement on 10 CFR Part 61 'Licensing Requirements for Land Disposal of Radioactive Waste,'" issued November 1982 (NRC, 1982b)

- NUREG/CR-4370, "Update of Part 61 Impacts Analysis Methodology," issued January 1986 (NRC, 1986a)

- NUREG/CR-1759, "Data Base for Radioactive Waste Management, Review of Low-Level Radioactive Waste Disposal History," issued November 1981 (NRC, 1981b)

Appendix F to this document summarizes the assumptions and calculations in NUREG-0782 that were used to develop the waste classification concentrations.

Examples H.1 and H.2 provide scenarios that a licensee may encounter when determining whether the future disposal of DU would represent a significant quantity of long-lived radionuclides that would be subject in the technical analyses to a 10,000-year compliance period followed by a performance period.

Example H.1

A previously licensed disposal site is in an arid (18 cm/yr (7 in/yr) precipitation), oxidizing environment with a shallow water table and non-potable groundwater. The licensee proposes to dispose of 2,000 MT of pure DU in an oxide form in a disposal unit volume of approximately 20,000 m^3. The bottom of the DU is 7 m from the current land surface, and the top of the waste is 4 m from the current land surface. The licensee wishes to determine whether they should apply a 1,000-year or 10,000-year compliance period. The licensee had previously completed an intruder dose assessment, but it did not include the impacts from uranium in that assessment.

Conclusion: Although the licensee had previously completed an intruder assessment, it did not evaluate the impacts from uranium. The site is arid; therefore, the licensee must assess the potential impacts from radon. The licensee estimates the maximum radium that will be produced from ingrowth from uranium decay. The licensee uses Regulatory Guide 3.64, "Calculation of Radon Flux Attenuation by Earthen Uranium Mill Tailings Covers," issued June 1989 (NRC, 1989a), to estimate the flux of radon into an intruder's basement. The licensee demonstrates that Regulatory Guide 3.64 is appropriate to use at its site. The flux, when converted to dose, results in a potential dose to an inadvertent intruder, if they were to construct a residence in the area where the DU was disposed, of 423 mrem/yr. This result, using conservative generic parameters in the assessment, was below the performance objective of 500 mrem/yr. The groundwater at the site is not potable now or in the foreseeable future. Therefore, the licensee and regulator conclude that the licensee may apply a 1,000-year compliance period.

Example H.2

A disposal site has a humid, oxidizing environment with a water table 30 m below the disposed waste. The groundwater is potable now and in the foreseeable future. The licensee proposes to dispose of 7,000 MT of DU in a disposal unit volume of approximately 7,000 m^3. The bottom of the DU is 7 m from the current land surface, and the top of the waste is 5 m (16 ft) from the current land surface. The licensee has not completed an intruder assessment. The licensee wishes to continue to apply a 1,000-year compliance period that was used when the disposal facility was initially licensed.

The licensee uses Figure H-3 and determines that it needs to evaluate the potential impacts to groundwater. The DU is in oxide form and is solidified in cement. Therefore, the licensee uses a value of uranium solubility from the literature for cement-modified water. The licensee ignores any dispersion associated with transport and calculates uranium concentrations in the aquifer accounting for dilution. The uranium source term does not comprise a large area, and the aquifer has properties such that a well to support a yield necessary to supply water to a residence produces a large capture radius and significant wellbore dilution. The licensee estimates that the water withdrawn from the well would result in a drinking water dose of 35.3 mrem/yr to a member of the public located 100 m from the edge of the buffer zone of the facility.

Conclusion: Because the dose from the screening assessment is above the dose limit in the 10 CFR 61.41 performance objective (25 mrem/yr), the licensee applies a 10,000-year compliance period for the licensing analysis to demonstrate compliance with 10 CFR 61.41.

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ATTACHMENT H-1
CALCULATION OF GENERIC SCREENING VALUE

This attachment steps through the derivation of the value of plutonium (Pu)-239 found in Table H-1 and then provides an example of how a licensee may modify inputs to account for differences in its disposal site compared to the assumptions used to develop the screening values. Table H-1-1 provides the parameters used in the calculation.

Table H-1-1 Parameters Used in the Screening Value Calculations

Parameter ID	Description	Units	Value
DL	Dose limit from ingestion	mrem/yr	25
DW	Drinking water ingestion rate	L/yr	730
DCF_Pu239	Dose conversion factor for ingestion of Pu-239	mrem/pCi	3.5372×10^{-3}
WasteVolume	The volume of waste to be disposed	m^3	100,000
WasteThick	The thickness the disposed waste will occupy	m	5
Infiltration	The amount of infiltration per year that flows through the disposal unit	cm/yr	5
AquiferFlow	The groundwater velocity in the aquifer	m/day	0.1
PlumeThick	The thickness of the plume at the receptor location	m	5
FRR_container	The fractional release rate describing the spreading of releases occurring as the result of distributed failures in time	1/yr	0.033
Kd	The distribution coefficient for Pu-239 describing waste release	ml/g	100
GeochemFactor	The parameter describing the fractional release of radionuclides from the waste to the water once failure has occurred	1/yr	Lookup, use Figure H-3

Starting with the dose limit (DL), the rate of drinking water ingestion (DW) and the most limiting dose conversion factor for effective dose from Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," issued September 1988 (EPA, 1988), for Pu-239 (DCF_Pu239) the limiting concentration for ingestion (LCI) can be calculated by (unit conversion factors not shown throughout calculations):

$$LCI = \frac{\frac{DL}{DCF_{Pu239}}}{DW} = 9.6818\ x\ 10^{-9}\ Ci/m^3$$

This is the concentration that if ingested will produce the dose limit at the given ingestion rate. The value would be a vector by the number of radionuclides considered.

The dilution factor of flow through the facility compared to the amount of flow in the aquifer is needed to calculate the concentration of radionuclides in water near the waste. The flow through the facility (*Quz*) is given by the following:

$$Quz = \frac{WasteVolume}{WasteThick}\ Infiltration = 1{,}000\ \frac{m^3}{yr}$$

The characteristic length of the facility perpendicular to the flow (*CharLength*) is given by the square root of the facility plan view area:

$$CharLength = \sqrt{\frac{WasteVolume}{WasteThick}} = 141.4\ m$$

The flow of water through the aquifer (*Qsz*) is given by the following:

$$Qsz = AquiferFlow\ CharLength\ PlumeThick = 25{,}806\ \frac{m^3}{yr}$$

The aquifer dilution factor (ADF) can be calculated by the following:

$$ADF = \frac{Qsz}{Quz} = 25.8$$

The concentration of radioactivity in water after release from the waste (Cwaw) is calculated as follows:

$$Cwaw = LCI\ ADF = 2.50\ x\ 10^{-7}\ \frac{Ci}{m^3}$$

The concentration of radioactivity that can be inside the waste container (Ciwc) prior to release is calculated as follows:

$$Ciwc = \frac{Cwaw}{FRR_container} = 7.58\ x\ 10^{-6}\ \frac{Ci}{m^3}$$

The concentration of radioactivity in the waste is a function of the fractional rate of the radioactivity that is released from the waste into the water. Two conservatisms were introduced. First, the effective distribution coefficients assigned were the values from the commonly cited Sheppard and Thibault reference (1990) for sand. The disposal of potentially significant quantities of long-lived radioactivity will likely use some form of engineered wasteforms to limit the dispersion and retention of the radioactivity. Second, based on the geometric mean K_d value for a particular element, the value was rounded down to the next smaller unit value on a log scale. For instance, the value for plutonium in sand is 550 mL/g. That value was then rounded down to 100 mL/g. Then the lookup values described in the body of the report were used, in this instance 0.0003 L/yr.

The concentration of radioactivity in the waste (Cw) is calculated as follows:

$$Cw = \frac{Cwic}{GeochemFactor} = 0.0250\ \frac{Ci}{m^3}$$

By examining these calculations, licensees and regulators should be able to understand and replicate the values provided in Table H-1. If the only changes necessary are to account for parameter differences, modification of the concentrations should be straightforward. For example, if the flow rate in the aquifer was 10 m/day rather than 0.1 m/day, the dilution factor would increase by a factor of 100, and the corresponding concentration (Cw) that would represent a significant quantity would increase by a factor of 100.

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11. ABSTRACT (200 words or less)

This document provides guidance on conducting technical analyses to demonstrate compliance with the performance objectives in 10 CFR Part 61. This document provides implementing guidance for the amendments to 10 CFR Part 61 that are detailed in the U.S. Nuclear Regulatory Commission's "Integrated Low-Level Radioactive Waste Disposal" rule. Though this guidance emphasizes technical analyses, it also provides guidance on many other topics relevant to technical analyses.

The information in this document is intended to supplement existing guidance on low-level radioactive waste on issues pertinent to conducting technical analyses to demonstrate compliance with the performance objectives. This document provides detailed guidance in new areas, such as the inadvertent intruder assessment and analyses for the compliance and performance periods. This guidance discusses approaches to selecting the compliance period (1,000 or 10,000 years after disposal site closure, depending on the quantities of long-lived radionuclides disposed), and covers the performance period analyses (beyond 10,000 years) that should be performed for analysis of long-lived waste.

This guidance should facilitate implementation of the rule by existing licensees and potential applicants and assist regulators in reviewing the technical analyses. This guidance applies to waste streams disposed at 10 CFR Part 61 low-level waste disposal facilities, including significant quantities of depleted uranium, blended waste, and Greater-Than-Class C (GTCC) waste.

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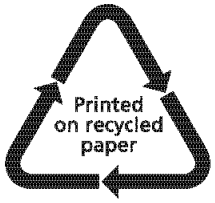
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