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Cementitious Barriers Partnership

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OVERVIEW OF SENSITIVITY AND UNCERTAINTY ANALYSIS

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LIST OF ACRONYMS AND ABBREVIATIONS

AEA	Atomic Energy Act
ALARA	As Low As Reasonably Achievable (ALARA)
ARARs	Applicable or Relevant and Appropriate Requirements
ASAM	Coordinated Research Project on Application of Safety Assessment Methodologies for Near-Surface Waste Disposal Facilities
BWR	Boiling Water Reactor
C	Carbon
CA	Composite Analysis
CATEX	CATegorical EXclusion
CBP	Cementitious Barriers Partnership
CDI	Canyon Disposition Initiative
CEQ	Council on Environmental Quality
CERCLA	Comprehensive Environmental Response, Compensation, and Liability Act
CFR	Code of Federal Regulations
CIG	Components in Grout
CNWRA	Center for Nuclear Waste Regulatory Analyses
COC	Contaminant of Concern
COPC	Contaminant of Potential Concern
CRESP	Consortium for Risk Evaluation with Stakeholder Participation
CSM/CMI	Corrective Measures Study/ Corrective Measures Implementation
CWI	CH2M-WG Idaho, LLC
DCGL	Derived Concentration Guideline Levels
D&D	Decontamination and Decommissioning
DOE	United States Department of Energy
DP	Decommission Plan
DST	Double-Shell Tank
EA	Environmental Assessment
EIS	Environmental Impact Statement
ELLWF	E-Area Low-level Waste Facility
EPA	Environmental Protection Agency
ECN	Energy Research Center of the Netherlands (ECN)
ETR	Engineering Test Reactor
FFA	Federal Facility Agreement
FONSI	Finding of No Significant Impact
FTF	F-Tank Farm
GAO	Government Accounting Office
HLW	High-level waste
HSRAM	Hanford Site Risk Assessment Methodology
HSWA	Hazardous and Solid Waste Amendment
HWMA	Hazardous Waste Management Act
I	Iodine
IAEA	International Atomic Energy Agency

LIST OF ACRONYMS AND ABBREVIATIONS (contd)

ICDF	INEEL CERCLA Disposal Facility
IDEQ	Idaho Division of Environmental Quality
IDF	Integrated Disposal Facility
ILV	Intermediate Level Vault
INEEL	Idaho National Engineering and Environmental Laboratory
INEL	Idaho National Engineering Laboratory
INL	Idaho National Laboratory
INTEC	Idaho Nuclear Technology and Engineering Center
ISAM	Coordinated Research Project on Improvement of Safety Assessment Methodologies for Near Surface Waste Disposal Facilities
K_d	Distribution Coefficient
LANL	Los Alamos National Laboratory
LAWV	Low-Activity Waste Vault
LLW	Low-Level Waste
LFRG	Low-Level Waste Disposal Facility Federal Review Group (LFRG)
LOCA	Loss-of-Coolant Accidents
LTP	License Termination Plan
LTR	License Termination Rule
MCM	Mixing Cell Model
MCL	Maximum Contaminant Level
MEPAS	Multimedia Environmental Pollutant Assessment System
mrem	Millirem
mSv	Milli Sievert
MREM	Millirem (milli Roentgen Equvilent in Man)
MWMF	Mixed Waste Management Facility
NAS	National Academy of Science
NCP	National Contingency Plan
NCRP	National Council on Radiation Protection and Measurements
NDAA	Ronald W. Reagan National Defense Authorization Act for Fiscal Year 2005
NEPA	National Environmental Policy Act
NIST	National Institute of Standards and Technology
Np	Neptunium
NPL	National Priorities List
NRC	United States Nuclear Regulatory Commission
NRCDA	Naval Reactor Component Disposal Area
NSARS	Safety Assessment of Near Surface Radioactive Waste Disposal Facilities
NTS	Nevada Test Site
NUREG	Nuclear Regulatory Commission Regulation
NWPA	Nuclear Waste Policy Act
OMB	United States Office of Management and Budget
ORNL	Oak Ridge National Laboratory
PA	Performance Assessment
PET	Potential Evapotranspiration
PRA	Probabilistic Risk Analysis

LIST OF ACRONYMS AND ABBREVIATIONS (contd)

PRG	Preliminary Remediation Goal
PSDAR	Post-Shutdown Decommissioning Activities Report
Pu	Plutonium
PUREX	Plutonium Recovery and Extraction
PWR	Pressurized Water Reactor
RCRA	Resource Conservation and Recovery Act
RDs/RAs	Remedial Designs/Remedial Actions
RESRAD®	RESidual RADioactivity
RFA/RFI	RCRA Facility Assessment/RCRA Facility Investigation
RIs/FSs	Remedial Investigation/Feasibility Study
ROD	Record of Decision
RVAI	Reactor Vessel Assembly and Internals
RWMC	Radioactive Waste Management Complex
SARA	Superfund Amendments and Reauthorization Act
SCDHEC	South Carolina Department of Health and Environmental Control
SDA	Subsurface Disposal Area
SER	Safety Evaluation Report
Sr	Strontium
SRNL	Savannah River National Laboratory
Sv	Sievert
SWMU	Solid Waste Management Unit
SWSA	Solid Waste Storage Area
TBD	To Be Determined
Tc	Technitium
TEDE	Total Expected Dose Equivalent
TFF	Tank Farm Facility
TBP	Tributyl Phosphate
TRU	Transuranic
TSDF	Treatment, Storage, or Disposal Facility
TWRS	Tank Waste Remediation System
USOMB	United States of Management and Budget
UST	Underground Storage Tank
WCF	Waste Calcining Facility
WSRC	Washington Savannah River Company
WILD	Waste Inventory and Location Database

Overview of Sensitivity and Uncertainty Analysis

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1.0 INTRODUCTION

Performance assessments (PA) and PA-like analyses are conducted to provide a projection of the potential post-closure effects associated with a waste management activity. The results of such an assessment are used as part of the basis for decision-making regarding a specific waste management action. The importance of sensitivity and uncertainty analysis for these projections has been recognized for as long as PAs and PA-like analyses have been conducted. However, there has not been general agreement regarding the specific approaches used to implement such sensitivity and uncertainty analyses. Views on sensitivity and uncertainty analysis can be different depending on the regulatory environment, technical difficulty of a specific problem, and analyst preference as well as other reasons. For example, assessments for waste forms from waste processing may have different goals than soil and groundwater assessments for remediation, which may also be somewhat different than decommissioning assessments. However, there are also similarities in the different approaches that can and should be shared from the perspective of consistency and continuous improvement. Approaches for uncertainty analysis are also an important consideration for assessments of cementitious barriers in a PA approach.

Sensitivity and uncertainty analysis are used in a PA or PA-like analysis as a means to better understand important aspects of system behavior and to quantify the effects of uncertainty on the results of the assessment in order to better inform decisions. Throughout the iterative PA process, sensitivity analyses are used to identify parameters with the greatest influence on the decision to be made and provide a means to focus attention on those parameters for both the operator and the regulator. In this manner, effective use of sensitivity analysis has proven to be an important contributor to cost-effective and defensible assessments.

A common source of debate regarding sensitivity and uncertainty analyses is the choice of deterministic and/or probabilistic approaches. For many years, in the LLW disposal community, it was common to use deterministic approaches, which involved a base case and multiple sensitivity cases targeted at explaining or better illustrating the effects of changes in different parameters on the overall results of the assessment. Over time, there has been increased use of probabilistic approaches to replace or supplement the deterministic calculations. At a recent workshop sponsored by United States Department Of Energy Office of Environmental Management (USDOE-EM) and the Low-Level Waste Disposal Facility Federal Review Group (LFRG), the benefits of using a hybrid approach that provides the benefits of both deterministic and probabilistic assessments to better inform decision-making was discussed.

This document is intended to provide examples of sensitivity and uncertainty analysis approaches that have been employed for PA and PA-like analyses for near-surface facilities by the US Department of Energy (DOE) and US Nuclear Regulatory Commission (NRC). The examples are intended to be summaries that provide general perspective about approaches that have been used without passing judgment regarding a specific case. In addition, examples will be provided that compare and contrast the approaches that have been used. Conclusions are then provided with some recommendations for future needs and a path forward. The emphasis of this document is on applications for near surface disposal applications. The deep disposal programs for the Waste Isolation Pilot Plant and Yucca Mountain include detailed information regarding probabilistic approaches and can be consulted as part of the consideration of future approaches to be applied for near surface disposal. International approaches should also be explored as part of any path moving forward.

2.0 BACKGROUND

Properly addressing uncertainty is of critical importance to communicating human health risk assessment results in a transparent fashion for PAs and PA-like assessments. PAs commonly assess performance for potentially very long time frames in what can often be a combination of engineered and geologic systems, regardless of whether they are conducted for waste disposal, remediation, or decontamination and decommissioning (Brown 2008; IAEA 1995; Kozak et al. 1993; NCRP 2005; Seitz et al. 1992; USNRC 2000; Vovk & Seitz 1995). The fact remains that uncertainties are unavoidable in any site evaluation. Decisions must be made in the face of these uncertainties. Uncertainty and the need for additional information cannot be allowed to delay necessary remedial actions or permit assessors to generate risk information biased by preconceived notions. Therefore, to provide transparency, meaningful exposure, risk, and uncertainty information must be provided as well as input on how these uncertainties might impact the decision-making process.

Two typical ways of classifying uncertainties in health risk assessments like those performed in CERCLA and RCRA can be found in the literature (NAS 1994). One method classifies uncertainties based on where in the risk assessment process they occur (Bogen 1990; NAS 1994). A more common approach categorizes uncertainties into more abstract, general categories. For example, one set of such categories is bias, randomness, and variability (NAS 1994). Another set (i.e., parameter, model, and scenario) was suggested by Linkov and Burmistrov (2003)[1]:

[1] A similar categorization was provided earlier by Konikow and Bredehoeft (1992).

- *Parameter uncertainty*: Lack of knowledge in the "true" value of an input parameter to a model.
- *Model uncertainty*: Lack of knowledge about the structure and accuracy of the model used (including impact of simplifying assumptions and mathematical representations).
- *Scenario uncertainty*: Lack of information regarding missing or incomplete information needed to adequately define the model; this lack of information is sometimes referred to as "modeler uncertainty" (Linkov & Burmistrov 2003).

The first two categories above comprise the preferred taxonomy in the National Academy of Science (NAS) report entitled *Science and Judgment in Risk Assessment* (NAS 1994); however, the third category may be critical and can, in some cases, dominate the overall uncertainty in risk estimates[2]. Kozak et al. (1993) highlighted these broader uncertainties associated with future scenarios and explored potential ways to address these uncertainties as illustrated in Figure 1.

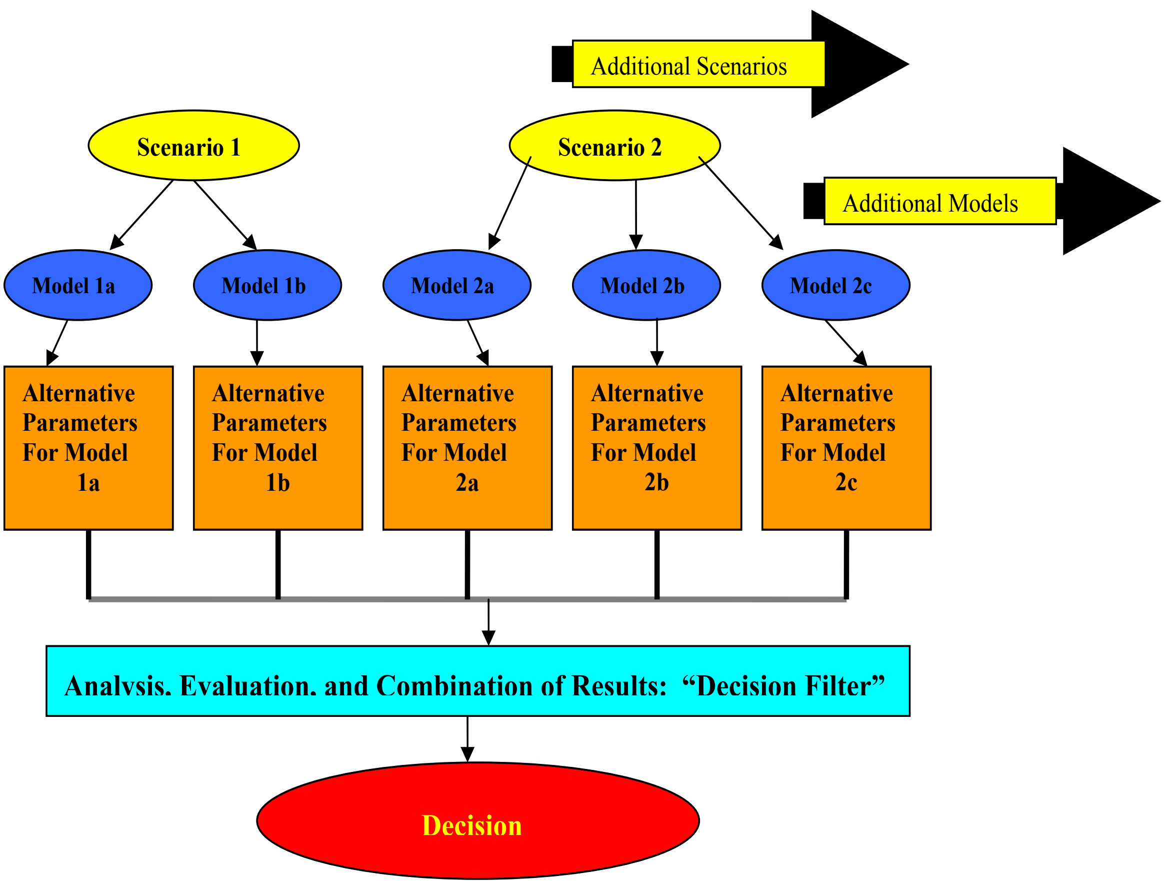

Figure 1. Representation of Approach to Address Scenario and Conceptual Model Uncertainty (Kozak et al. 1993)

[2] One study found that the greatest uncertainty resulted from modeler's interpretation of scenarios resulting in differences in predictions of seven orders of magnitude (Linkov & Burmistrov 2003).

Various other taxonomies for classifying uncertainties have been proposed (Cullen & Frey 1999; Morgan, Henrion & Small 1990; NCRP 2005; Stirling 2003; USDOE 2000b; USEPA 1992; USEPA 1997a; USEPA 1997b; Yoe 1996). One element that runs through these taxonomies and risk assessment is the need for expert judgment to determine the appropriate parameter values, distributions, models, and scenarios. Expert judgment is valuable in that experts often have the greatest experience with these types of problems; however, their judgments often suffer from the same biases as lay people, especially when forced to rely upon intuition (Kahneman, Slovic & Tversky 1982; Slovic 1987; Slovic, Fischhoff & Lichtenstein 1979). Stakeholder input must be included in the process, or there is likely to be a lack of transparency resulting in mistrust of the analysis based upon expert subjectivity or preconceived notions and attitudes.

It is important to consider the challenges specifically associated with development of input distributions to support a probabilistic assessment. This has proven to be the most common aspect resulting in comments on probabilistic PAs for near surface facilities in the DOE system (Seitz et al. 2008). Mishra (2002) includes a number of practical recommendations for development of distributions that are used for probabilistic assessments.

Uncertainties will be a part of any risk assessment (including those relying on point estimates) and cannot be removed entirely from the analysis. However, this does not mean that meaningful estimates and comparisons of risks cannot be made. A better approach is a consistent approach to classifying, estimating, and reducing uncertainties commensurate with their potential impact on the decision-making process (Brown 2008). The use of sensitivity analysis to help prioritize this effort has been a recognized part of PAs for many years (Basalt Waste Isolation Project 1987, Seitz et al. 1992, Vovk and Seitz 1995, IAEA 1995, USNRC 2000, NCRP 2005).

Multiple methods are available for characterizing uncertainties in risk assessments. Two popular methods used in risk assessments are Monte Carlo simulation and sensitivity analysis (USEPA 1989; USEPA 1992; USEPA 1997b; USEPA 2001). For risk assessments, Monte Carlo analysis involves characterizing the uncertainty and variability in risk estimates by repeatedly sampling probability distributions representing risk equation inputs and using the results to estimate the range of risks (USEPA 2001). On the other hand, sensitivity refers to variation in model output with respect to changes in model input(s) and can provide a rank-ordering of model inputs based on their relative contributions to model output variability and uncertainty (USEPA 2001). In addition to evaluating model inputs, sensitivity analysis can be used to develop semi-quantitative bounds on exposure or risk often when information is insufficient to fully describe input distributions but is sufficient to describe input ranges (USEPA 1989). However, limitations on the information used to estimate model input ranges and the impact of the type of sensitivity analysis (e.g., one-input-at-a-time) on the resulting risk or exposure bounds should be described, because such analyses only capture local sensitivities.

The methods for evaluating uncertainties in risk assessments also speak to the types of risk assessments used to support site cleanup activities: *deterministic (often point-value)* and *probabilistic* analyses, a more recent addition to the human health risk assessment landscape (Brown 2008; IAEA 1995; Kozak et al. 1993; NCRP 2005; Seitz et al. 1992; USNRC 2000; Vovk & Seitz 1995). By the early 1990s, most assessments were based on using *point values* intended to result in upper-bound risk estimates (Finley & Paustenbach 1994). However, because of "compounding conservatism" concerns (Burmaster & Harris 1993; Cullen 1994), PA and risk assessors began in the early 1990s to investigate the well-established probabilistic risk analysis (PRA) techniques developed for reactor safety analysis (Keller & Modarres 2005; Kozak et al. 1993; Rechard 1999; Seitz et al. 1992) for probabilistic analyses being conducted in the HLW program. At a national level, the U.S. agencies regulating human health were lagging behind by the mid-1990s; there was no regulatory guidance for performing probabilistic health assessments (Finley & Paustenbach 1994)[3]. However, less than a decade later guidance for introducing probabilistic techniques into human health risk assessment had been provided at both state and federal levels (USEPA 2001; USNRC 2000).

EPA guidance recommends using a tiered and iterative approach that begins with a relatively simple analysis and progresses stepwise to more complex analyses when considering probabilistic techniques to support risk management decisions (USEPA 1997b; USEPA 2001)[4]. This is consistent with approaches that have been recommended for application to PAs as well (Brown 2008; IAEA 1995; Kozak et al. 1993; NCRP 2005; Seitz et al. 1992; USNRC 2000; Vovk & Seitz 1995).

This approach is extended to the overall risk assessment and uncertainty evaluation approach; that is, a point-value analysis should be the starting point for the analysis of exposure and risk. For example, if the results from the point-value analysis clearly indicate that the risks posed by a contaminated site are of no significant impact when considering uncertainties, there is no reason to pursue the probabilistic analysis unless required by the assessor and/or decision-makers. Furthermore, the point-value analysis forms the basis of the Monte Carlo analysis for probabilistic exposure and risk assessment.

Characterizing the properties and reducing uncertainties in understanding and predicting the fundamental behavior of cementitious barriers is needed to evaluate and improve system designs for near surface engineered waste disposal systems, e.g., waste forms, containment structures, entombments, and environmental remediations, and decommissioning activities[5]. Uncertainty reduction should benefit from coupling multi-scale and multi-physics processes, including physical-chemical evolution and transport phenomena applied to heterogeneous, cementitious materials with changing boundary conditions. Ultimately, benefit can be realized by intergrating these processes into a set of tools to predict the structural, hydraulic, and chemical performance of cement-based barriers over extended time frames (e.g., >100 years for operating facilities and > 1000 years for waste management).

3.0 REGULATORY DRIVERS

Performance assessments and PA-like analyses are conducted within a number of different regulatory frameworks. This diversity of regulatory environments often involves different regulators and different analysts conducting assessments for projects for a single facility or site. In order to foster improved consistency and sharing of information, it is important to gain a fundamental understanding of the different regulatory environments that are involved and the analysis expectations within those regulatory environments.

The following sections provide a basic overview of regulations associated with PAs and PA-like analyses and include discussion of guidance or recommendations related to sensitivity and uncertainty analysis. The level of detail provided in each section will vary depending on the level of specific guidance that is available. In many cases, specific guidance for treatment of uncertainty and more specifically in the context of cementitious barriers has not been developed.

[3] At the regional level, the USEPA issued guidance on the use of probabilistic techniques for human health risk assessment as early as 1994 (USEPA 1994; USEPA 2001).

[4] A tiered approach signifies the balance between the benefits of conducting a complex analysis and the costs of the additional time, resources, and challenges for risk communication (USEPA 2001).

[5] The simulation tools will also support analysis of structural concrete components for nuclear facilities (including spent fuel pools, dry spent fuel storage units, and recycling facilities, e.g., fuel fabrication, separations processes).

However, in some regulatory regimes, there has been some guidance provided.

3.1 Performance Assessment Drivers

Performance assessments, or safety assessments as they are termed internationally, are used as a means to quantitatively assess the potential post-closure effects on human health associated with a low-level waste disposal facility. PAs are also a means to make decisions regarding siting, design, operation and development of closure plans for a disposal facility or CERCLA site. Different regulators can be involved depending on the nature for the facility. Generally speaking, post-closure performance of USDOE disposal facilities are regulated under USDOE Orders, USDOE Tank Closures in South Carolina and Idaho are regulated under Section 3116 of the Ronald W. Reagan National Defense Authorization Act for Fiscal Year 2005. Commercial disposal facilities are regulated in accordance with 10 CFR Part 61. The International Atomic Energy Agency (IAEA) publishes recommended standards and guidelines that are not mandatory, but are used as a point of comparison for US activities.

The importance of adequately addressing sensitivity and uncertainty analysis for performance assessment for LLW disposal PAs has been recognized for many years (Seitz et al. 1992, Kozak et al. 1993, Vovk and Seitz 1995, IAEA 1995, USNRC 2000, NCRP 2005). Over this same time frame, the merits of deterministic and probabilistic PAs have also been debated in the context of near-surface waste management activities, and it is recognized that different approaches may be most appropriate for specific problems in the context of a graded approach (NCRP 2005, Seitz et al. 2008).

3.1.1 DOE Order 435.1 and Supporting Manuals: LLW Disposal

3.1.1.1 Assessment Related Requirements

United States Department of Energy Order 435.1, Radioactive Waste Management, is the implementing regulatory document for radioactive waste management activities conducted under DOE authority in accordance with the Atomic Energy Act. The Order itself is very short. Specific requirements related to implementation of the Order are documented in DOE Manual 435.1-1. Chapter IV of DOE M 435.1-1 includes the specific requirements related to siting, design, operation, and closure of disposal facilities for low-level radioactive waste that are regulated under DOE authority. Requirements related to performance assessments and composite analyses to be conducted in support of disposal facilities are addressed in Section IV.P.

The specific requirements in IV.P include deterministic performance objectives for all pathways, air pathway, and for release of radon. The requirements related to performance assessments include, for example: the need to demonstrate compliance with the performance objectives and the need to establish limits on waste concentrations based on the intruder performance measures, identification of a baseline point of compliance, the need to conduct a sensitivity/uncertainty analysis, and the need to address requirements related to protection of water resources. Section IV.P(2)(e) includes the specific requirement to include a sensitivity and uncertainty analysis in the PA.

3.1.1.2 Guidance Related to Sensitivity and Uncertainty Analysis

DOE Guide 435.1-1, Section IV.P(2) includes additional discussion regarding the rationale and expectations for a sensitivity and uncertainty analysis at a relatively high level. There is no specific prescribed approach, but the Guide identifies the importance of identifying the key assumptions relative to the results of the PA and also highlights the importance of providing insights regarding uncertainties associated with the dose projected in the PA. There is no specific recommendation regarding approaches to be used to conduct the sensitivity and uncertainty analysis. This responsibility is left to the analyst.

In the context of cementitious barriers, the primary area of interest tends to be related to the durability of the barriers from both a physical and chemical perspective. Uncertainties can be large when trying to project degradation of barriers over very long time frames, especially considering the large number of processes that can be considered.

3.1.2 NRC 10 CFR Part 61: Commercial LLW Disposal

3.1.2.1 Assessment Related Requirements

NRC regulated LLW disposal facilities must comply with 10 CFR Part 61, which was promulgated in 1982. Part 61 was intended to be applied to commercial LLW disposal facilities and includes requirements for the full lifecycle of a disposal facility. Specific requirements for protection of human health and inadvertent intruders are identified in Subpart C. These requirements form the basis for performance assessment calculations. The specific post closure requirements include dose limits for all pathways of exposure, protection of inadvertent intruders, and minimizing the need for active maintenance after closure.

3.1.2.2 Guidance Related to Sensitivity and Uncertainty Analysis

There are no requirements or recommendations in Part 61 regarding specific approaches to be used for the sensitivity and uncertainty analysis. Supporting calculations for Part 61 were conducted on a deterministic basis and there is a requirement in Part 61 that a site is capable of being modeled. Thus, there is no prescribed approach. NRC Staff published NUREG-1573, *A Performance Assessment Methodology for Low-Level Waste Disposal Facilities – Recommendations of NRC's Performance Assessment Working Group* (USNRC 2000). This document includes NRC Staff perspectives regarding approaches for conducting performance assessment calculations. The NUREG is not a regulatory document and is not binding, but does reflect NRC Staff perspectives on acceptable approaches and provides insight into what would be expected in a PA.

Sensitivity and uncertainty analysis was flagged as one of five key issues in the document. In Section 3.2.4 of NUREG-1573, NRC Staff provide perspective on the need for sensitivity and uncertainty analysis in a PA. There is an introductory discussion of the different types of uncertainties inherent in PA calculations, followed by a discussion of the role of sensitivity and uncertainty analysis as a part of the process of interpreting results and optimizing strategies for building confidence in compliance demonstrations.

A flexible approach is advocated for sensitivity and uncertainty analysis recognizing the potential use of deterministic and probabilistic approaches to address uncertainty. The importance of considering different conceptual models and using sensitivity analysis to identify assumptions that should be the focus of additional work is emphasized. For the compliance demonstration using a probabilistic approach, it is recommended that the entire distribution be evaluated, but the emphasis of compliance should be the peak of the mean dose curve compared against the performance objectives from Part 61.

Section 3.3.2 of NUREG-1573 includes more detailed suggestions for sensitivity and uncertainty analysis. There is substantial emphasis placed on providing a range of potential outcomes, including the need to address model and scenario uncertainties, which can be the most significant uncertainties in a PA. Several references are also provided for examples of quantitative approaches to address parameter uncertainty, which is the more common aspect of traditional uncertainty analyses. It is emphasized that there is no universal "best" approach for conducting sensitivity and uncertainty analyses for PAs. The merits of deterministic and probabilistic approaches are discussed with cautions regarding the use of each approach. There is also a discussion of considerations for the conduct of parametric sensitivity analyses.

NUREG-1573 includes numerous references to additional information regarding more detailed approaches for sensitivity and uncertainty analysis.

3.1.3 NDAA Section 3116: HLW Tank and Facility Closures

3.1.3.1 Assessment Related Requirements

Final disposition of HLW remaining after tank closure as LLW is regulated under the Ronald W. Reagan National Defense Authorization Act for Fiscal Year 2005 (Section 3116). Section 3116 is very short and specifies that the performance objectives from Subtitle C of Part 61 must be met in order for the residues remaining at the time of closure activities to be managed as LLW. The NRC is assigned monitoring responsibilities to ensure that DOE has demonstrated that the objectives in Subtitle C will be met. These requirements were described in Section 2.1.2.1.

3.1.3.2 Guidance Related to Sensitivity and Uncertainty Analysis

There is no specific requirement in Section 3116 for the conduct of sensitivity and uncertainty analysis. However, NRC Staff prepared Draft Final NUREG-1854, *NRC Staff Guidance for Activities Related to U.S. Department of Energy Waste Determinations.* NUREG-1854 includes recommendations for reviews of PAs conducted for Section 3116 issues. Sensitivity and uncertainty analysis is addressed in Section 4.5 of NUREG-1854 and there is a discussion of probabilistic and deterministic modeling approaches in Section 4.4.1.1. Emphasis is placed on the preference for a "risk-informed" approach for PA using probabilistic sampling for modeling parameters with irreducible uncertainty. Nevertheless, NRC Staff indicate that a deterministic approach is acceptable for demonstrating compliance with performance objectives. However, such an approach should be supported with a demonstration that uncertainties have been suitably addressed.

Section 4.5 of NUREG-1854 discusses considerations for reviews of sensitivity and uncertainty analysis aspects of a PA. The preference for probabilistic approaches is reinforced in this discussion. In respect to recommended approaches, references are provided to NUREG-1573 as discussed above and also to NUREG-1757 (Vol. 2, Appendix I, Section 1.7) (USNRC 2003a). The importance of using the results of a sensitivity analysis to focus the review on important parameter and model assumptions is also emphasized. The concept of "risk dilution" is introduced as a caution against using overly broad distributions for input parameters. The choice of distribution type and metrics for input distributions is identified as a key area for reviews as well as the need to consider alternative conceptual models, as appropriate.

3.1.4 International Atomic Energy Agency

3.1.4.1 Assessment Related Requirements

The International Atomic Energy Agency (IAEA) publishes non-binding requirements related to radioactive waste safety and guidance for implementation. Internationally, the term Safety Assessment is used rather than Performance Assessment. In 1999, the IAEA published a safety requirements document on Near Surface Disposal of Radioactive Waste and a safety guide on Safety Assessment for Near Surface Disposal of Radioactive Waste. The Safety Requirement is intended to establish requirements that must be met to ensure safety. These are non-binding, but are often cited as examples for what needs to be included in regulations.

The Safety Requirement sets out the dose objectives and identifies the need to conduct a safety assessment to demonstrate the ability of the facility to meet the dose objectives. The dose objectives are expressed in a deterministic manner without further elaboration regarding how to interpret results of a quantitative uncertainty analysis in the context of the deterministic standard. Uncertainties regarding human behavior in the future are addressed by specifying that current human habits should be used as the basis for projections of exposures and doses in the future.

Updated requirements and safety guides are in the process of being developed but are still in draft form.

3.1.4.2 Guidance Related to Sensitivity and Uncertainty Analysis

The Safety Requirement described in Section 2.1.4.1 is written at a high level intended to mimic the level of detail in a regulation, and thus, does not include any specific guidance regarding how to use or conduct sensitivity and uncertainty analyses. The Safety Guide on Safety Assessment identifies the need for sensitivity and uncertainty analysis to quantitatively address uncertainties inherent in the process and notably, the use of sensitivity analysis to identify important features of the system that may require more detailed consideration.

The Safety Guide includes a summary of key considerations for sensitivity and uncertainty analyses, including different types of uncertainty that need to be addressed (e.g., parametric, scenario, conceptual and future conditions). There is a brief high-level discussion of approaches for conducting sensitivity and uncertainty analyses. The use of Monte Carlo type analyses is identified as an option for conducting uncertainty analyses as well as simple one parameter at a time sensitivity analyses as a more direct approach. Emphasis is placed on avoiding extreme combinations of input parameters and assumptions. The importance of the need to defend input distributions for a Monte Carlo-type approach to uncertainty analysis is also stressed.

3.1.5 NCRP Guidance on PA for LLW Disposal

In 2006, the National Council on Radiation Protection and Measurements (NCRP) issued NCRP Report Number 152, *Performance Assessment of Near-Surface Facilities for Disposal of Low-Level Radioactive Waste* (NCRP 2005). It is provided as a technical resource and does not carry any regulatory authority.

3.1.5.1 Assessment Related Requirements

The NCRP does not establish requirements for PAs. However, in their guidance document, the NCRP reviews concepts underlying PAs for LLW disposal and approaches to conducting such assessments.

3.1.5.2 Guidance Related to Sensitivity and Uncertainty Analysis

The NCRP guidance includes some detailed discussions of considerations for conducting sensitivity and uncertainty analyses. There is a significant discussion of the merits of deterministic and probabilistic approaches that provides insights into the challenges associated with each method. The report includes a discussion of the role of importance analysis as a specific application of sensitivity analysis that focuses on parameters that will change conclusions of the assessment rather than simply addressing sensitive parameters. The report also recommends considering the use of both deterministic and probabilistic approaches to gain insights into performance of the system. This has been more recently referred to as a hybrid approach (Seitz et al. 2008).

3.2 Performance Assessment-Like Analysis Drivers

The cornerstones of the U.S. Department of Energy's authority to manage and regulate radioactive wastes are the Atomic Energy Act (AEA) and Nuclear Waste Policy Act (NWPA). However, other legislation including the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA), Resource Conservation and Recovery Act (RCRA), and National Environmental Policy Act (NEPA) as well as correlative state and local laws may play critical regulatory roles. These additional statutes often go well beyond the AEA, NWPA, or Section 3116 of the

NDAA. Perhaps more importantly, these other laws are not administered by the USDOE but instead by the U.S. Environmental Protection Agency (USEPA) and by the states (NAS 2006).

Uncertainties are present in nearly every aspect of an environmental restoration or facility disposition (USDOE 2000a). Primary uncertainties often include how contaminated media or those wastes generated during a project must be managed. However, other important uncertainties can be technical in nature (e.g., contaminants present or extent of contamination) or regulatory (e.g., will wastes meet RCRA land disposal restrictions) or programmatic (e.g., is additional funding available if new regulatory obligations are found) (USDOE 2000a).

There are no formal requirements for the management of uncertainties in CERCLA, RCRA, NEPA, or the NRC License Termination Rule (LTR). Instead guidance has been developed by the USEPA (administors of CERCLA, RCRA, and NEPA) and the USNRC (for the LTR and other regulations) that directs how uncertainties should be managed under the various processes involved. For example, one area in which uncertainties play a major role is in assessments where exposures of receptors to contaminants from regulated sites are estimated for conversion to dose for the USNRC or risk for the USEPA.

Traditional risk assessments have been based on "deterministic" or point-value techniques intended to produce bounding or "conservative" estimates of exposure and risk (Lester, Green & Linkov 2007). For these types of assessments, the analysis of uncertainty is typically restricted to a qualitative or semi-quantitative evaluation perhaps including sensitivity analyses. Probabilistic techniques began to be used in the 1990s because of concerns of "compounding conservatism" introduced into estimates of exposure and risk (Burmaster & Harris 1993; Cullen 1994). According to Lester, et al. 2007, the primary Federal regulatory drivers for the use of formal probabilistic analysis techniques for "influential risk assessments" are recent guidance documents from the U.S. Office of Management and Budget (USOMB) (OMB 2003; OMB 2006). However, despite the guidance from the USOMB and recognition by the USDOE of the importance of probabilistic techniques (Brewer et al. 2003; USDOE 1993), it appears that probabilistic risk assessment has not made significant inroads into the USDOE for risk assessments for sites regulated under CERCLA.

3.2.1 CERCLA

In 1980 the U.S. Congress enacted and the President signed into law the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) (Pub. L. 96510) to identify and remediate sites where hazardous substances were, or could be, released into the environment (USDOE 1994a). CERCLA was amended by the Superfund Amendments and Reauthorization Act of 1986 (SARA) (Pub. L. No. 99-499). CERCLA applies to all Federal agencies (USDOE 1994a).

3.2.1.1 Requirements for Management of Uncertainty

Various assessments are required under CERCLA including baseline risk assessments, remedial investigations, and feasibility studies (USDOE 1994a). Risk estimates made in CERCLA assessments are conditional on assumptions and simplifications made throughout the assessment process. Uncertainties in these risk assessments result from dynamic variability in natural systems, variability in human behavior and physiology, and the methods designed to characterize both for prediction purposes (USDOE 1995). Examples of typical sources of uncertainty found in CERCLA risk assessments are provided in Table 1 as well as the likely impact of the various assumptions required to address common information gaps (USDOE 1995). Numerous assumptions must be made to develop conceptual models and select assessment model and input parameters.

Table 1. Typical Sources of Uncertainty in CERCLA Risk Assessments (reproduced from USDOE 1995)

Data Gaps/Uncertainty	Typical Assumptions	Likely Impact on Risk Estimate
Hazard Identification		
Insufficient number of samples	Use of various estimation methods	Overestimation
High detection limits	Contaminant level below detection limit	Underestimation
Contaminant degradation during sampling	Degradation occurs	Underestimation
Exposure Assessment		
Limited information on intake factors, population characteristics, exposure duration, etc.	Various assumptions required	Overestimation and/or underestimation
Limited or no chemical bioavailability data	100% bioavailability	Overestimation
Limited or no data on degradation, transformation, and fate of chemicals	No degradation and/or transformation	Overestimation and/or underestimation
Limited dermal absorption factors	Conservative default factors	Overestimation
Toxicity Assessment		
Toxicity values for low doses in humans derived from high doses in animal studies	Linearity of dose-response curves at low doses	Overestimation and/or underestimation
Limited information on shape of carcinogenic dose-response curve at low doses	95% upper confidence limit on cancer slope factors	Overestimation
Risk Characterization		
No toxicity information on individual chemicals	Use of reference doses (RfDs) and cancer slope factors of similar chemicals	Overestimation
No toxicity information on individual chemicals	Not factored into quantitative analysis	Underestimation
No interactive toxicity information on mixtures of chemicals	Dose additivity	Overestimation if antagonistic interaction; underestimation if synergistic interaction
Limited quality and size of sources of information	Quantification of risks, but no quantitative analyses of uncertainty possible	Risk assessment open to differing interpretations

CERCLA and SARA provide no specific guidance on how to address uncertainties in baseline or other risk assessments (USDOE 1995). However, guidance documents have been developed to incorporate uncertainty analysis in CERCLA risk assessments (USEPA 1988; USEPA 1989; USEPA 1991a; USEPA 1991b; USEPA 1997a; USEPA 1998; USEPA 2001; USEPA 2004). These documents acknowledge the impacts of missing and uncertain information on exposure and risk estimates as well as the impacts associated with the assumptions and simplifications that must be made to manage missing and uncertain data and the models used to estimate exposure and risk (USDOE 1995). These USEPA guidance documents suggest procedures for managing uncertainties; however, the suggestions are general in nature and do not provide for specific methodology.

3.2.1.2 Guidance for Cementitious Barriers and Uncertainty

There are no specific requirements or recommendations in CERCLA or SARA regarding assessment or uncertainty approaches when cementitious barriers are used for remedial purposes. Credit may be taken for waste forms and barriers when projecting exposure media concentrations and risk into the future. However, this credit likely adds complexity and model uncertainty to the situation, which must be accounted for in the decision-making process (USEPA 1989). The evaluation of the potential impacts of uncertainties related to cementitious barriers and their remedial uses should follow the more general guidance developed by the USEPA (USEPA 1988; USEPA 1989; USEPA 1991a; USEPA 1991b; USEPA 1997a; USEPA 1998; USEPA 2001; USEPA 2004). One goal of the CBP is to allow more accurate predictions to be made when cementitious barriers are used in disposal.

3.2.1.3 Perspective on How Often Cementitious Barriers and Uncertainties are Modeled

By the early 1990s, most human health risk assessments were based on calculating point values intended to represent upper-bound risk estimates (Finley & Paustenbach 1994) with either qualitative or semi-quantitative uncertainty analyses. In fact probabilistic techniques for human health risk assessment are recent additions to the human health assessment landscape (Brown 2008). Concerns of "compounding conservatism" led assessors to investigate the well-established probabilistic techniques developed for reactor safety (Keller & Modarres 2005; Rechard 1999) in order to provide more comprehensive and meaningful information for decision-makers.

A review was performed of available literature (e.g., records of decision or RODs) concerning remedial alternatives considered and finally selected (as well as the corresponding uncertainty analyses) for various Superfund sites. For example, one summary of 30 RODs for CERCLA landfills was conducted (USEPA 1993) and, of these 30 decisions, a grout curtain or grout injection was considered in 26 instances but these options were screened out in every case based on the CERCLA cost, effectiveness, and implementation criteria. A similar study was performed by the authors to examine remedial alternatives for Idaho Superfund sites, especially those involving the Idaho Site. Of the 22 RODs involving the Idaho Site (USEPA CERCLIS ID 4890008952), seven involved consideration of cementitious barriers (primarily grouting) and three remedies were selected. The risk evaluations were based on point-value analyses supplemented by semi-quantitative sensitivity analyses to evaluate the impacts of uncertainties on the results.

3.2.2 RCRA

The Resource Conservation and Recovery Act (RCRA) (Pub. L. 94-580) was signed into law in 1976 to protect human health and the environment using a comprehensive approach to hazardous and solid waste management at operating facilities (USDOE 1994a). In 1984, Congress amended RCRA with the Hazardous and Solid Waste Amendments (HSWA) to help reduce the total quantity of hazardous waste generated and to help prevent releases of such wastes into the environment (Pub. L. 98-616).

3.2.2.1 Requirements for Management of Uncertainty

The assessments required under Resource Conservation and Recovery Act (RCRA): include facility assessments and investigations, corrective measures studies, and selections and implementations of the corrective measures. These analyses can be considered analogous in many ways to those in CERCLA (USDOE 1994a; USDOE 1994b)[6]. Furthermore, the risk analyses needed in the RCRA assessment process are also analogous to those described above for CERCLA assessments. Examples of typical sources of uncertainty found in CERCLA risk assessments were provided in Table 1 (USDOE 1995). They are also relevant for RCRA risk assessments[7]. Numerous assumptions must be made to develop conceptual models and select assessment models and input parameters.

RCRA and Hazardous and Solid Waste Amendment (HSWA) provide no specific guidance on how to address uncertainties in risk evaluations. The guidance documents that were developed to address uncertainty in CERCLA risk assessments (USEPA 1988; USEPA 1989; USEPA 1991a; USEPA 1991b; USEPA 1997a; USEPA 1998; USEPA 2001; USEPA 2004) also apply to the RCRA process. These documents acknowledge the impacts of missing and uncertain information on exposure and risk estimates and the impacts associated with the assumptions and simplifications that must be made. They also suggest procedures for managing uncertainties. The recommendations in the guidance documents are general in nature and do not provide for a specific methodology.

3.2.2.2 Guidance for Cementitious Barriers and Uncertainty

Like under CERCLA, there are no specific requirements in RCRA and HSWA or recommendations in EPA guidance documents regarding assessment or uncertainty approaches when cementitious barriers are used for remedial purposes. However, credit may be taken for waste forms and barriers when projecting exposure media concentrations and health risk into the future. However, this credit likely adds complexity and model uncertainty to the assessment, which must be accounted for in the decision-making process (USEPA 1989). The evaluation of the potential impacts of uncertainties related to cementitious barriers and their remedial uses should follow the general guidance developed by the USEPA.

3.2.2.3 Perspective on How Often Cementitious Barriers and Uncertainties are Modeled

In RCRA assessments, there are a number of steps where cementitious barriers and associated uncertainties may be considered. For health risk assessments, any contaminants of potential concern that may be in a cementitious waste form, contained in a cement-based container or structure, or both may be accounted for in terms of contaminant release and transport

[6] The USEPA has suggested that the RCRA corrective action is substantially "equivalent" to the CERCLA site investigation/remediation process (USDOE 1994b).

[7] Because various environmental regulations may apply to the disposition of a contaminated site, the USDOE and various Sites have developed strategies to integrate actions under the various laws including CERCLA, RCRA, and NEPA (Cook 2002; Shedrow, Gaughan & Moore-Shedrow 1993).

like their CERCLA counterparts. For example, the Waste Calcining Facility (WCF)[8] at the Idaho National Engineering and Environmental Laboratory (INEEL) was closed under an innovative approach for closing a nuclear facility at the INEEL (Demmer et al. 1999)[9].

Because it was deemed impractical to remove the process residues, decontaminate the equipment, and remove the filters in the waste pile, the WCF closure was developed in accordance with the closure and post-closure requirements applying to landfills (Demmer et al. 1999). The risk assessment took credit for the concrete cap and grout placed in the WCF to estimate risks to receptors. The potential impacts of uncertainties were introduced in the risk assessment by making conservative assumptions and further relying on semi-quantitative sensitivity analyses. This risk assessment approach was found to be typical of the RCRA closures for the DOE sites. In general, the impact of cementitious barriers were included in the risk analysis and conservative assumptions and semi-quantitative sensitivity analyses were used to evaluate the impacts of uncertainties on the predicted risks to important receptors.

3.2.3 National Environmental Policy Act (NEPA)

The National Environmental Policy Act (Pub. L. 91-190) was the first of the major environmental laws enacted in the U.S. Growing concerns about environmental pollution and quality were encapsulated in NEPA, which was the foundation for inserting environmental considerations into federal decision-making (Bear 1989). NEPA established the U.S. national environmental policies (CEQ 2007).

Because various environmental regulations may apply, USDOE and its Sites have developed strategies to integrate actions under the various laws including CERCLA, RCRA, and NEPA (Cook 2002; Shedrow, Gaughan & Moore-Shedrow 1993). NEPA reviews are required for siting, construction, and operation of treatment, storage, and disposal facilities that, in addition to supporting CERCLA actions, also serve waste management or other purposes (Cook 2002; USDOE 1994c). For example, the Savannah River Site (SRS) strategy tiers RCRA/CERCLA activities to NEPA reviews and integrates elements of the NEPA and RCRA/CERCLA processes, where applicable (Shedrow, Gaughan & Moore-Shedrow 1993). USDOE typically relies on the CERCLA process for review of actions taken under CERCLA—no separate NEPA process is typically required (Cook 2002)[10]. USDOE addresses NEPA values in the CERCLA process by including a discussion of environmental impacts in CERCLA documents and taking steps to ensure early public involvement in the process.

3.2.3.1 Regulatory Requirements for Management of Uncertainty

The foremost technical difficulty posed to decision-makers when considering risks is pervasive uncertainty in estimates of the effects associated with exposure to a contaminant, the economic effects of a proposed regulatory action, or extent of current and possible exposures to receptors (NAS 1983; NAS 1994). This difficulty has no foreseeable resolution when

[8] This Waste Calcining Facility (WCF) at the INEEL is often referred to as the "Old Waste Calcining Facility" in deference to a newer calcining facility.

[9] Previous closures of nuclear facilities focused on decontamination and removal of equipment and structures, which involved extensive removal, packaging of wastes, and remediation of the area (Demmer et al. 1999). Since the WCF was included on the INEEL RCRA Part A permit application, a closure plan was required. Because the WCF could not be decontaminated, the systems were closed in accordance with landfill requirements. The DOE evaluated the WCF landfill closure using an Environmental Assessment (EA) to evaluate exposure risks

[10] The DOE approach to NEPA review for RCRA corrective actions tend to be project-specific where most DOE RCRA actions have fallen within the scope of a categorical exclusion (Cook 2002). When proposed RCRA actions have not qualified for a categorical exclusion, DOE has often been able to rely on the CERCLA process.

considering the many gaps in knowledge (e.g., causal mechanisms of carcinogenesis or cumulative effects) that remain despite new scientific information (NAS 1994). The systematic analysis of the uncertainties in the risk analyses can provide a framework for evaluating the potential impacts of the uncertainties on the decision-making process.

The assessments required under NEPA include analyses resulting in 1) CATegorical EXclusion (CATEX) for those actions deemed to not have a significant effect, 2) environmental assessments (EA) when there is uncertainty concerning the environmental impacts of the proposed action, and 3) environmental impact statement (EIS) for any proposed major federal action that may significantly affect the quality of the human environment (CEQ 2007). There are no specific requirements in NEPA concerning uncertainty analysis during the NEPA assessment process. The methodology for addressing uncertainties, either qualitatively or quantitatively, is within the purview of the lead agency[11]. However, the United States Office of Management and Budget has proposed that uncertainty be characterized with respect to the major findings and that the nature and quantitative implications of model uncertainty be disclosed and a sensitivity analysis be performed (USOMB 2006).

3.2.3.2 Guidance for Cementitious Barriers and Uncertainty

Like CERCLA and RCRA, there are no specific requirements or recommendations in NEPA regarding the approaches that must be used for the assessment of cementitious barriers and the impacts of the resulting uncertainties. However, NEPA does require that all "reasonable" alternatives, including those incorporating barriers or grouting, be considered during the EIS process[12]. Credit can be taken for waste forms and barriers when predicting exposures and risks although any increases in modeling complexity and uncertainty should be taken into account in the decision-making process[13]. One goal of the CBP is to allow more accurate predictions to be made when cementitious barriers are considered in proposed Federal alternatives.

3.2.3.3 Perspective on How Often Cementitious Barriers and Uncertainties are Modeled

In the NEPA assessment process, the EIS is the most likely stage where cementitious barriers and the uncertainties from their use may be considered. Available EAs for SRS, Hanford, and the Idaho Site were reviewed and none contained reference to either cementitious barriers or uncertainty analysis. On the other hand, the available Final EISs[14], the focal point of which is a detailed analysis of the potential impacts of proposed actions, were examined for the Savannah River, Hanford, and Idaho Sites. Of the Final EISs described in Table 2, cementitious barriers are considered as alternatives (or incorporated into the alternatives considered) in all but one of the Final EISs for the three sites (i.e., DOE/EIS-0222 for Hanford). For these Final EISs, the typical method of managing uncertainties is to evaluate conditions that are intended to provide bounding estimates of environmental impacts.

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[11] Originally, NEPA required that a "worst-case" analysis be performed, but that requirement was replaced in 1986 with a process for evaluating "reasonably foreseeable" impacts (Bear 1989).

[12] The EAs available on the USDOE site (http://www.gc.doe.gov/NEPA/environmental_assessments.htm accessed March 17, 2009) for SRS, Hanford, and the Idaho Site were examined. There were no discussions of uncertainty or references to cementitious barriers in these brief assessments.

[13] For example, the Final Hanford Site Solid Waste Program EIS describes alternatives incorporating cementitious barriers (i.e., grouting) and a detailed analysis of uncertainty management (USDOE-RO 2004).

[14] The Final Environmental Impact Statements (EISs) related to the U.S. Department of Energy are available at http://www. gc.doe.gov/NEPA/final_evironmental_impact_statements.htm (accessed on March 17, 2009).

Table 2. Final Environmental Impact Statements Related to the Savannah River, Hanford, and Idaho Sites (http://www.gc.doe.gov/NEPA/final_ environmental_impact_statements.htm)

EIS Number	Site	Title	Cementitious Barriers Considered	Uncertainty Approach for Barriers
DOE/EIS-0189	Hanford	Final Environmental Impact Statement for the Tank Waste Remediation System (08/1996)	Grouting tank wastes and tank farms	Bounding approach for accidents and sensitivity analyses for risks including Monte Carlo
DOE/EIS-0212	Hanford	Final Environmental Impact Statement Safe Interim Storage of Hanford Tank Wastes (10/1995)	Grouting option dismissed due to potential impact on future decisions	Not applicable
DOE/EIS-0222	Hanford	Final Hanford Comprehensive Land-Use Plan Environmental Impact Statement	No discussion of cementitious barriers	Not applicable
DOE/EIS-0244	Hanford	Final Environmental Impact Statement - Plutonium Finishing Plant Stabilization (05/1996)	Cementing plutonium-containing liquid effluents	Only maximally exposed individual doses and health effects
DOE/EIS-0286F	Hanford	Final Hanford Site Solid (Radioactive and Hazardous) Waste Program Environmental Impact Statement Richland, Washington (01/2004)	Interim storage of immobilized low-activity waste (ILAW) in grout vaults and trenches	Bounding, sensitivity, and stochastic analyses
DOE/EIS-0287	Idaho	Idaho High-Level Waste & Facilities Disposition, Final Environmental Impact Statement (09/2002)	Grouting of low-level wastes, tank heels, and newly-generated liquid wastes	Accidents at least as severe as “reasonably foreseeable” and includes both sensitivity and uncertainty analyses
DOE/EIS-0290	Idaho	Idaho National Engineering and Environmental Laboratory Advanced Mixed Waste Treatment Project Environmental Impact Statement (01/1999)	Macroencapsulation into a grout waste form (which would then be drummed for disposal)	Conservative assumptions and analytical approaches used to produce a credible projection of the bounding potential environmental impacts
DOE/EIS-0303	SRS	The Savannah River Site High-Level Waste Tank Closure Final Environmental Impact Statement (05/2002)	Grouting tank farms	Accidents at least as severe as “reasonably foreseeable” and scenario-based analysis

3.2.4 USNRC License Termination Rule, 10 CFR Part 20 Subpart E

The U.S. Nuclear Regulatory Commission (USNRC) grants licenses to companies for the commercial operation of nuclear reactors and radiological facilities[15]. Any company holding such a license must seek NRC permission to decommission the facility. For a power reactor, a Post-Shutdown Decommissioning Activities Report (PSDAR) must be submitted that includes a discussion of how environmental impacts will be bounded by pertinent environmental impact statements. For a power reactor, the licensee must submit an application for termination of its license for NRC approval as well as a license termination plan (LTP). The licensee must demonstrate that the requirements of the License Termination Rule (LTR) (10 CFR §20.1401 *et seq.*) will be satisfied.

For a radiological material site licensed by the USNRC, a decommissioning plan (DP) is submitted to the NRC if required. Once the licensee demonstrates compliance with its decommissioning plan, it must then request license termination from the NRC for *unrestricted* or restricted release. For *unrestricted* release, a full technical review guided by NUREG-1757 (USNRC 2003a; USNRC 2003b; USNRC 2003c) is undertaken with results documented in an Environmental Assessment (EA) and a Safety Evaluation Report (SER). For plans proposing restricted release for material sites, the review is conducted in two phases. The first phase focuses on the financial assurance and institutional control provisions of the plan. After these provisions are found to comply with the LTR, the remainder of the review is completed to address the rest of the technical review guided by NUREG-1757 including an EIS.

3.2.4.1 Regulatory Requirements for Management of Uncertainty

The primary assessment required under the LTR (10 CFR §20.1401 *et seq.*) is the assessment of predicted dose for restricted release (10 CFR §20.1403) or unrestricted release (10 CFR §20.1402) of facilities licensed by the NRC (10 CFR §20.1401). A site is acceptable for *unrestricted release* if the residual radioactivity[16], translates to a total expected dose equivalent (TEDE) to an average member of the critical group from all sources that does not exceed 0.25 mSv (25 mrem) per year (10 CFR §20.1402). A site will be considered acceptable for *restricted release* if the licensee meets several LTR conditions (10 CFR §20.1403(a)-(e)). The licensee can use either conservative default scenarios for on-site use or site-specific models for more realistic scenarios for the dose assessments (USNRC 2004).

There are no legal requirements in the LTR for how uncertainties must be addressed in the dose assessment. However, the NRC guidance states that the licensee should include a discussion of effects of uncertainties on the predicted dose results (NRC 2003a; NRC 2003b)[17]. The NRC also discusses the use of uncertainty and sensitivity analyses as a means to focus on parameters important to the dose assessment (USNRC 2003b).

3.2.4 2 Guidance for Cementitious Barriers and Uncertainty

Like CERCLA, RCRA, and NEPA, there are no specific requirements in the LTR regarding the approaches that must be used for the assessment of cementitious barriers and the impacts of the resulting uncertainties. However, unlike these laws

[15] The NRC does not have regulatory authority over defense nuclear facilities.
[16] ALARA considerations must be taken into account for these assessments.
[17] The uncertainty in engineered barrier performance should also be accounted for in designing the long-term monitoring strategy (USNRC 2003b).

administered by the EPA, the LTR provides specific guidance for the assessment of the performance of engineered barriers including: (a) design and functionality, (b) technical basis for design and functionality, (c) degradation mechanisms and sensitivity analysis, (d) uncertainty in design and functionality, and (e) suitability of numerical models (USNRC 2003b). The assessment of the barrier performance for unrestricted release should evaluate potential breach and degradation processes over time (including uncertainties) because monitoring and maintenance are assumed to be inactive.

When considering complex and high-risk decommissioning sites and those sites with long-lived radionuclides, the NRC suggests employing probabilistic analyses (NRC 2003a)[18]. Point-value analyses may be inadequate in these cases. For simpler, low-risk sites and those with short-lived radionuclides, point-value analysis with sensitivity analysis may be sufficient (NRC 2003a).

For engineered barriers that must have very long-term performance, natural analogs should be considered because the greatest uncertainties result from extrapolating short-term information to long-term performance (NRC 2003a). The behavior of the barrier should be considered an evolving component of a larger, dynamic ecosystem (Waugh, Weston & Richardson 1997). Table 3 summarizes selected guidance and reference reports that may have relevance to the application of engineered barriers at decommissioning sites (USNRC 2003a).

The USSNRC provides specific guidance for cement-based engineered barriers. The performance of these barriers can be divided into those based on either 1) hydrologic effectiveness or physical containment to reduce water contact or 2) chemical effectiveness to limit radionuclide transport (Waugh, Weston & Richardson 1997). Concrete degradation mechanisms (e.g., sulfate attack, chloride corrosion, and cracking) can cause contact of water with the waste and corresponding contaminant release (USNRC 2003a). For chemical containment, the effectiveness of cement-based materials strongly depends on the source release characteristics; performance is very difficult to predict and is strongly related to bulk hydraulic properties and quantity of cement present (USNRC 2003a). A cement-based barrier may also limit intruder contact with waste for up to hundreds of years if it remains unexposed to aggressive environmental conditions (USNRC 2003a). Because the performance of the cement-based engineered barriers may have to be assessed over hundreds if not thousands of years, the aforementioned uncertainty issues for cement-based barriers are likely critical to the assessment.

3.2.4.3 Perspective on How Often Cementitious Barriers and Uncertainties are Modeled

The USNRC regulates the release of contaminated solid materials including building concrete from licensed facilities on a case-by-case basis (NAS 2002; USNRC 2003b). Such material can be removed if the facility license is terminated based on meeting the LTR dose limit for unrestricted use (10 CFR §20.1402). However, before license termination, solid material including concrete can only be released if it satisfies the "few mrem/yr criterion" (NRC 2004). For retrospective cases involving concrete disposition, if offsite releases were performed in an approved manner, these releases should be considered final. For prospective cases, disposition of concrete with volumetric contamination may be approved under the "few mrem" criterion rather than use of the LTR criteria (NRC 2003b). At materials sites, disposition of concrete with surficial contamination is evaluated using the appropriate NRC guidelines (NRC 1993); disposition of concrete with volumetric contamination follows 10 CFR 20.2002. If the licensee proposes to

[18] Point value methods are suggested for selecting the design flood for the development of long-term erosion controls (USNRC 2003a).

Table 3. Summary of Selected Reports Related to Engineered Barriers (reproduced from USNRC 2003a)

Report	Brief Summary
NUREG/CR-5542, "Models for Estimation of Service Life of Concrete Barriers in Low-Level Radioactive Waste Disposal," U.S. Nuclear Regulatory Commission, Washington, DC, September 1990.	Provides primarily empirically based models for typical concrete formulations to estimate degradation rates.
NISTIR 89-4086, NUREG/CR-5466, "Service Life of Concrete," National Institute of Standards and Technology (NIST) Gaithersburg, MD, 1995.	Examines degradation processes in cement-based materials and discusses considerations of their occurrence, extent of potential damage, and mechanisms.
NISTIR 7026, "Condition Assessment of Concrete Nuclear Structures Considered for Entombment," National Institute of Standards and Technology (NIST), Gaithersburg, MD, 2003.	Provides assessment of cement-based engineered barrier structures based on characterization of intact concrete and crack properties. Material property uncertainties are incorporated into a Monte Carlo simulation.
NISTIR 6747, "Validation and Modification of the 4SIGHT Computer Program" National Institute of Standards and Technology (NIST) Gaithersburg, MD, 2001.	Discusses the validation and verification of the fluid transport mechanisms incorporated in the concrete degradation code 4SIGHT using reference and laboratory data.
NISTIR 6519, "Effect of Drying Shrinkage Cracks and Flexural Cracks on Concrete Bulk Permeability," National Institute of Standards and Technology (NIST) Gaithersburg, MD, 2000.	Discusses a model for predicting both the width and spacing of flexural and drying-shrinkage cracks to estimate composite (intact and cracked) concrete structure permeability.
NISTIR 5612, "4SIGHT, Manual: A Computer Program for Modeling Degradation of Underground LLW Concrete Vaults," National Institute of Standards and Technology (NIST) Gaithersburg, MD, 1995.	User Manual for numerical computer modeling of concrete degradation, 4SIGHT, to facilitate assessment of concrete vaults for isolating radioactive waste in Low Level Waste (LLW) disposal applications.
"Barrier Containment Technologies for Environmental Remediation Applications," edited by Ralph R. Rumer and Michael E. Ryan, John Wiley and Sons, 1995.	Review and evaluation of knowledge and practices of containment technologies suitable for remediation. Identifies areas where practical improvements could be developed.
National Research Council, National Academy of Sciences, "Barrier Technologies for Environmental Management," Summary of a Workshop, 1997.	Papers presented in the Workshop on the use of Engineered Barriers to prevent the spread of contaminants and its migration.
"Field Water Balance of Landfill Final Covers," Albright, W, Benson, C., Gee, G., Roesler, A., Abichou, T., Apiwantragon, P., Lyles, B., and Rock, S., Journal of Environmental Quality, 33(6), 2317-2332, 2004.	Results of large-scale field research study to assess the ability of landfill final covers to control infiltration into underlying waste. A comprehensive current publication summarizing ACAP experience.
"Assessment and Recommendations for Improving the Performance of Waste Containment Systems," U.S. EPA, EPA/600/R-02/099, 2002.	Discusses issues related to the design, construction and performance of waste containment systems used in landfills, surface impoundments and waste piles and in the remediation of contaminated sites.
National Research Council, National Academy of Sciences, "Research Needs in Subsurface Science," 2000.	Examines gaps in the understanding of the performance of subsurface facilities and recommends research needs in the area.

Table 3. Summary of Selected Reports Related to Engineered Barriers (reproduced from USNRC 2003a) (contd)

Report	Brief Summary
Dwyer, Stephen F., "Water Balance Measurements and Computer Simulations of Landfill Covers," PhD Dissertation, University of New Mexico, 2003.	Provides a comprehensive summary of data collection, analysis, and computer simulations associated with DOE's ALCD program. Also includes a summary of measurements of infiltration at various sites with engineered covers.
O'Donnell, E., R. Ridky, and R. Schulz. "Control of water infiltration into near-surface, low-level waste-disposal units in humid regions," *In-situ* Remediation: Scientific Basis for Current and Future Technologies, G. Gee and N.R. Wing eds., Battelle Press, Columbus, OH, 295-324, 1994.	Summary of NRC sponsored research at USDA, Beltsville, MD, on engineered covers for low-level waste facilities.
Interstate Technology & Regulatory Council, "Technical and Regulatory Guidance for Design, Installation, and Monitoring of Alternative Final Landfill Covers," Washington, DC, 2003.	Guidance document primarily written for decision makers associated with the plan development, review, and implementation of alternative covers. Focuses on the decisions and facilitating the decision processes related to the design, evaluation, construction, and post-closure care associated with alternative covers.
Interstate Technology & Regulatory Council, "Permeable Reactive Barriers: Lessons Learned/New Directions," Washington, DC, 2005.	Summary of current understanding and experience with permeable reactive barriers, including numerous case studies.
National Research Council, National Academy of Sciences, "Long-Term Institutional Management of U.S. DOE Legacy Waste Sites," 2000.	Discusses long-term management of DOE waste sites and identifies characteristics and design criteria for effective long-term institutional management.

leave concrete with surficial or volumetric contamination onsite after license termination, the concrete should be evaluated as part of the licensee's decommissioning plan according to the LTR.

There are a number of commercial power reactors that have been permanently shut down. Table 4 lists those nuclear power plants that have both completed the decommissioning process and have had their operating licenses terminated under the LTR[19].
Table 5 provides an overview on the status (as of January 2008) of nuclear power reactors that are in the process of undergoing decommissioning (USNRC 2008).

Because of the nature of the decommissioning process applied to nuclear power reactors, it can be safely assumed that cementitious barriers (including the disposition of contaminated concrete) are considered in each case. An example is the decommissioning of the Big Rock Point plant near Charlevoix, MI. In 1965, this plant began producing electricity and became the fifth commercial nuclear power plant in the U.S. (Tompkins 2006). By April 2006, this plant

[19] This information is available at http://www.nrc.gov/reading-rm/doc-collections/fact-sheets/decommissioning.html (accessed March 18, 2009).

Table 4. Nuclear Power Plants That Have Completed the Decommissioning Process With Their Operating Licenses Terminated (http://www.nrc.gov/reading-rm/doc-collections/fact-sheets/decommissioning.html)

Reactor	Type*	Thermal Power	Location	Shutdown	Status**	Fuel Onsite
Big Rock Point	BWR	67 MW	Charlevoix, MI	8/97	ISFSI Only	Yes
CVTR	Pressure Tube, Heavy Water	65 MW	Parr, SC	1/67	License Terminated	No
Fort St. Vrain Nuclear Generating Station	HTGR	842 MW	Platteville, CO	8/18/89	License Terminated	Yes
Haddam Neck - Connecticut Yankee	PWR	1825 MW	Haddam Neck, CT	7/22/96	ISFSI Only	Yes
Maine Yankee Atomic Power Station	PWR	2772 MW	Bath, ME	12/96	ISFSI Only	Yes
Pathfinder	Superheat BWR	190 MW	Sioux Falls, SD	9/16/67	DECON NRC Part 30	No
Saxton	PWR	28 MW	Saxton, PA	5/72	License Terminated	No
Shoreham	BWR	2436 MW	Suffolk Co., NY	6/28/89	License Terminated	No
Trojan	PWR	3411 MW	Portland, OR	11/9/92	ISFSI Only	Yes
Yankee Rowe Nuclear Station	PWR	600 MW	Franklin Co., MA	10/1/91	ISFSI Only	Yes

*BWR – boiling water reactor; HTGR – high-temperature gas reactor; PWR – pressurized water reactor

**An independent spent fuel storage installation (ISFSI) is a stand-alone facility constructed for the interim storage of spent nuclear fuel. Under DECON (immediate dismantlement), portions of the facility containing radioactive contaminants are removed or decontaminated to a level that permits release of the property and termination of the USNRC license.

had undergone the complete process to shutdown, to decommissioning, and finally to site restoration. The reactor vessel was removed whole, grouted, and disposed at the Chem-Nuclear Systems, L.L.C., Barnwell, S.C. low-level radioactive waste disposal facility. The concrete reactor cavity was cut into pieces. The interior surfaces of the concrete structures were removed, assessed, and sorted for disposal and then the outer shell of the containment sphere was dismantled and the building's walls removed. More than 53 million pounds of low-level radioactive waste were shipped to disposal facilities in South Carolina, Tennessee, and Utah, and more than 1,000 shipments totaling more than 59 million pounds of nonradioactive building materials were surveyed, packaged, and shipped to an industrial landfill (Tompkins 2006).

For the decommissioning steps involving contaminated concrete at Big Rock Point or any other reactor, the uncertainties in the assaying techniques must be taken into account. The Big Rock Point reactor vessel was grouted prior to disposal, which required modeling and the uncertainties associated with the grouting process to be managed. This process can thus be seen

Table 5. Power Reactor Sites Undergoing Decommissioning as of January 2008 (Compiled from (USNRC 2008) http://www.nrc.gov/info-finder/decommissioning/power-reactor/)

	Reactor	Location	PSDAR* Submitted	LTP Submitted	LTP Approved	Decomm. Completion
1	Dresden – Unit 1	Dresden, IL	6/98	TBD**	TBD	2036
2	Fermi – Unit 1	Newport, MI	4/98	2009	2010	2012
3	Humboldt Bay	Eureka, CA	2/98	2009	2010	2012
4	Indian Point – Unit 1	Buchanan, NY	1/96	2020	2022	2026
5	La Crosse	La Crosse, WI	5/91	TBD	TBD	2020
6	Millstone – Unit 1	Waterford, CT	6/99	TBD	TBD	TBD
7	Nuclear Ship Savannah	Baltimore, MD	TBD	2014	TBD	2018
8	Peach Bottom – Unit 1	Delta, PA	6/98	TBD	TBD	2034
9	Rancho Seco	Sacramento, CA	12/94	2006	2007	2009
10	San Onofre – Unit 1	San Clemente, CA	12/98	2025	2027	2030
11	Three Mile Island – Unit 2	Harrisburg, PA	2/79	TBD	TBD	2014
12	Vallecitos Boiling Water Reactor (VBWR)	Pleasanton, CA	7/66	TBD	TBD	2021
13 & 14	Zion – Units 1 & 2	Waukegan, IL	2/00	TBD	TBD	2018

*Post-Shutdown Decommissioning Activities Report
**TBD – to be determined

as an excellent example of how cementitious materials are evaluated for dispositioning during the reactor decommissioning process.

4.0 PERFORMANCE ASSESSMENT APPROACHES FOR SENSITIVITY AND UNCERTAINTY ANALYSIS

Sensitivity and uncertainty analysis is an area of active growth for near-surface waste management activities. The approaches being used include deterministic, probabilistic, and combinations of the two. There are also variations in the implementation of the different approaches. USDOE-EM has recognized the rapid growth in the use of sensitivity and uncertainty analysis approaches for PAs and has sponsored technical exchanges to better share information and foster improved consistency moving forward (Seitz et al. 2008). A critical need that has become apparent within the USDOE is the need for better communication between people conducting PA and PA-like modeling in support of decisions in the different regulatory environments described in Section 2.

One goal of the examples in Section 3 is to illustrate how modeling has been implemented in the different environments to illustrate differences in how the modeling is being done. Examples from several sites that encompass deterministic, probabilistic and combined (hybrid) approaches to illustrate the breadth of types of analyses that are conducted.

4.1 Nevada Test Site

Sensitivity and uncertainty analyses have been conducted as part of PAs for DOE LLW disposal facilities for many years. The approaches have evolved over time from purely deterministic to more routine use of probabilistic approaches either alone or in conjunction with deterministic assessments. The approaches used within the DOE system are beginning to show more similarities as a result of efforts to share information. However, there are still differences and preferences for specific technical and implementation approaches.

4.1.1 NTS Area 5 PA

The Nevada Test Site (NTS) completed a performance assessment for the Area 5 Radioactive Waste Management Site (RWMS) disposal facility (USDOE 2006). The NTS was the first DOE Site to adopt a fully probabilistic approach to a PA. Due to the local conditions, there is no groundwater pathway evaluated as part of the performance assessment. This eliminates challenges associated with probabilistic flow modeling and thereby provided the opportunity for a detailed evaluation of other pathways.

4.1.1.1 Modeling Approach

All PA models for the NTS PA are integrated within the GoldSim® modeling platform, a fully probabilistic modeling environment developed originally for PA modeling. Native GoldSim® capabilities include Monte Carlo simulation, simulation of discrete events, and contaminant transport modules with radioactive decay and ingrowth capabilities. Integration of all models allows uncertainty and sensitivity analysis of the total system model.

The Area 5 RWMS is modeled as four one-dimensional (1-D) virtual disposal units corresponding to groups of actual disposal units with similar depths of burial. Virtual disposal units and their covers are divided into a series of mixing cells. The rate of change of radionuclide mass within each cell is described by a 1-D mass balance expression accounting for radioactive decay and mass transfer processes. In the graphical GoldSim® environment, these mass balance equations are represented as a series of cells connected by links that represent each transport process.

Since groundwater is not considered, there was no need for abstraction or upscaling from a complex deep groundwater model to a simplified model. There were detailed investigations conducted in support of the processes considered, but the underlying models were relatively straightforward. For example, the upward movement of water in the vadose zone was simplified as a one-dimensional vertical flux rate.

4.1.1.2 Parameter Assumptions and Distributions

The uncertainty analysis approach implemented for the RWMS PA involved a rigorous consideration of input parameter distributions as well as development of probability density functions for specific assumptions such as the probability of drilling into the waste and the length of the institutional control period. Since groundwater was not considered, the efforts on parameters were focused on surface pathways for exposure including upward migration via advection and diffusion in vadose zone pore water, effects of flora and fauna and gas phase migration.

Input distributions were developed for many of the inputs for the GoldSim® model. They are too numerous to identify here. Examples are provided in this section to illustrate the approaches used to develop distributions. In general, the philosophy was to develop distributions for parameters that are important in terms of the conclusions of the analysis and also moving from a conservative bias towards a more realistic representation of the expected range of conditions.

Consideration of inadvertent intrusion was a critical input for the RWMS PA. Thus, an expert panel was convened to assess the probability of inadvertent intrusion and also to assess the probability of a loss of institutional memory. The panel determined that each of these inputs should be represented with log-normal distributions. The distribution for intrusion was an estimated median of 245 years, mean of 400 years, and standard deviation of 500 years. Site knowledge was assumed to have a median of 100 years, a mean of 140 years, and a standard deviation of 140 years.

The inventory estimates were assumed to be governed by a lognormal distribution. Geometric means and standard deviations were developed for each radionuclide.

4.1.1.3 Sensitivity and Uncertainty Analysis Approach

The analysis approach included a combination of deterministic and probabilistic simulations. The GoldSim® model was run in deterministic mode during model development for inter-comparisons and benchmarking of models as new versions were developed. Probabilistic models were also run for the different iterations. The final results were presented in a probabilistic manner.

Latin Hypercube sampling and Monte Carlo simulations were the technical approaches used for the uncertainty analysis with a focus on results for the mean and 95th percentile. Up to 8,000 realizations of the model were used to gain reasonable convergence for these two results.

Sensitivity analysis was used in an iterative manner throughout the PA process to help prioritize areas for refinement in the evolving GoldSim® model and to prioritize parameters for which distributions were needed. A mixture of probabilistic and deterministic sensitivity analyses were conducted to provide feedback regarding the influence of individual perturbations as well as feedback on global sensitivities.

Specifically, detailed sensitivity analyses were conducted for the resident farmer scenario using a gradient-boosting regression algorithm to estimate sensitivity indices. The GoldSim® model was refined based on sensitivity results used to identify inputs having a significant effect on the conclusions and thus could benefit from further study. Plant/soil concentration ratio for Tc was identified as an important parameter. Burrow shape parameters were important for the air pathway. Radon flux at the surface was highly dependent on the assumed emanation coefficient. Variability in inventory was not shown to be as significant. Partial dependence plots and sensitivity indices were used to illustrate the importance of different parameters.

4.2 Idaho Site

4.2.1 Waste Management Complex PA

The active disposal facility at the Radioactive Waste Management Complex (RWMC) at the Idaho National Laboratory is operated in accordance with DOE Order 435.1 (DOE NE-ID 2007). This facility is located within the historic waste burial grounds and thus the inventories are also included in the CERCLA assessment described in Section 3.2.2. The PA for the active disposal facility in the RWMC was conducted using a hybrid approach with the compliance case and several sensitivity cases being run in a deterministic manner and a probabilistic approach being used for the detailed sensitivity and uncertainty analysis. This section includes a brief summary of the approach adopted for the sensitivity and uncertainty analysis.

4.2.1.1 Modeling Approach

Two parallel modeling efforts were involved for the RWMC PA. Process-specific models and experiments were conducted to capture details for behavior of key elements of the system (e.g., corrosion, and geochemistry). These specifics were implemented into a detailed source term model (DUST-MS®) and a three-dimensional groundwater model (TETRAD). The linkages of the different models are illustrated in Figure 2.

In parallel, an abstracted representation of the near-field and vadose zone system was developed in Mixing Cell Model (MCM) (Rood 2005) and the aquifer was modeled using GWSCREEN (Rood 2003). The MCM model directly used the source term results from DUST-MS®. The TETRAD and MCM/GWSCREEN models were benchmarked to develop good agreement in the projected results. Benchmarking was conducted for multiple radionuclides to build confidence of the ability of the MCM/GWSCREEN model to adequately represent the results from TETRAD.

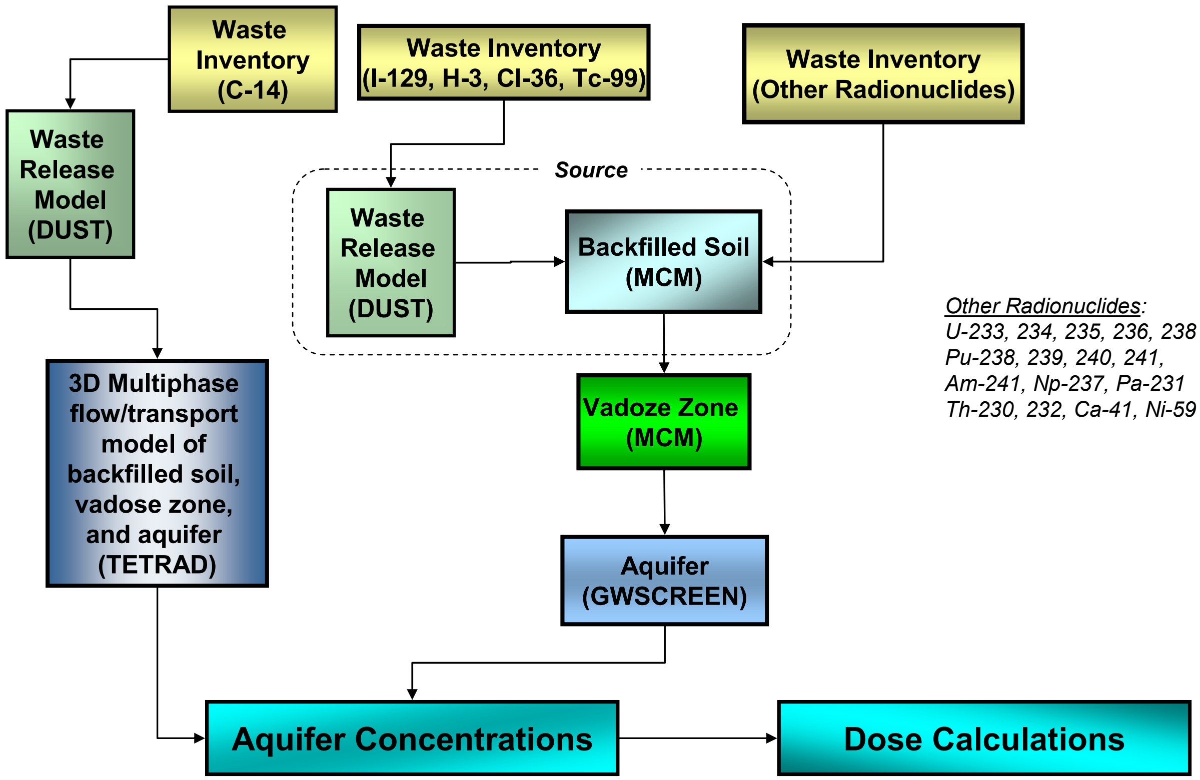

Figure 2. Idaho RWMC Modeling Approach (DOE NE-ID 2007)

4.2.1.2 Parameter Assumptions and Distributions

Input distributions were established for fifteen primary parameters in the model, including inventory, cover longevity, infiltration, aquifer velocity and dispersivity, and geochemistry. The distributions were intended to represent a reasonable range of conditions based on field, experimental and/or literature information. A few examples of distributions for these parameters are provided in this section. Table 6 is a list of all of the distributions used.

Engineered cover longevity was identified as a parameter of interest and expert judgment was used as a basis for developing a distribution of potential failure times. In the initial draft of the PA, a range of 100 to 100,000 years was used with a log-uniform distribution. Based on review comments from the Low-Level Waste Disposal Facility Federal Review Group (LFRG) the range for this distribution was reduced to 100 – 1,000 years, still with a log-normal distribution. Failure implied that the average infiltration rate for the facility returned to background levels.

Distribution coefficients for several key elements (e.g., U, Th, Ra, Ac, and Pa) were assigned log-normal distributions based on site-specific studies and general literature reviews. In the case of uranium, a truncated log normal distribution was used to allow values close to zero without actually using zero and to limit the upper value to 152 ml/g. Uniform distributions were also developed for the solubility of three uranium isotopes (234, 235, and 238) with the deterministic value used as a minimum and five times the deterministic value used as the maximum. The intent was to explore the impacts of worse than expected solubilities without taking credit for lower solubilities

Table 6. Input Distributions for RWMC Uncertainty Analysis (DOE NE-ID 2007)

Parameter	Distribution	Comments/Reference
Inventory scaling factor	Uniform: minimum 0.5, maximum 2	Assumed to be ± a factor of 2 between deterministic value and upper-bound estimate.[b]
Engineered cover longevity (year)	Log-uniform: minimum 100 years, maximum 1,000 years	Assumed. The minimum is equal to the start of institutional control (2010). The maximum was selected to include the time of maximum dose
Cap infiltration rate (m/year)	Triangular: minimum 0.0005; mode 0.001, maximum 0.002	Assumed to be ± a factor of 2 from the deterministic value.
Background infiltration rate through vadose zone (m/year)	Triangular: minimum 0.005, mode 0.01, maximum 0.02	Assumed to be ± a factor of 2 from the deterministic value.
Infiltration rate through source before cap placement (1984–2010) (m/year)	Triangular: minimum 0.02, mode 0.05, maximum 0.10	Assumed based on variability of infiltration rates across SDA as given in the RI/FS (Magnuson and Sondrup 2006).
Longitudinal dispersivity in aquifer (m)[a]	Triangular: minimum 10, mode 20, maximum 40	Assumed to be ± a factor of 2 from the deterministic value, same as 2000 PA (Case et al. 2000).
Darcy velocity in aquifer (m/year)	Triangular: minimum 0.37, mode 0.75, maximum 1.5	Same as 2000 PA (Case et al. 2000).
Uranium K_d (mL/g)[c] (Parent)	Truncated Lognormal: GM 15.4, GSD 5, maximum 152, minimum 0.001	GM value is the deterministic value, GSD is based on Sheppard and Thibault (1990).
Thorium K_d (mL/g)[c] (Progeny)	Lognormal: GM 500, GSD 1.9	GM value is the deterministic value, GSD is based on Sheppard and Thibault (1990).
Radium K_d (mL/g)[c] (Progeny)	Lognormal: GM 575, GSD 6.3	GM value is the deterministic value, GSD is based on Sheppard and Thibault (1990).
Actinium K_d (mL/g)[c] (Progeny)	Lognormal: GM 225, GSD 1.9	GM value is the deterministic value, GSD not available in Sheppard and Thibault (1990) so assumed same GSD as uranium.
Protactinium K_d (mL g)[c] (Progeny)	Lognormal: GM 8, GSD 1.9	GM value is the deterministic value, GSD not available in Sheppard and Thibault (1990) so assumed same GSD as uranium.
U-234 solubility (mg/m^3)	Uniform: minimum 0.02, maximum 0.102	Minimum is the deterministic value, maximum assumed to be 5× deterministic value.
U-235 solubility (mg/m^3)	Uniform: minimum 4.8, maximum 24	Minimum is the deterministic value, maximum assumed to be 5× deterministic value.
U-238 solubility (mg/m^3)	Uniform: minimum 907, maximum 4,435	Minimum is the deterministic value, maximum assumed to be 5× deterministic value.

a. The transverse and vertical dispersivity were correlated to the longitudinal dispersivity. The transverse dispersivity was 0.25 × the longitudinal dispersivity. The vertical dispersivity was 0.085 × the longitudinal dispersivity. The factors for transverse and vertical dispersivity were based on the deterministic ratio of the transverse or vertical dispersivity to the longitudinal dispersivity.

b. After sampling, the scaling factor was multiplied by the radionuclide-specific release rate for H-3, Cl-36, and Tc-99, or the deterministic radionuclide inventory for the uranium isotopes. All scaling factors were sampled independently.

c. Partition coefficients for the source (alluvium) and interbeds. Unsaturated zone K_d values were assumed to be zero and aquifer K_d values were 1/25th the alluvium/interbed K_d values.

GSD = geometric standard deviation.

GM = geometric mean

than expected. These distributions are expected to have a conservative bias.

4.2.1.3 Sensitivity and Uncertainty Analysis Approach

A hybrid approach using a parallel combination of detailed deterministic analyses and less detailed probabilistic analyses was used to provide broad perspective regarding important aspects of system behavior. The sensitivity and uncertainty analysis was also conducted using a combination of deterministic and probabilistic calculations. A few focused parametric sensitivity analyses were conducted along with a probabilistic uncertainty and sensitivity analysis that included input distributions for many parameters.

Single parameter sensitivity analyses were conducted to illustrate the effect of changes in individual parameters on the expected dose. The single parameter cases were focused on addressing specific questions asked during reviews. Given the significance of tritium concentrations at 100 m downstream during the time of institutional control, there were concerns that allowing the tritium to be released early could be reducing the concentrations at 100 m after loss of institutional control. Sensitivity of the projected mass flux of tritium to delays in release times was explored to address this question. Six delay times from 10 to 76 years were considered and the resulting dose was shown to decrease as the delay times increased.

A second sensitivity analysis was conducted to address a change in the average infiltration rate through the intact engineered cover. Although performance of the cover is actually expected to be better than the 0.1 cm/yr assumed in the base case, there was a desire to include a case with an infiltration rate of 1 cm/yr to illustrate the impacts of a significant increase. The results for the increased average infiltration rate were roughly a factor of 2 larger than the base case, but remained well below the performance objective of 25 mrem/yr.

A full Monte Carlo simulation with random sampling was also conducted using distributions for 15 input parameters in the model. The probabilistic analysis was conducted using a Perl script as the Monte Carlo driver for 500 MCM/GWSCREEN realizations. Results from the Monte Carlo simulations were provided for a range from the 5th to 95th percentile (see Fig. 3). In Figure 3, the 50th percentile curve as well as the base case and 10x infiltration case were plotted over the range of results to illustrate both the probabilistic uncertainty analysis as well as some perspective from a deterministic sensitivity case. All of the results were well below the performance objective.

A sensitivity analysis was conducted using the Monte Carlo results using a regression technique in the Crystal Ball software package (Decisioneering Inc. 2000). Rank correlation coefficients were generated for the parameters of interest and then the percentage contribution of each parameter to the total variance was estimated. These statistics were calculated for four different times (end of institutional control and 500, 1,000 and 2,000 years after disposal). The times were selected based on the timing of peaks in the analysis results.

The key parameters based on maximum percent variance at each time fit well with the peaks that were observed. For example, the tritium inventory/release assumptions were most important at the early times, the Cl-36 assumptions and cover longevity were most important at 500 years, and Cl-36 assumptions were most important at 1,000 years, and uranium K_d was most important at 2,000 years.

4.3 Savannah River Site

4.3.1 F-Tank Farm PA

The F-Tank Farm is being closed under the Ronald Reagan National Defense Authorization Act (NDAA) for Fiscal Year 2005 - Section 3116 in order to manage the residual materials that will remain in the tanks and ancillary equipment as LLW. A PA was conducted

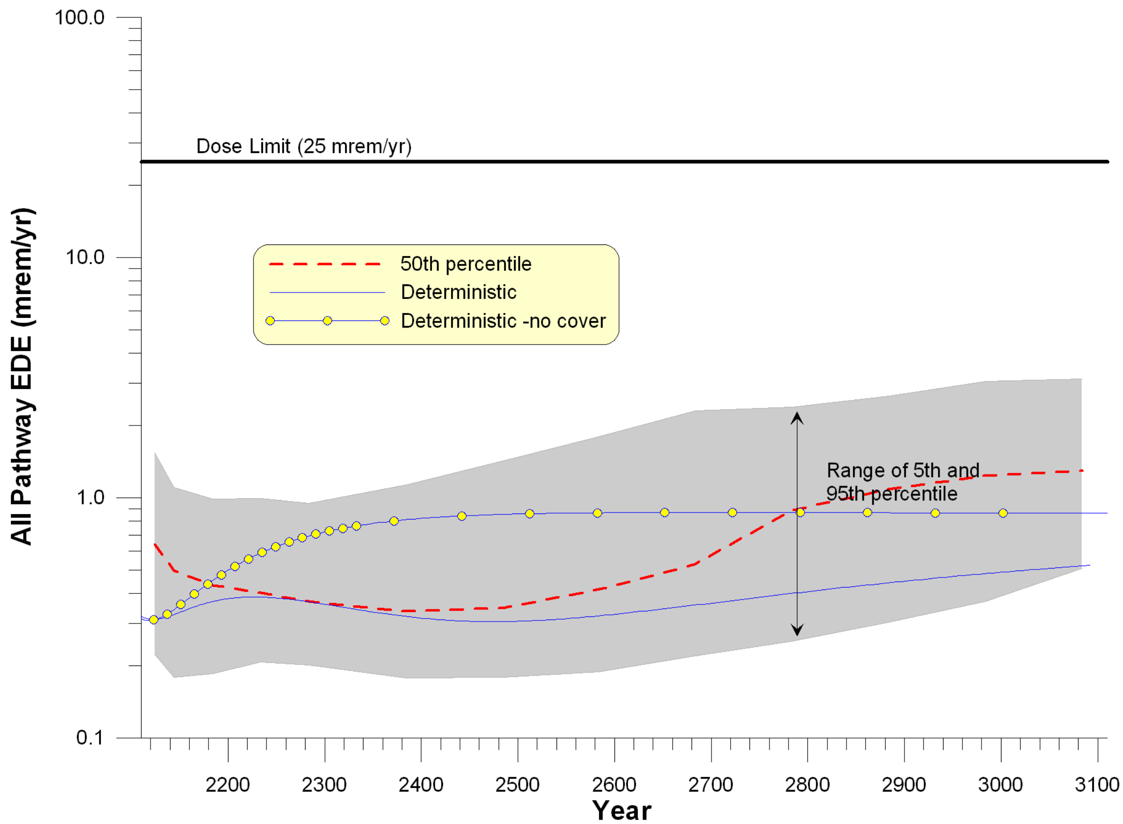

Figure 3. Uncertainty Analysis Results from the RWMC PA (DOE-NE/ID 2007)

to demonstrate that the waste that remains can meet the performance objectives in 10 CFR Part 61. The PA for the F-Tank Farm was conducted using a hybrid approach with the compliance case and several sensitivity cases being run in a deterministic manner and a probabilistic approach being used for the detailed sensitivity and uncertainty analysis (SRS 2008). This section includes a brief summary of the approach adopted for the sensitivity and uncertainty analysis.

4.3.1.1 Modeling Approach

Two parallel modeling efforts were involved for the F-Tank Farm PA (see Figure 4). Process-specific models and experiments were conducted to capture details for behavior of key elements of the system (e.g., concrete degradation, corrosion, and geochemistry). These specifics were implemented in a two-dimensional cover model (HELP®) and near-field and vadose zone fate and transport model (PORFLOW®) for several different failure scenarios. Detailed data were also developed to support each of those models.

In parallel, an abstracted one-dimensional representation of the near-field and vadose zone system was developed in the GoldSim® modeling platform. The two models were benchmarked in an iterative manner with improvements made to both models as a result of the intercomparisons. Benchmarking was conducted for multiple radionuclides and failure scenarios in order to assess the comparison for different sets of conditions.

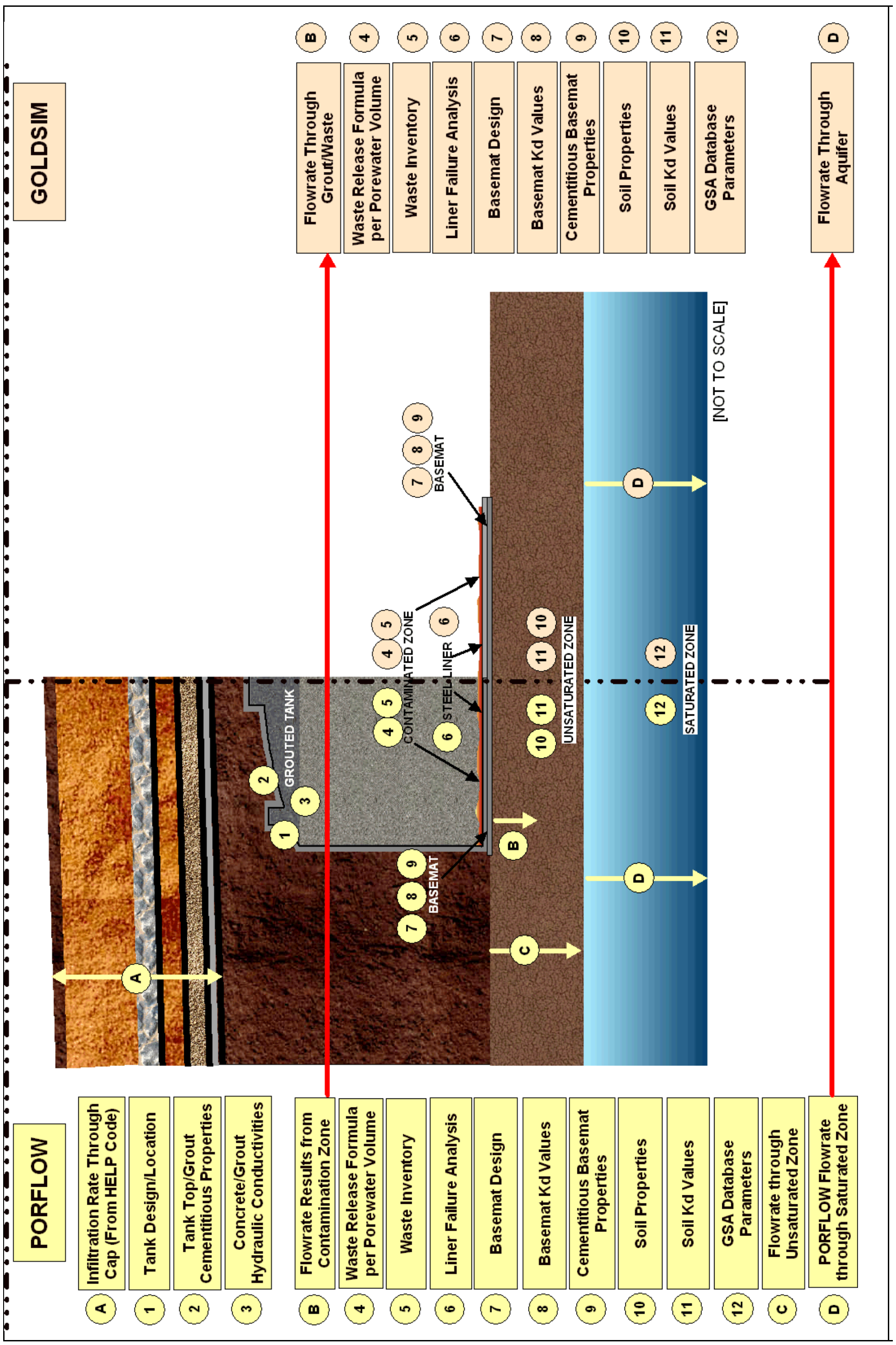

Figure 4. Groundwater Pathway Models for the SRS F-Tank Farm (SRS 2008)

4.3.1.2 Parameter Assumptions and Distributions

Input distributions were established for many different parameters in the models, including contaminant inventories, physical properties of barriers and the natural environment, geohydrology, geochemistry, and exposure assumptions. The distributions were developed based on experimental work, detailed modeling and literature searches. A few examples of distributions for engineered features, hydrological and geochemical parameters are provided in this section.

The thickness of the basemat at the bottom of each type of tank was represented with a triangular distribution using minimum and maximum thicknesses as the bounds reflecting engineering tolerances and design information. The most likely value for the thickness was calculated based on a weighted median of the design parameters and was assigned as the peak of the distribution. This is an example of a distribution based on actual design data.

Distribution coefficients were primarily used to represent processes that would limit the mobility of radionuclides in the material of interest. Distributions for K_ds were developed for key radionuclides on an element-specific basis. K_ds are assumed to be log-normally distributed, but the distributions were treated differently if the mean K_d was greater than 1000 ml/g or less than 1000 ml/g. The lower and upper bounds for the log-normal distribution are obtained using a multiplier of 3.3 for K_ds greater than 1000 ml/g and 1.9 for K_ds less than 1000 ml/g. For example, the initial K_d for Tc in oxidizing cementitious media was assumed to be 0.8 ml/g. Thus, the upper bound would be 1.52 ml/g and the lower bound would be 0.42 ml/g. In reducing cementitious media, the K_d for Tc is assumed to be 5,000 ml/g. Thus, the upper bound would be 16,500 ml/g and the lower bound would be 1,515 ml/g.

The thickness and Darcy velocity for the saturated zone were also assigned distributions to reflect their influence on the amount of dilution that would occur as radionuclides migrate from the unsaturated zone into the water table. Normal distributions were used to represent these two parameters.

Probabilities (or discrete distributions) were also assigned to several parameters. For example, probabilities for the different failure scenarios for each of the different types of tanks were developed based on expert judgment and probabilities were assigned to different inventory multipliers to reflect uncertainty about the actual inventory as well as uncertainty regarding how much inventory would be removed. A distribution was also developed to identify the aquifer from which a resident would obtain water based on information obtained regarding current drilling practices.

4.3.1.3 Sensitivity and Uncertainty Analysis Approach

A hybrid approach using a parallel combination of detailed deterministic analyses and less detailed probabilistic analyses was used to provide a broad perspective regarding important aspects of system behavior. The sensitivity and uncertainty analysis was also conducted using a combination of deterministic and probabilistic calculations. Numerous focused parametric sensitivity analyses were conducted along with a probabilistic uncertainty and sensitivity analysis that included input distributions for many parameters. The analyses considered multiple receptor locations to identify the point or points of maximum dose.

Single parameter sensitivity analyses were run with the PORFLOW® model to explore the effects of changes on the model output. For example, sensitivities to changes in inventories, assumed K_ds for Tc-99 and Pu-239, and specific aspects of the failure scenarios were investigated individually. Process-specific sensitivity analyses were also conducted for specific input parameters, such as the failure time for the carbon steel liner. These simulations were conducted to address “what-if” type questions individually.

The sensitivity analysis for inventories highlighted the importance of solubility assumptions for Pu-239, Pu-240, Tc-99, and U-238 in selected tanks, which showed that there was no increase in dose for increased concentrations of any radionuclide that is present at or above its solubility limit in a given tank. This can impact decisions regarding the benefit of additional cleaning of a tank.

Single parameter sensitivity analyses were also conducted to assess the impact of assumptions regarding Pu and Tc K_d values assumed for materials beneath the waste. Tc fluxes were shown to be relatively insensitive to changes in K_d, but Pu fluxes were shown to be sensitive to changes. Additional sensitivity cases were conducted to explore changes in assumptions regarding failure scenarios but are not discussed here.

A full Monte Carlo simulation with Latin Hypercube Sampling was also conducted using distributions for many of the input parameters in the model. The probabilistic analysis was conducted using the GoldSim® modeling platform and involved 1,000 realizations. The sensitivity analysis involved 5,000 realizations. Summary statistics (mean, median and a few percentiles) for doses and concentrations for key radionuclides and well locations over different time frames were compared for the 1,000 realization and 5,000 realization cases. All of the summary statistics showed good agreement for the different number of realizations, which provided confidence that 1,000 realizations were sufficient for the uncertainty analysis. Results for a 10,000-year compliance period are presented in Figure 5. The mean and median results in Figure 5 are all below the 25 mrem/yr dose standard and the 95th percentile dose was slightly above the standard.

A gradient boosting method model was fitted to the GoldSim® results and variance decomposition was

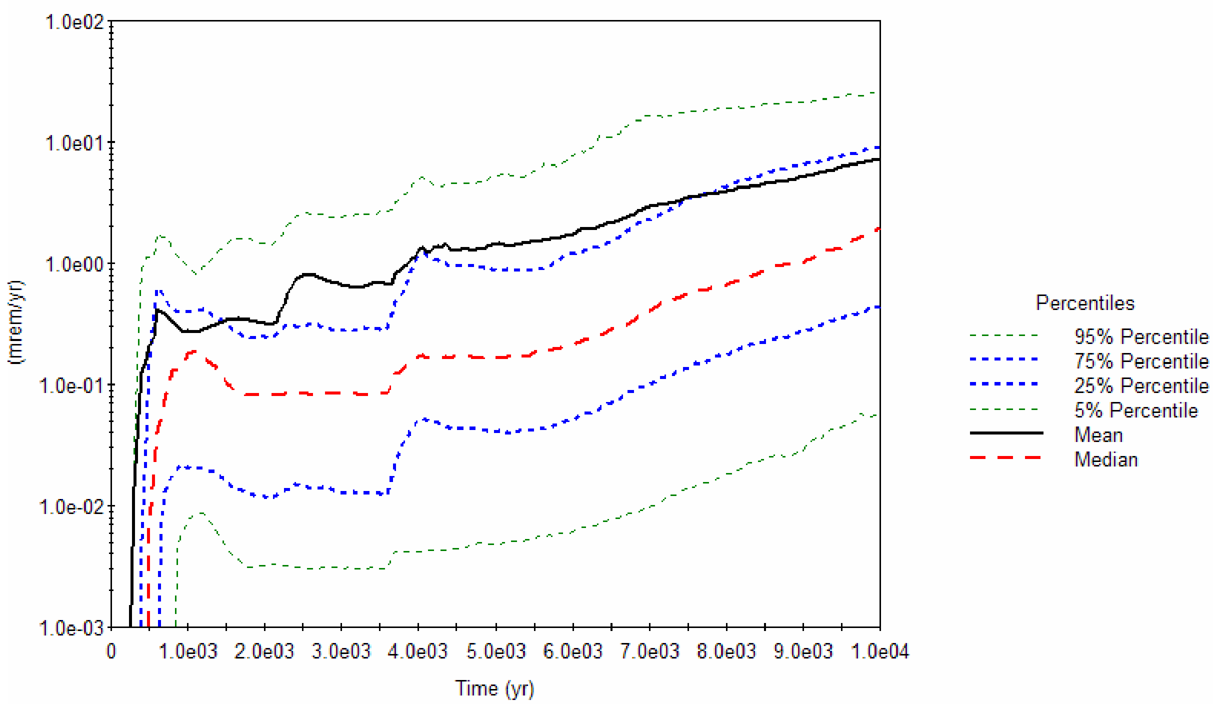

Figure 5. SRS F-Tank Farm PA Maximum Exposure Results for the 10,000 Year Compliance Period (SRS 2008)

used to calculate sensitivity indices for parameters of interest. Sensitivity indices were calculated for the doses at wells yielding the largest doses and for the inadvertent intruder scenarios. Indices were also calculated for key radionuclide concentrations at the wells yielding the largest doses.

Examples of the sensitivity indices for the results at Wells 6 and 33 for the 10,000 year simulation are shown in Table 7. The sensitivity indices are relatively small and distributed among several parameters, which illustrates that a single parameter does not have an overwhelming influence on the results. However, the results show that the assumed K_d for Pu in sandy soil is important for the doses at Well B, which is linked to significant Pu inventories in an upstream tank and the assumed failure scenario for Tank 34 is important for the results at Well 33, which is downstream of that tank. When global sensitivity was considered, the saturated aquifer thickness was the most sensitive parameter.

4.4 Hanford Site

4.4.1 Integrated Disposal Facility PA

The Hanford Site completed the sixth iteration of the performance assessment for the Integrated Disposal Facility (IDF) in 2005 (Mann et al. 2005). The IDF PA is based fully on the use of deterministic models. A variety of different scenarios and parameter sensitivity studies were conducted in a deterministic manner to address uncertainty analysis needs. The IDF PA involves a combination of several detailed modeling approaches focused on specific aspects of the problem and substantial efforts to better understand the processes critical to performance.

4.4.1.1 Modeling Approach

The IDF PA involved the integration of results from several detailed experimental and modeling efforts including: waste form release, infiltration through a cover, vadose zone flow and transport, groundwater flow and transport, and dose. Base case analyses representing different waste management strategies (i.e., glass, bulk vitrification, and advanced grout) were run and supplemented by numerous targeted sensitivity cases to illustrate the relative influence of changing assumptions on the performance of the system.

Two-dimensional modeling approaches were used for the near-field (STORM, Bacon et al. 2004) and vadose zone simulations (VAM3DF, Huyakorn and Panday 1999). The Hanford Site groundwater model (CFEST-96, Gupta et al. 1987) was used as the basis for calculating migration in the aquifer (see Figure 6). STORM is a coupled unsaturated flow, chemical reactions, and contaminant transport simulator that was used for the glass and bulk vitrification waste form releases. An analytical model was used to estimate the contaminant releases from the other waste forms in the reference case. It was also used for near field modeling in many of the sensitivity cases.

Table 7. Example Sensitivity Analysis Results for the F Tank Farm PA (SRS 2008)

First 10,000 years	Sensitivity Index		
	Well A	Well B	All Wells
Tank 34 failure scenario	11	Not significant	3.7
Vadose zone thickness	5.6	6.8	3.1
Pu K_d (sandy soil)	4.9	11	5.5
Saturated Aquifer Thickness	4.4	6.4	7.3
Pu K_d (clayey soil)	Not significant	4.9	Not significant

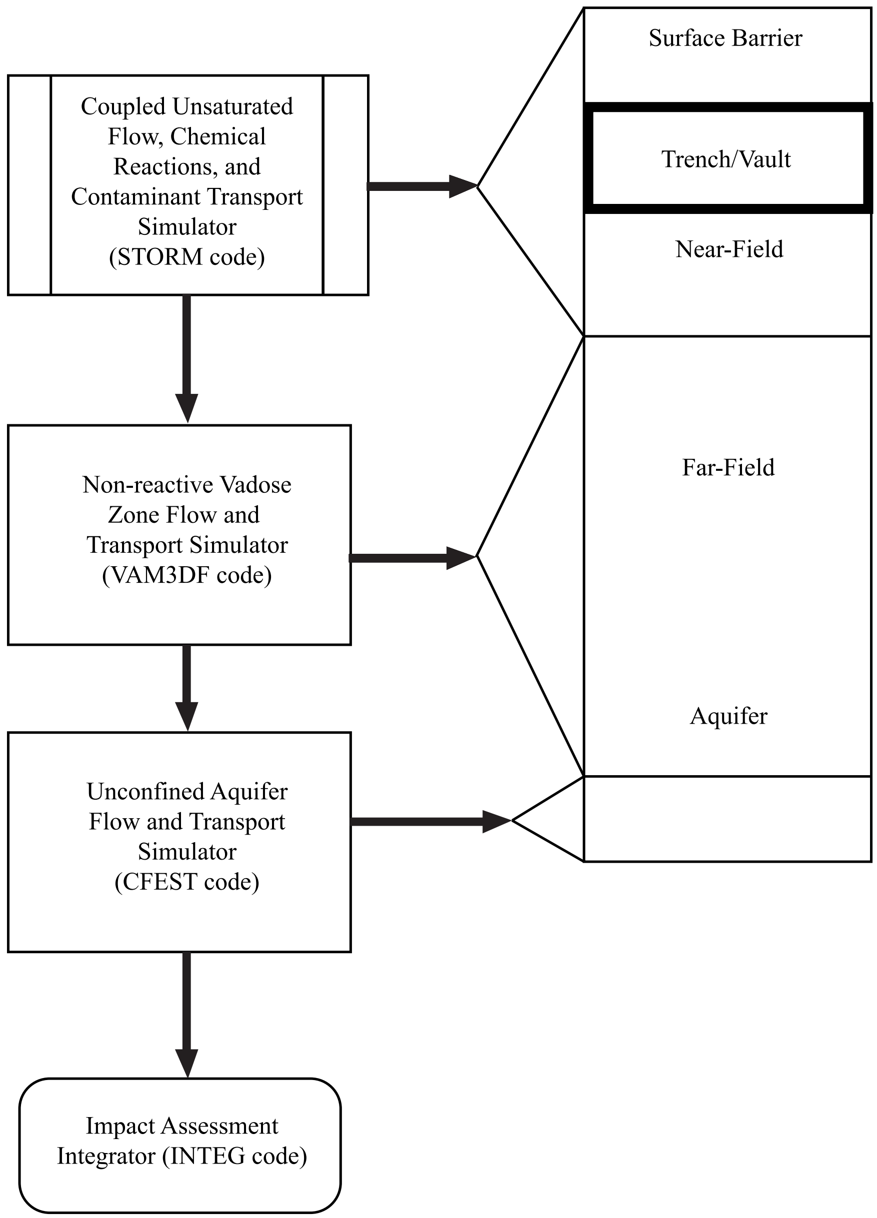

Figure 6. Hanford IDF PA Modeling Approach (Mann et al. 2005)

4.4.1.2 Parameter Assumptions and Distributions

Distributions were not developed for any parameters because all calculations were conducted in a deterministic manner. Detailed data packages were developed to document the basis for the parameter values that were selected. The intent was to develop realistic and defensible values for input parameters important to performance in the reference base cases. A few examples of parameters where ranges of values were considered are provided in this section.

The base case infiltration rate was assumed to be 1 mm/yr, but a range from 0.01 mm/yr to 50 mm/yr was considered in the sensitivity analyses. Likewise, a range of effective diffusion coefficients was assumed for different radionuclide species in cementitious waste forms. "Best" and base case K_d values in the sandy vadose zone soils were also considered for different classes of radionuclides. Many other parametric sensitivity cases were considered for specific material and geochemical properties and exposure parameters.

4.4.1.3 Sensitivity and Uncertainty Analysis Approach

Sensitivity analyses were used to assess and illustrate the role of uncertainty relative to the projected doses. Parametric sensitivity cases were run to assess the ranges of parameter values assumed for key parameters such as those identified above. In addition to parametric sensitivity cases, additional scenario-based sensitivity cases were also considered. For example, the effect of different pumping rates for water wells was evaluated. Different inventory and infiltration rate scenarios were also considered as well as different waste disposal configurations. Figure 7 shows the sensitivity results for a case evaluating the impacts of different recharge rates. The overall approach was to provide a wide range of sensitivity cases to illustrate the impacts of changes in a variety of uncertain inputs. The end result was a relatively broad look at the effects of changes in a variety of input parameters that illustrated that performance of the facility remained compliant within the expected realm of uncertainty.

5.0 PA-LIKE EXAMPLES

5.1 Idaho Site

In the previous section, examples of performance assessments (PAs) for engineered systems were described for various DOE facilities that incorporate cementitious barriers. In this section, the summary

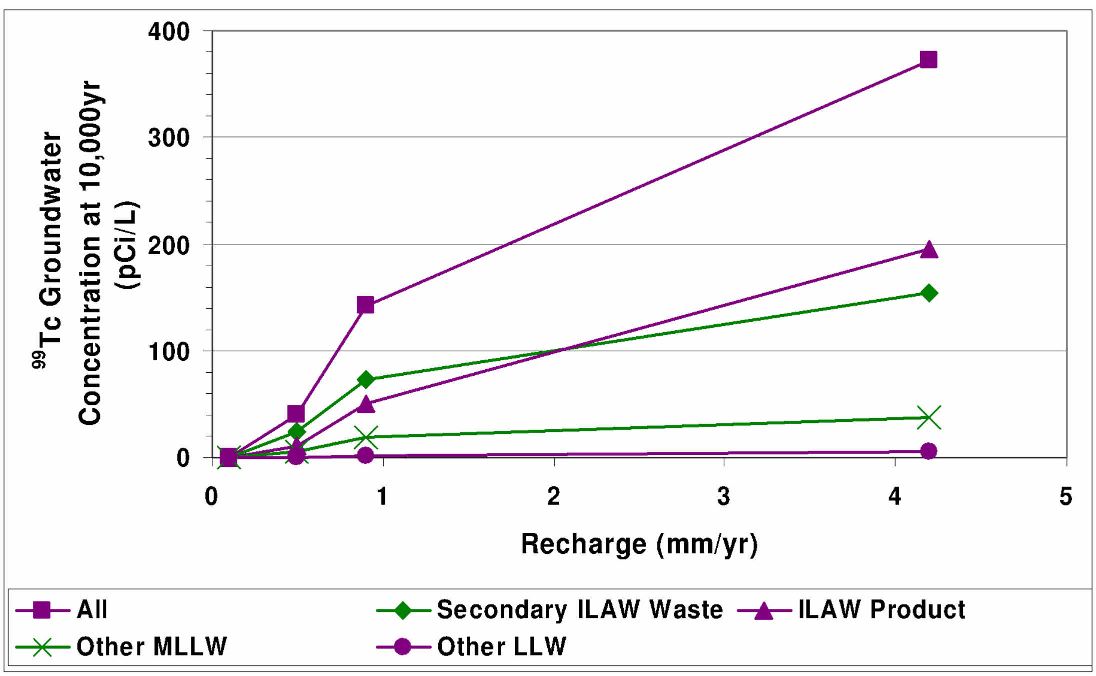

Figure 7. Sensitivity of Tc-99 Concentration in Groundwater at 10,000 Years to Changes in Recharge Rate (Mann et al. 2005)

is extended to examples of other types of risk assessments for DOE facilities including the Idaho, Hanford, and Savannah River Sites. These examples will demonstrate the similarities and differences between PA and other types of risk assessments performed to support other regulatory processes (e.g., CERCLA, RCRA, etc.).

5.1.1 Engineering Test Reactor CERCLA Non-Time Critical Removal Action

The Engineering Test Reactor (ETR) located on the Idaho National Laboratory (INL) is in the process of being decommissioned (including decontamination and dismantling) (USDOE-ID 2007). The decommissioning strategy involves removing the pressure vessel, grouting and disposal of the vessel at the INEEL CERCLA Disposal Facility (ICDF), and demolishing the reactor building (USDOE-ID 2007). These actions are consistent with the joint USDOE/USEPA policy that established the CERCLA non-time-critical removal action for decommissioning (USDOE & USEPA 1995).

On-site disposal of the ETR reactor vessel was justified using an iterative modeling approach involving multiple screening steps and a final risk assessment for contaminants of concern (McCarthy 2006; Staley 2006). The approach used to manage uncertainties in these analyses was an attempt to bound actual risks that might result using "conservative" assumptions in point-value calculations (Staley 2006). The screening phases for the groundwater pathway were ordered to be increasingly accurate though always bounding. This section provides a brief summary of the analysis of uncertainty used in the risk assessment process.

5.1.1.1 Modeling Approach

Separate assessments were performed to support ETR decommissioning. The first assessment was performed to demonstrate whether current estimates of contaminant inventories could remain in place and be protective in terms of the groundwater pathway or, alternatively, how much could remain in place (McCarthy 2006). The second assessment evaluated the protectiveness of contaminants that would remain in the surface soil for two D&D scenarios: 1) leaving the ETR vessel in-place or 2) removing and disposing the vessel offsite. Each of these phases will be described separately.

The groundwater assessment was performed in two phases: 1) radionuclide screening using the factors provided by the National Council on Radiation Protection and Measurements (NCRP) (NCRP 1996a; NCRP 1996b), and 2) radionuclide and hazardous chemical screening using a "simple and conservative" application of GWSCREEN (Rood 1994) to estimate dose, risk, or concentration[20]. The conceptual model for defining the NCRP screening factors (SF_{gw}) (for ingestion of contaminated groundwater in this case) can be represented by the following expression (McCarthy 2006; NCRP 1996a):

$$SF_{GW}\left(\frac{Sv}{Bq}\right) = A_0(Bq) \times T\left(\frac{yr}{L \cdot Bq}\right) \times U_{DW}\left(\frac{L}{yr}\right) \times DCF_{ing}\left(\frac{Sv}{Bq}\right)$$

where A_0 is the initial inventory, T is the environmental transfer factor, U_{DW} is the exposure or uptake factor, and DCF_{ing} is the dose conversion factor for drinking water.

The factors used to derive the NCRP factors incorporate fate and transport processes and an assumed exposure scenario to relate annual dose to a hypothetical receptor per unit activity (McCarthy 2006). The

[20] GWSCREEN considers dispersion and unsaturated transit time where the NCRP factors do not (McCarthy 2006). The screening application of GWSCREEN is consistent with the Track 2 approach used in the INL CERCLA process for sites with low hazard probabilities (INEL 1994).

NCRP screening factors used for screening in this study can be used to demonstrate compliance with environmental standards or other reference levels for radionuclide releases to the various environmental pathways (McCarthy 2006). Of the 52 possible radionuclides, 24 nuclides were screened out using the NCRP method (using a limit dose of less than 1×10^{-5} Sv (1 mrem)), which left 28 radionuclides for additional analysis.

The GWSCREEN code was used in the next phase of the ETR groundwater screening risk assessment. The conceptual model for GWSCEEN is illustrated in Figure 8. The application of the model for ETR radionuclide and hazardous chemical screening was intended to be conservative (i.e., produce higher than expected doses) using assumptions including (McCarthy 2006):

- For the source, radionuclides are assumed mixed homogeneously with soil in a volume represented by the volume of the ETR belowground structure.
- The receptor well is on the downgradient facility boundary.
- There are no containment structures, engineered barriers, gradual releases via corrosion, or solubility-limited releases.
- There was no dispersion in the unsaturated zone, which may or may not be "conservative."
- The aquifer was a homogeneous isotropic media of infinite lateral extent and finite thickness.

GWSCREEN was developed to evaluate INL CERCLA sites (Rood 1994) and can provide conservative estimates of groundwater concentrations and corresponding ingestion doses and risks.

In the ETR application, contaminants were screened based on predicted peak doses and risks for radionuclides and predicted peak concentrations for nonradionuclides; the remaining contaminants were denoted contaminants of concern (COCs) (McCarthy 2006). For radionuclides, COCs have either predicted peak doses greater than 4×10^{-6} Sv/yr (0.4 mrem/yr) or peak risks greater than 10^{-6}; C-14, Cl-36, H-3, Ni-59, and Pu-239 were defined as COCs (McCarthy 2006). Using a limit of one-tenth the MCL for hazardous chemical screening produced barium, beryllium, chromium, copper, manganese, and nickel as COCs.

A more detailed and site-specific evaluation of doses and risks was conducted for the five radionuclide and six chemical COCs obtained from the first two screening phases (McCarthy 2006)[21]. Many conservative assumptions are retained in this analysis; however, specific assumptions are relaxed (i.e., infiltration rates, dispersivity, and source release), to more accurately represent the ETR source release and flow and transport. Changes in these parameters and the bases for the changes will be described in the next section. However, the basic conceptual model for this more detailed evaluation is still represented by Figure 8. The more detailed evaluation indicated that C-14 was the only radionuclide predicted to have a groundwater pathway risk of greater than 1×10^{-6} and that chromium was the only hazardous chemical to have a predicted concentration greater than its MCL (McCarthy 2006).

A second set of separate dose and risk analyses were performed to evaluate the protectiveness of contaminants that would remain in the surface soil for two D&D scenarios: 1) leaving the ETR vessel in-place and 2) removing and disposing the vessel offsite (Staley 2006)[22]. Risks from residual contamination under these scenarios were evaluated using a worst-case contaminant source term and exposure scenarios listed below:

[21] For the COC screening calculations, the approach can be conceptualized as risk is the product of exposure and a risk per unit exposure factor derived for the scenario under consideration (Staley 2006).
[22] In these analyses, the groundwater pathway was not considered.

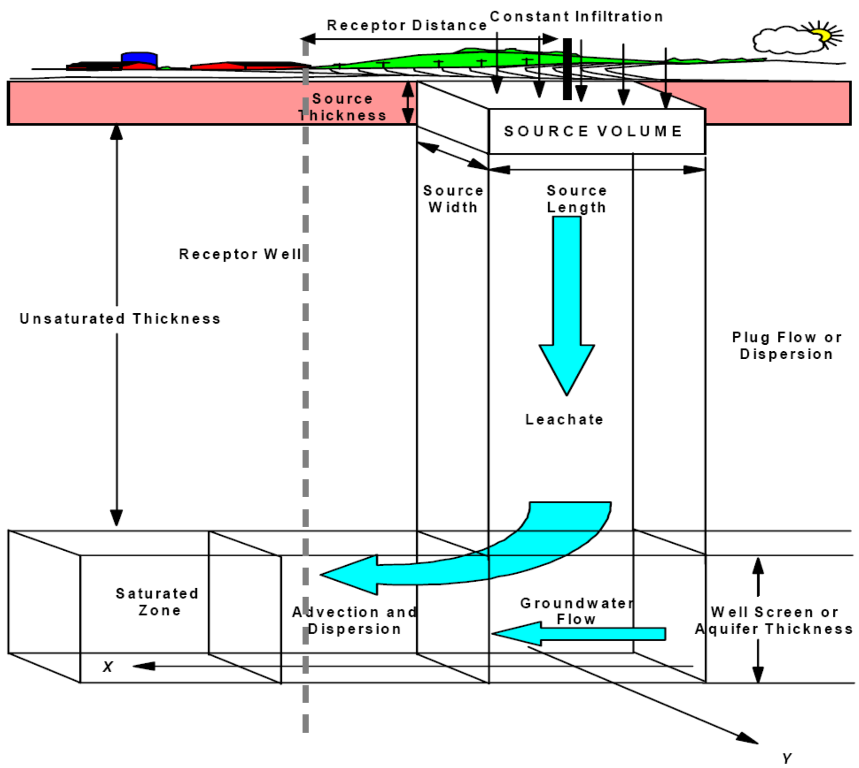

Figure 8. GWSCREEN Conceptual Groundwater Model (reproduced from McCarthy 2006)

- Any residual contamination down to 10 feet below grade is uniformly mixed in the top 3 m (10 feet) of soil and can impact an intruder 90 years from present.
- A resident will build a house on the ETR site including excavating 3 m (10 feet) of contaminated soil to build a basement and spreading the contaminated soil across the surface.
- The resident lives at the site for 30 years, including 6 years as a child, and is exposed to external radiation, ingests contaminated soil and fruits and vegetables grown on the site, and inhales fugitive dust (Staley 2006).

Standard USEPA risk assessment equations were used to estimate risks from radionuclides and hazardous chemicals in the soil (USEPA 1996; USEPA 2000). The soil concentrations were conservatively estimated from only the inventory and soil mass present and these were compared to soil screening levels for each pathway under consideration. These simple screening calculations indicated that removing the vessel (with resulting bounding risk of less than $1x10^{-6}$) would be protective; whereas, leaving the vessel in place would exceed the USEPA $1x10^{-4}$ risk limit and would require action be taken at the site. No hazardous chemicals posed unacceptable risks based on these bounding calculations.

5.1.1.2 Parameter Assumptions and Distributions

A series of screening risk analyses were performed to help inform the remedial actions needed for the ETR. For the groundwater pathway, three sets of screening calculations were performed that were intended to provide more and more accurate results as contaminants of potential concern were identified and then evaluated. An initial screening was performed using the NCRP screening factors. The more detailed analyses for the groundwater pathway used the GWSCREEN code developed to evaluate INL CERCLA sites. At least one radionuclide (i.e., C-14) was found to pose unacceptable risks via the groundwater pathway. A separate set of screening risk analyses were performed to evaluate whether or not the ETR vessel would have to be removed, and the results indicated that leaving the vessel in place would pose unacceptable risks. The important parameters in the various models are described in Table 8.

5.1.1.3 Sensitivity and Uncertainty Analysis Approach

As illustrated in Table 8, point-value dose and risk analyses were used as the bases for decision-making

Table 8. Example Exposure Parameters for ETR Screening Assessments for Radionuclides (McCarthy 2006; Staley 2006)

	Groundwater Pathway Analysis			Vessel Removal
Exposure Parameter	**NCRP-based Screening**	**Initial GWSCREEN Screening**	**Final GWSCREEN Screening (COCs)**	**USEPA Resident**
Drinking water intake	800 L/yr	2 L/yr		--
Soil ingestion rate	--	--		120 mg/d
Inhalation rate	--	--		20 m^3/d
Particulate Emission Factor		--		5.55E+08 m^3/d
Vegetable/fruit ingestion rate		--		42.7 kg/yr
Leafy vegetable ingestion rate		--		4.66 kg/yr
Exposure duration	1 yr	30 yr		30 yr
Dilution volume	91,000 L	--		--
Infiltration rate	0.18 m/yr	0.1 m/yr	0.01 m/yr	--
Waste thickness	0.5 m	6 m		--
Waste area	--	35 m x 35 m		--
Vadose zone thickness	0 m	18.3 m		--
Vadose zone dispersion	--	0 m	2.92 m	--
Distance to receptor well	0 m	17.5 m		--
Saturated zone thickness	--	15 m		--
Saturated zone K_d for Pu	--	22 mL/g	140 mL/g	--
Source term	Loose	Loose	Metal corrosion	Loose
Radionuclides of Concern	28*	^{14}C, ^{36}Cl, ^{3}H, ^{59}Ni, ^{239}Pu	^{14}C	11 (^{60}Co, ^{137}Cs)**

*There are too many radionuclides to list in the table.

**There are 11 radionuclides whose predicted concentrations exceeded their corresponding soil screening levels when the vessel is assumed to be left in-place. When the vessel is removed, then only Co-60 and Cs-137 exceed their screening levels (Staley 2006). Note that Co-60 and Cs-137 were two of the 28 radionuclides identified for additional study using the NCRP factors (McCarthy 2006).

for the ETR at INL. The decision to perform only point-value analyses was made despite recognition of various sources of uncertainty including inventories, source terms, soil concentrations, exposure characteristics, and fate and transport parameters. The approach used to manage the risk analyses in the face of these uncertainties for the removal action was to attempt to "err on the conservative side so that risks are over-estimated and bound any actual risk that might result…" (Staley 2006). For the groundwater pathway, the analyses progressed from a very simple NCRP screening analysis meant to be bounding for an initial screening to the next tier analysis employing GWSCREEN (with bounding assumptions) to identify contaminants of concern (COCs) and finally to an analysis to evaluate more representative risks associated with the COCs using GWSCREEN with more accurate parameters.

The approach to assessing groundwater pathway risks for the ETR included a progression from extremely simple and "conservative" calculations (using NCRP factors) to more and more accurate representations of expected conditions (using the GWSCREEN code). The screening assessment for vessel removal also only used simple and "conservative" calculations to address issues of uncertainty and did not take credit for any cementitious materials used. The results for both sets of analyses were the identification of (1) a number of contaminants of (potential) concern and (2) overall risks to receptors higher than the NCRP *de minimus* limit of $1x10^{-6}$ but lower than the action limit of $1x10^{-4}$. Therefore, it was deemed unnecessary to perform even more accurate analyses that might have taken credit for cementitious materials although this may have provided additional evidence to stakeholders that the measures taken were protective of human health and the environment.

5.1.2 Radioactive Waste Management Complex CERCLA Disposition

The Radioactive Waste Management Complex (RWMC) was created in 1952 for disposal of radioactive wastes at the USDOE Idaho Site. The complex consists of three major areas: the Subsurface Disposal Area (SDA), the Transuranic Storage Area, and the Administration and Operations Area. The SDA is the focus of remedial decision-making because buried wastes are the primary source of contamination (USDOE-ID 2008). A Final Record of Decision (ROD) was completed for the closure of the RWMC under the CERCLA process (USDOE-ID 2008). The final ROD was agreed upon based on an iterative set of baseline risk assessments and supporting studies performed under the CERCLA remedial investigation/feasibility study process (Becker et al. 1998; Holdren et al. 2006; Holdren et al. 2002). The baseline risk assessments performed for the SDA were based on point-value evaluations where uncertainty was addressed via multiple bounding sensitivity analyses. A brief summary of the approach adopted for uncertainty analysis in the SDA baseline risk assessments is provided in this section.

5.1.2.1 Modeling Approach

Because of the complexity of the RWMC, exposure and risk modeling relied on a modular approach as illustrated in Figure 9, which can also be seen as a representation of the conceptual model for risk analysis. To have risk, one needs inventory and a source term, release of contamination into the environment, transport of sufficiently persistent contaminants to receptors where they are exposed, and possible uptake of contaminants resulting in potential impacts. The modules used to estimate risks for the SDA contaminants follow this same basic conceptualization (Holdren et al. 2006):

- Waste Inventory and Location Database (WILD®) provides inventory estimates for each source area in the SDA (McKenzie et al. 2005).
- Disposal Unit Source Term – Multiple Species (DUST-MS®) computes the release of contaminants for the shallow subsurface (Anderson & Becker 2006; Sullivan 2001).

- TETRAD® computes contaminant fate and transport in the groundwater and at the surface for volatile inhalation (Magnuson & Sondrup 2006).
- DOSTOMAN® computes biotic uptake concentrations for surface pathways including external exposure, crop ingestion, soil ingestion, and dust inhalation.
- Risk calculations use standard USEPA methods to convert concentrations obtained from TETRAD® or DOSTOMAN® into a carcinogenic risk or hazard index.

For the modules identified in Figure 9, those for inventory (WILD®) and source-release modeling (DUST-MS®) would be most impacted by cementitious materials in the SDA[23]. The inventory impacts of cementitious materials are simply represented by whether or not contaminants originally buried in the SDA were stabilized in a cementitious waste form, within a cement-based container, or both. Historical information was used to differentiate contaminant inventories based on location, containment, and waste form in WILD® (McKenzie et al. 2005).

The DUST-MS® model was developed to estimate releases from low-level waste (LLW) disposal facilities due to infiltrating water (Figure 10) (Sullivan 2006). A LLW disposal facility is a “complex and heterogeneous collection” of wastes, waste forms, containers, soils, and engineered structures (including concrete vaults, backfill, vault covers, and drains) (Sullivan 2006). Contaminant release is often controlled by infiltrating water contacting a waste form resulting in release and potential transport outside the disposal unit. These release and transport processes are influenced by design of the unit, hydrological and geochemical properties, and waste form and container characteristics. Waste forms may include cements, resins, activated metals, and dry solids (Sullivan 2006).

DUST-MS® can be used to model container degradation, waste form release, and one-dimensional flow and transport using the method illustrated in Figure 11. The complexity of a disposal facility makes development of a three-dimensional, time-dependent model an extremely difficult task. The

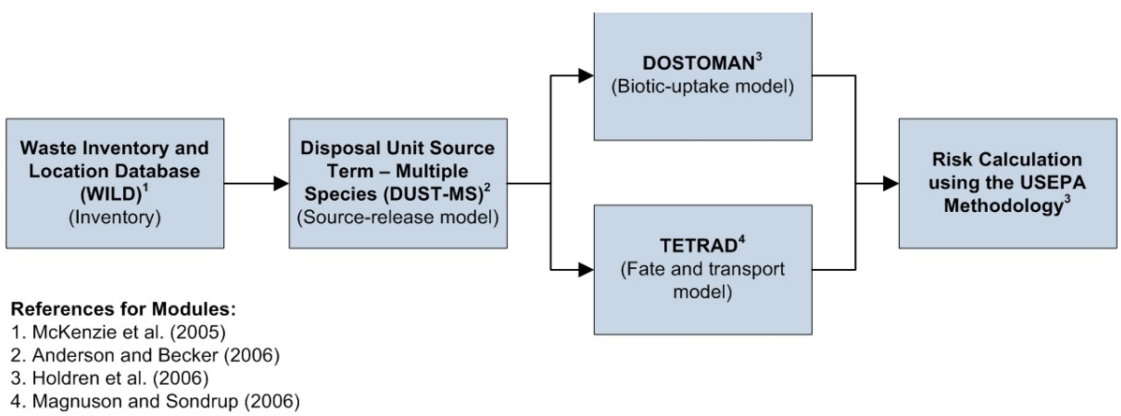

Figure 9. Risk Analysis Modules for the Idaho Subsurface Disposal Area (adapted from Holdren et al. 2006)

[23] Because this is a baseline risk assessment and thus no cementitious materials are being considered, fate and transport would not be impacted.

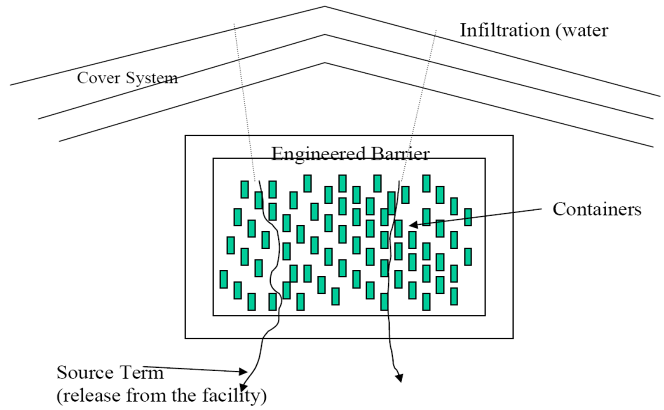

Figure 10. Low-Level Waste Disposal Facility Layout (reproduced from Sullivan 2006)

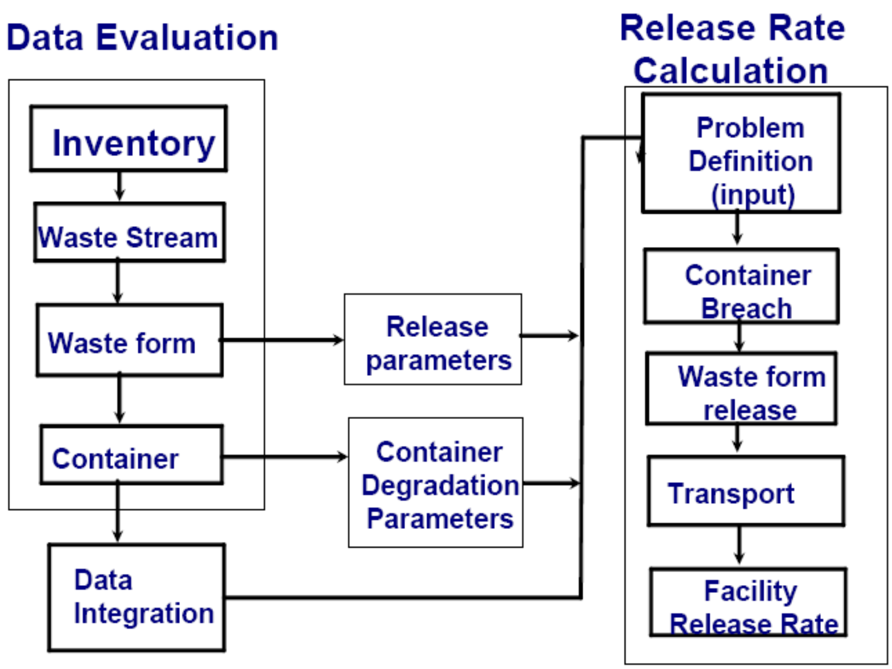

Figure 11. Procedure for Estimating Release Rates for a LLW Disposal Facility Using DUST-MS® (reproduced from Sullivan 2006)

applicability of such a model is impacted by data limitations (Sullivan 2006). Container degradation can result from failures that are instantaneous, uniformly distributed, or Gaussian. Four waste form release mechanisms are modeled: a) rinse with partitioning, b) diffusion release, c) dissolution release, and d) solubility-limited release (Sullivan 2006). The simplifications in the DUST-MS® model appear appropriate in that important processes are captured while retaining sufficient accuracy to make predictions that are not excessively conservative and thus useful for contaminant screening, parameter sensitivity analysis, and prediction of bounding release rates (Sullivan 2006).

5.1.2.2 Parameter Assumptions and Distributions

According to historical records, wastes were buried in the SDA in several types of containers including polyethylene bags, bottles, cardboard boxes, 55-gallon drums, wooden boxes, concrete casks, welded stainless steel containers, and resin tanks (Anderson & Becker 2006). To simplify modeling, only two types of containers were analyzed (only drums and polyethylene bags). Conservative failure times were used for the drums based on their configuration during original placement. No credit was taken for the potential effect of containment in concrete casks in the final baseline risk assessment (Anderson & Becker 2006; Holdren et al. 2006). The failure distribution and parameters used in the SDA baseline risk assessment are provided in Table 9.

Various waste forms were identified for the wastes buried in the SDA including activated metals, glass, resins, soil, sludge, concrete, and fuel specimens. These forms were evaluated and a reduced set of waste forms were analyzed in the final SDA baseline risk assessment: activated metals (including stainless steel and beryllium), Vycor glass, materials undergoing surface wash, resins, and fuel test specimens. For release purposes, the concrete and other cementitious waste forms buried in the SDA were assumed loose (i.e., contaminants are available for immediate release) or treated as materials prone to the surface wash mechanism. These waste materials have surface contamination that is readily leached by infiltrating water, which is controlled by partitioning between the waste form and water (Anderson & Becker 2006). Because waste-to-water distribution coefficients were not available for the various types of materials undergoing surface wash (including cementitious of materials), soil-to-water distribution coefficients were used. The parameters used in modeling the surface wash release in the SDA baseline risk assessment are provided in Table 10. Over the three phases of the baseline risk assessment process, site-specific values were used whenever possible.

5.1.2.3 Sensitivity and Uncertainty Analysis Approach

Because of the complexity of the SDA site, the results of point-value analyses using the modules identified in Figure 9 were used as the primary inputs for decision-making purposes under the CERCLA process. Known uncertainties in inventory, infiltration rates, interbed properties, etc. were evaluated using one-factor-at-a-time sensitivity and qualitative uncertainty analyses. The sensitivity analyses were primarily focused on effects via the groundwater pathway and included (Holdren et al. 2006):

- Inventory impacts: Risks were estimated using upper-bound inventories and produced estimates

Table 9. SDA Container Failure Assumptions and Parameters (Anderson & Becker 2006)

Container	Failure Distribution	Mean Time to Failure (years)	Standard Deviation (years)	Initial Drum Failure Fraction
Loose, boxes, concrete containers, other	None	N/A	N/A	N/A
Stacked drums	Gaussian	34.1	14.6	0.0
Dumped drums	Gaussian	11.7	5.0	0.285
Volatile organic compound drums	Gaussian	45.0	22.5	0.3

Table 10. Distribution Coefficients Used in Release Modeling for the SDA Baseline Risk Assessment (Anderson & Becker 2006)

Contaminant(s)	ABRA* (cm^3/g or mL/g)	RI BRA** (cm^3/g or mL/g)	Basis for Distribution Coefficient or Change
^{227}Ac	400	225	Based on sieving interbed material
$^{241, 243}$Am	450	225	Based on sieving interbed material
^{14}C (surface wash)	0.1	0.4	Plummer et al. (2004) suggest 0.5 ± 0.1 mL/g. Lower bound used.
^{14}C (resins)	0.1	19	Anderson and Becker (2006)
^{36}Cl	0		
^{129}I	0.1	0	Riley and Lo Presti (2004)
^{94}Nb	500		
^{237}Np	8	23	Leecaster and Hull (2004)
^{231}Pa	8		
^{210}Pb	270		
^{238}Pu	5100	2500	Based on sieving interbed material
^{239}Pu (mobile)	5100	0	Mobile fraction source release, surficial sediments, A-B interbed
^{239}Pu (nonmobile)	5100	2500	Nonmobile fractions and mobile fractions in B-C and C-D interbeds
^{240}Pu (mobile)	5100	0	Mobile fraction source release, surficial sediments, A-B interbed
^{240}Pu (nonmobile)	5100	2500	Nonmobile fractions and mobile fractions in B-C and C-D interbeds
^{226}Ra	575		
^{228}Ra	N/A	575	Not modeled in the ABRA; coefficient same as for ^{226}Ra
^{90}Sr	60		
^{99}Tc (surface wash)	0		
^{99}Tc (resins)	0	19	Anderson and Becker (2006)
$^{228, 229, 230, 232}$Th	500		
$^{232, 233, 234, 235, 236, 238}$U	6	15.4	Riley and Lo Presti (2004)
Chromium	N/A	0.1	Not modeled in the ABRA; coefficient from Becker et al. (1998)
Nitrate	0	0	
Carbon tetrachloride 1,4-Dioxane Methylene chloride Tetrachloroethylene Trichloroethylene	N/A	0	Not modeled in the ABRA; Release is diffusion-controlled so a distribution coefficient is not used.

*ABRA – Ancillary Basis for Risk Analysis (Holdren et al. 2002)
**RI BRA – Remedial Investigation and Baseline Risk Assessment (Holdren et al. 2006)

of approximately the same order of magnitude for most contaminants with the resulting total cumulative risk higher by approximately a factor of 2.

- Infiltration impacts —Three cases were examined: 1) reduced background infiltration outside the SDA producing slightly higher risk estimates, 2) low infiltration inside the SDA producing lower risk estimates, and 3) high uniform infiltration inside the SDA resulting in higher risks.
- Interbed regions—The potential effect of neglecting known gaps in the B-C sedimentary interbed was evaluated by eliminating this interbed in the model, which produced a negligible impact on predicted risks. Plutonium sorption was also neglected in the interbed sediments and this extremely conservative case increased risk predictions by several orders of magnitude.
- Low-permeability zone—Effects of the postulated low-permeability zone assumed for the SDA were evaluated using a sensitivity case that neglected such a region in the aquifer resulting in significantly lower risk estimates suggesting that the base-case model results are conservative.

The baseline risk assessments performed to support the CERCLA remedial investigation process for the SDA concluded that unacceptable risks were posed by the contaminants in the SDA. These assessments neglected the potential impacts from cementitious materials (i.e., concrete containers and waste forms) in estimating baseline risks for the SDA or evaluating the impacts of other uncertainties in the analyses. However, it is unlikely that consideration of cementitious materials would have changed the primary conclusion of the baseline risk assessment although it may have had impacts on the contaminants of concern identified in the process. Cementitious materials are included for the SDA remedial action for both the early action to grout the beryllium blocks to reduce the tritium and C-14 source term and in the selected remedial action for the SDA in which *in situ* grouting of soil vaults and trenches will be used to reduce the mobility of Tc-99 and I-129 and future risks to the aquifer and potential receptors (USDOE-ID 2008).

5.1.3 Waste Calcining Facility EPA Environmental Assessment and RCRA Landfill Closure

In 1998, the Waste Calcining Facility (WCF) located at the Idaho Nuclear Technology and Engineering Center (INTEC) on the USDOE Idaho Site was closed under an approved Hazardous Waste Management Act/Resource Conservation and Recovery Act (HWMA/RCRA) Closure Plan (INEL 1996). Because it was found not practical to clean close the WCF, the vessels, cells, and waste pile were grouted and covered with a concrete cap. This method of closing a RCRA facility as a landfill with mixed waste liabilities is considered innovative. Regulations for the WCF waste piles required preparation of closure and post-closure plans. The State of Idaho desired that the risk of release to be consistent with the Federal Facilities Agreement/Consent Order (FFA/CO) remedial goals (DOE-ID 1991); therefore, the USDOE assessed the radionuclide risks in parallel with the RCRA closure for hazardous constituents (Demmer et al. 1999)[24] . The risk assessment was performed in phases of increasing accuracy to help manage recognized uncertainties in assumptions and parameters (USDOE-ID 1996).

5.1.3.1 Modeling Approach

The risk assessment approach developed to support the WCF closure was also considered innovative (Demmer et al. 1999). To represent the source term a model was developed based on conservative

[24] The USDOE also assessed the WCF landfill closure using an Environmental Assessment (EA) to evaluate potential risks associated with hazardous and radioactive constituents using the same risk assessment methodology.

assumptions that represented process conditions and residual contaminants. The primary impact of cementitious materials on the WCF risk assessment was felt in the modeling performed to estimate risks from ingestion of groundwater contaminated by contaminants originally residing in the WCF. Impact modeling for the WCF was performed in two phases: a simple screening phase and a more detailed phase.

The initial screening phase was performed based on conservative assumptions (i.e., no concrete cap or grouting) using the GWSCREEN model for the groundwater pathway (Rood 1994). The conceptual model for GWSCREEN is illustrated in Figure 8. The RESidual RADioactivity (RESRAD®) model was used to estimate external exposure to residual radionuclides in the initial screening (Yu et al. 2001). RESRAD® is typically used to estimate doses and risks from residual radioactive materials to calculate operational guidelines for soil contamination (Yu et al. 2001). The exposure pathways considered in RESRAD® are illustrated in the cartoon in Figure 12 and the interrelationships among the various RESRAD® pathways are illustrated in Figure 13. Using the GWSCREEN and RESRAD® models and conservative assumptions resulted in four contaminants of potential concern (COPCs) for the WCF based on the NCP *de minimus* limit of $1x10^{-6}$: Np-237, Pu-239, Pu-240, and Tc-99 (USDOE-ID 1996). The more detailed screening using the PORFLOW® model indicated that Tc-99 (and overall risk) would exceed the *de minimus* limit but be well below the NCP action limit of $1x10^{-4}$.

The second phase of the assessment analyzed groundwater risks taking credit for both grouting within the WCF and the concrete cap using the PORFLOW® transport model (ACRi 2002). PORFLOW® is designed to solve problems involving the coupled transport of flow, heat and multiple chemical species in a complex 3D geometry, transient or steady-state fluid flow, fully or partially saturated media, single or multiple phase systems, and phase changes between liquid and solid and liquid and gaseous phases (ACRi 2002). The processes considered in PORFLOW® are represented in Figure 14 (ACRi 2002). For the WCF detailed screening, the concrete is assumed to crack allowing water to enter the cracked waste form in turn leaching contaminants; these contaminants are then transported into the surrounding soil.

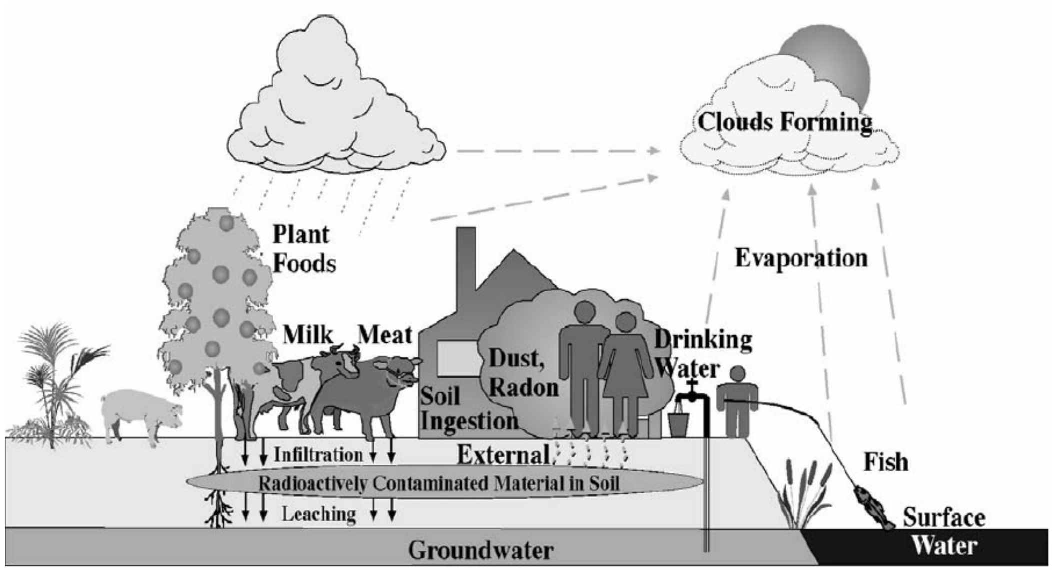

Figure 12. Exposure Pathways and Processes Considered in RESRAD®
(reproduced from Yu et al. 2001)

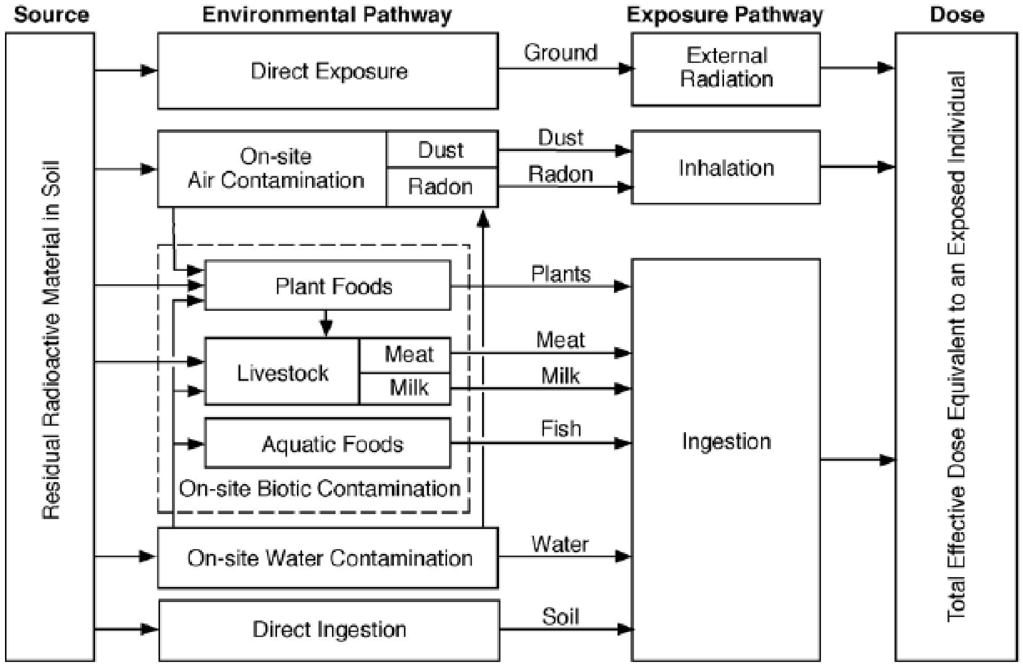

Figure 13. Schematic Representation of the RESRAD® Exposure Pathways (reproduced from Yu et l. 2001)

5.1.3.2 Parameter Assumptions and Distributions

Two different cases were simulated using the screening assessment approach developed for the WCF. Both the initial and detailed screening phases of the assessment used the same exposure parameters for the 30-yr future resident scenario. However, site-specific hydraulic transport parameters (e.g., hydraulic conductivity, pose size, moisture content, sorption and diffusion.) were included in the detailed assessment for the grouted waste form, concrete, sediments, and basalt used in the WCF (USDOE-ID 1996). Perhaps even more importantly because of the potential impact of infiltration on contaminant release, the detailed assessment model incorporated a very simple conceptualization of cracking and failure for the cap and grouted waste form as illustrated in Table 11.

5.1.3.3 Sensitivity and Uncertainty Analysis Approach

Point-value dose and risk analyses were used as the bases for decision-making for the WCF at the DOE Idaho Site. The approach used to manage the risk analyses in the face of uncertainties: (1) the behavior of the cementitious materials used and (2) fate and transport of contaminants was to "err on the conservative side so that risks are over-estimated and bound any actual risk that might result…" (Staley 2006). For the groundwater pathway, the analyses progressed from a simple and conservative screening analysis GWSCREEN (and ignoring cementitious materials) to a more detailed analysis using PORFLOW® with additional site-specific information and credit for the cap and grouted waste form used in the closure[25]. The detailed screening analyses identified Tc-99 as

[25] A single screening analysis was performed using the RESRAD® model for external exposure and identified no contaminants of potential concern.

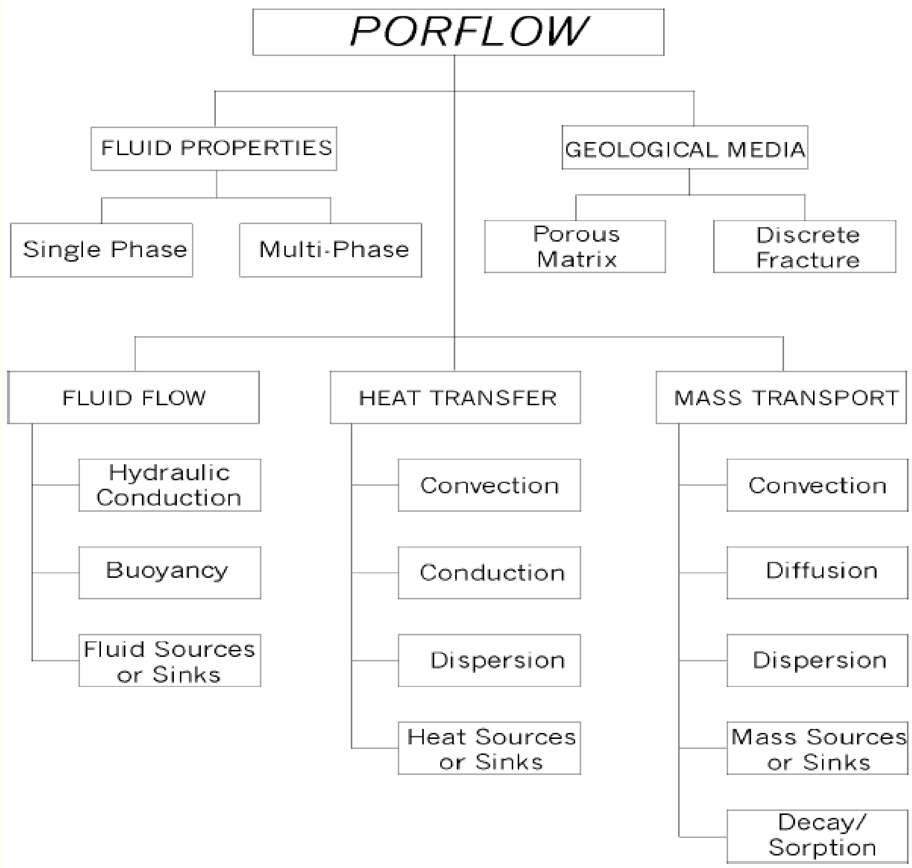

Figure 14. Properties Considered in PORFLOW® (SRS 1997a)

Table 11. WCP Screening Assessment Parameters and Assumptions for the Groundwater Pathway (USDOE-ID 1996)

Exposure Parameter or Assumption	GWSCREEN Screening	PORFLOW Screening (COCs)
Drinking water intake	2 L/d	2 L/d
Exposure duration	30 yr	30 yr
Infiltration rate	Not available	Not available
Waste thickness	Not available	Not available
Waste area	Not available	Not available
Vadose zone thickness	Not available	Not available
Distance to receptor well	Not available	Not available
Saturated zone thickness	Not available	Not available
Credit taken for cap or grout	No	Yes
Time to cracking for cap and grout	N/A	100 yrs*
Radionuclides of (Potential) Concern	^{237}Np, ^{239}Pu, ^{240}Pu, ^{99}Tc	^{99}Tc

*After cracking the cap and grout, water flows unimpeded through these barriers.

the only contaminant of concern with a groundwater pathway risk higher than the NCRP *de minimus* limit of $1x10^{-6}$ but lower than the action limit of $1x10^{-4}$. It was deemed unnecessary to perform even more accurate analyses that might have taken additional or more accurate credit for cementitious materials although protective of human health and the environment.

5.2 Savannah River Site

5.2.1 Tanks 17-F and 20-F Closure Actions under SCDHEC Industrial Wastewater Permits and NEPA Environmental Impact Statement

The 51 high-level waste (HLW) tanks in the SRS F-Area and H-Area Tank Farms are permitted under a waste water operating permit and closure will be at least partly through closure of the wastewater operating permit (Picha et al. 1999). In 1995 the DOE began to prepare for closure of HLW tanks by preparing both a closure plan (SRS 1996) and an Environmental Assessment (EA)[26] to evaluate alternatives for the closure of these tanks (USDOE-SR 1996a). SRS Tanks 17-F and 20-F were operationally closed in 1996 under South Carolina Department of Health and Environmental Control (SCDHEC) industrial wastewater permits (SRS 1997a; SRS 1997b). Bulk waste was removed to the extent practical, oxalic acid was used to clean the tanks, and grouting for closure was carried out in three stages[27] (Elmore & Henderson 2002; Picha et al. 1999). Point-value risk evaluations supported by sensitivity analyses were performed to demonstrate that tank closures would ensure overall protection of human health and the environment (SRS 1997a; SRS 1997b). The risk evaluation for the SRS Tank 17-F closure will be used as an example because that for Tank 20-F is very similar.

5.2.1.1 Modeling Approach

The primary impact of cementitious materials on the tank closure risk analysis was in modeling fate and transport of residual contaminants from the grouted material to the aquifers and ultimately receptors. A relatively simple conceptual model (as illustrated in-Figure 15) was developed for the Tank 17-F closure. Transport modeling for the groundwater pathway was performed using the Multimedia Environmental Pollutant Assessment System (MEPAS) computer code to estimate concentrations and doses to the receptors identified in Figure 15 (Droppo et al. 1989; Strenge & Chamberlain 1995). MEPAS is a "physics-based environmental analysis code that integrates source-term, transport, and exposure models" for site-specific assessments of endpoints including concentration, dose, or risk (Strenge & Chamberlain 1995)[28] and was thus appropriate for the analysis of the Tank 17-F and 20-F closures. MEPAS was used to estimate concentration, doses, and lifetime risks for both radiological and hazardous contaminants due to contaminant release and subsequent transport in the saturated zones under and near the SRS F-Tank Farm. The results of the MEPAS analysis indicated that none of the known performance objectives would be exceeded during the 10,000-yr period simulated.

5.2.1.2 Parameter Assumptions and Distributions

The primary driver for risk is a source of contamination. Without a source there is no risk. The inventory used for modeling Tank 17-F closure was intended to be conservative. Concentrations 20 percent greater than the analyzed concentrations were assumed for contaminants remaining in the tank after bulk waste

[26] The result of the EA process was a Finding of No Significant Impact (FONSI) in which it was concluded that closure of the SRS HLW tanks in accordance with the closure plan would not result in significant environmental impacts (USDOE-SR 1996b).

[27] A reducing grout was initially added to stabilize residual wastes. A large layer of a controlled low-strength grout material was then added and then each tank was capped by the addition of a high-strength grout (Picha et al. 1999).

[28] MEPAS contains a sensitivity module that can be used for uncertainty analysis (Strenge & Chamberlain 1995).

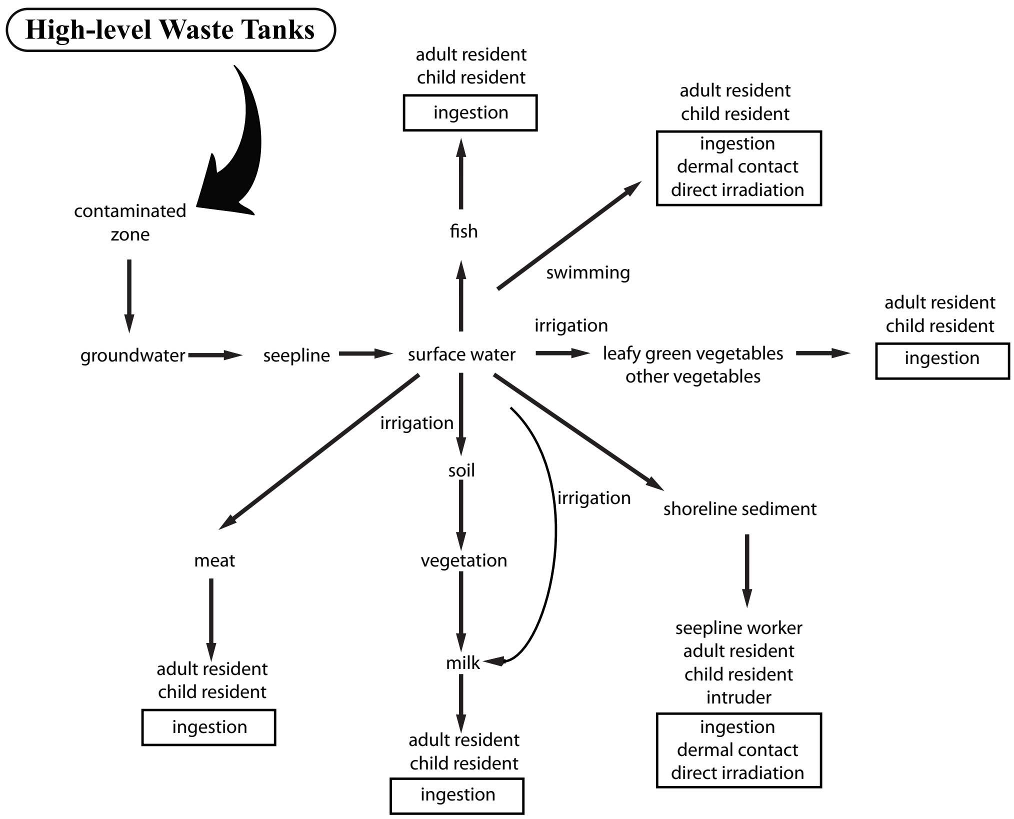

Figure 15. Potential Exposure Pathways for Human Receptors for SRS Tanks 17-F and 20-F (SRS 1997a)

removal and washing of the ancillary equipment and piping. One thousand three hundred and sixty killograms (3,000 lb) of mercury were assumed to remain in the tank.

The groundwater fate and transport model in MEPAS is based on a simple linear partitioning type model employing site-specific K_ds whenever possible (Whelan, McDonald & Sato 1996). These partition coefficients are a strong function of the REDOX conditions of the environment. For the Tank 17-F model, eight distinct strata were identified including the contaminated zone, concrete basemat, vadose zone, two clay layers, and three saturated zones. Distribution coefficients selected for these materials are provided in Table 12. Other parameters needed to model contaminant fate and transport through the vadose and saturated zones are summarized in Table 13.

Upon closure, the tanks were filled with three layers of grout. Based on the E-Area Vaults performance assessment (Cook & Hunt 1994), a conservative assumption was made that the basemat, grout, and tank top failed at 1,000 years (SRS 1997a). The leach rate of contaminants was ultimately limited by the layer with the lowest hydraulic conductivity either above or below the contaminated zone. Therefore, hydraulic conductivties are critical to the results of the risk assessment (SRS 1997a). Upon failure, the hydraulic conductivity of the basemat was assumed to be that of sand and the infiltration rate was increased to 40 cm/yr. The impact of an engineered cover over the tank

Table 12. Selected Radionuclide and Chemical Partition Coefficients (K_d) used in the Tank 17-F Model and 20-F (SRS 1997a)

Contaminant	SRS Soil (cm^3/g)	Note	Reducing contaminated zone (cm^3/g)	Note	Reducing concrete	Note	Clay (cm^3/g)	Note
^{14}C	2	a	0.1	b,c	0.1	c	1	d
$^{244, 245}Cm$	150	a	5000	c	5000	c	8400	d
^{129}I	0.6	a	2	c	2	c	1	d
Tritium	0	a	0	c	0	c	0	d
^{237}Np	10	a	5000	c	5000	c	55	d
$^{238, 238, 240, 241, 242}Pu$	100	a	N/A	j	N/A	j	5100	d
^{79}Se	5	a	0.1	c	0.1	c	740	d
^{99}Tc	0.36	a	1000	c	1000	c	1	d
Ba	530	e	1	c,h	1	c,h	16000	g
Cr(VI)	16.8	e,i	7.9	f,i	7.9	f,i	360	g,i
Pb	234	e	500	c	500	c	1830	g
Hg	322	e	5280	f	5280	f	5280	g
Nitrate	0	e	0	f	0	f	0	g
Ag	0.4	e	1	c	1	c	40	g
U	50	a	N/A	j	N/A	j	1600	d

a. WSRC (1994) value for soil
b. Assumed similar to selenium
c. Bradbury and Sarott (1995)
d. WSRC (1994) value for clay
e. MEPAS Default (soil < 10% clay and pH 5-9)
f. MEPAS Default (soil > 30% clay and pH > 9)
g. MEPAS Default (soil > 30% clay and pH 5-9)
h. Assumed the same as strontium (Bradbury & Sarott 1995)
i. All chromium modeled as Cr(VI)
j. Solubility limit used to estimate K_d (Cook & Hunt 1994)

Table 13. MEPAS Groundwater Parameters for Vadose and Saturated Zones for the Tank 17-F Model and 20-F (SRS 1997a)

Parameter*	Concrete basemat		Vadose zone	Water table aquifer	Tan Clay layer	Barnwell-McBean Aquifer	Green Clay layer
	Intact 0-1000 yr	Failed 1000-10,000 yr					
Thickness (ft)	0.58	0.58	5.4	40.0	3.0	60.0	5.0
Bulk density (g/cm^3)	2.21	1.64	1.59	1.59	1.36	1.59	1.39
Total porosity	0.15	0.38	0.35	0.35	0.40	0.35	0.40
Field capacity	0.15	0.09	0.12	0.35	0.334	0.35	0.325
Longitudinal dispersion (ft)	0.0058	0.0058	0.054	0.40	0.030	0.60	0.050
Vertical hydraulic conductivity (cm/s)	$9.6x10^{-9}$	$6.3x10^{-3}$	$7.1x10^{-3}$	$7.1x10^{-3}$	$1.6x10^{-6}$	$5.6x10^{-4}$	$4.4x10^{-9}$

*Parameters in this table are provided in the original units. Refer to SRS (1997a) for details concerning where values were taken as many reports are unavailable.

after closure was not evaluated[29]. Table 13 provides the hydraulic conductivity for the basemat and infiltration rate as a function of simulation time.

5.2.1.3 Sensitivity and Uncertainty Analysis Approach

Point-value dose and risk analyses were used as the bases for decision-making for the operational closures of the Tank 17-F and Tank 20-F at SRS. For the groundwater pathway, the analyses were based on a MEPAS model with site-specific information and credit taken for the cementitious materials used in the closure (including grout layers and a concrete basemat). The approach used to manage the impacts of recognized uncertainties in inventory, hydraulic properties, partition coefficients, site geometries, dispersion, etc. was to perform one-parameter-at-a-time sensitivity analysis based on these uncertainties. These results indicated sensitivities in predicted risks to the source term and strata properties including dispersion. The analysis indicated that none of the known performance objectives would be exceeded during the 10,000-yr period simulated even for "conservative" risk estimates incorporating known uncertainties. It was deemed unnecessary to perform even more rigorous analyses that might have taken more accurate credit for cementitious materials although this may have provided additional evidence to stakeholders that the measures taken were protective of human health and the environment.

5.2.2 P-Reactor In-Situ Decommissioning Risk Assessment

The P-Reactor facility is being decommissioned under the CERCLA process. A risk assessment was conducted as one input for selection of the preferred closure option in the feasibility study (Council 2008). The risk assessment included a combination of deterministic and probabilistic calculations using the GoldSim® platform. This section includes a brief summary of the approach adopted for the uncertainty analysis.

5.2.2.1 Modeling Approach

A relatively simple conceptual model was developed and implemented in the GoldSim® platform (e.g., see Fig. 16). The model in Fig. 16 was used for the reactor vessel portion of the facility, which will be used for this example. Models were also developed for other parts of the P Reactor Facility and the results of all of the models were summed to provide a comprehensive view of risk. As shown in Fig. 16, the reactor vessel was modeled as a one dimensional system with five different materials. Two or three dimensional aspects of the problem were not addressed. One diemensional problems are very well suited for implementation in GoldSim® for probabilistic assessments involving many realizations.

Since the base model is relatively simple, there was no need for abstraction or upscaling from a complex model to a simplified model. Only one conceptual model was used for each option considered in the assessment.

5.2.2.2 Parameter Assumptions and Distributions

Six different materials were simulated in the P- Reactor model: stainless steel, concrete, grout, and three different soils (vadose zone, A/AA Horizon, and the transmissive zone (TZ). As shown in Figure 16, the vadose zone was not included in the reactor vessel submodel. The input values assumed for the stochastic parameters are shown in Table 14. Values and input distributions for concrete and grout were taking from accual SRS materials. The

[29] Previous modeling of tank closure scenarios demonstrated that a cap over a grout-filled tank is likely to have little impact at the point of exposure (SRS 1997a). Impacts for a grout-filled tank with a cover were assumed to be the same as for a grout-filled tank with no cover with an appropriate delay.

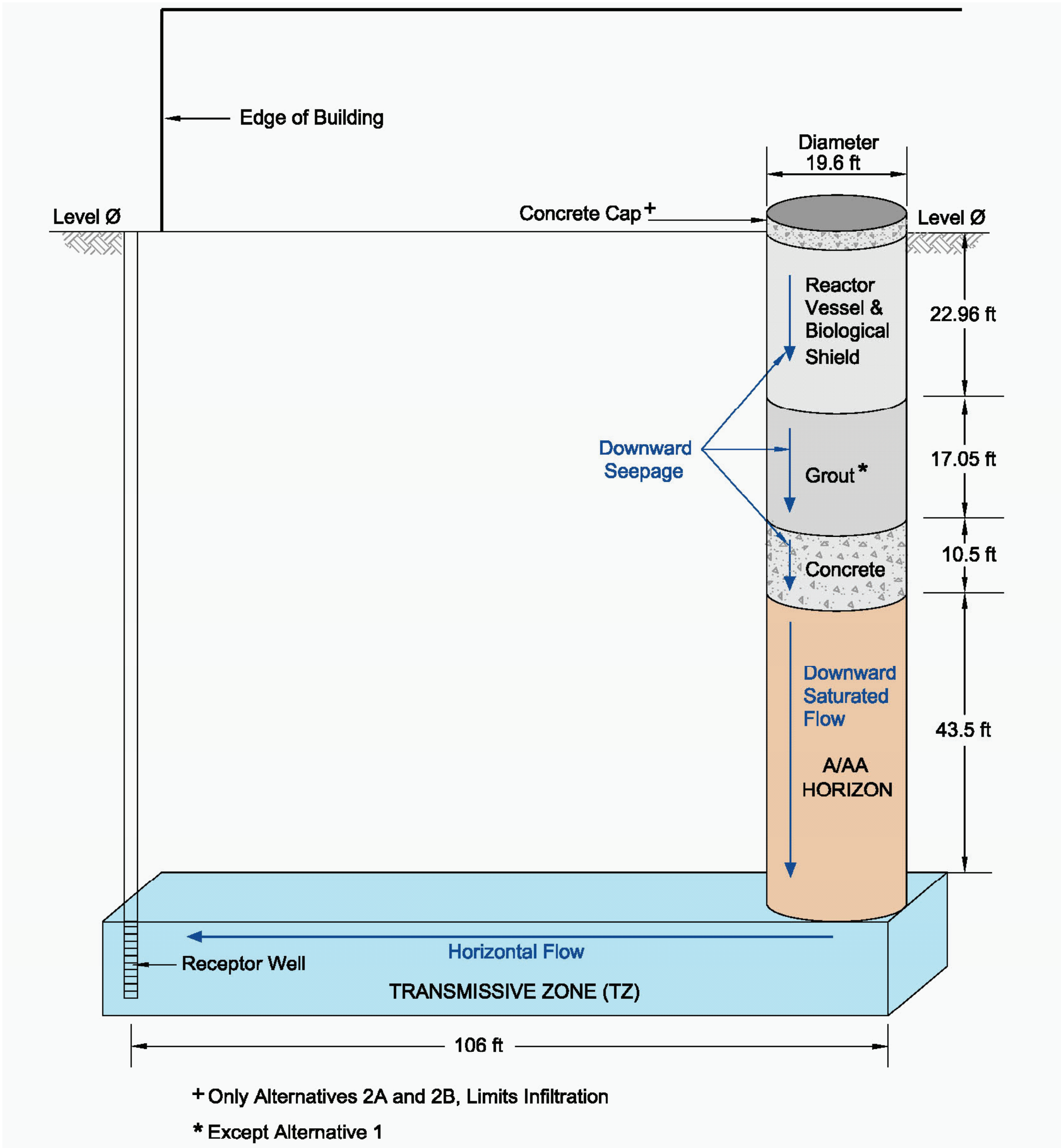

Figure 16. Conceptual Model for P-Reactor Vessel (Council 2008)

distributions for soils and cementitious materials were developed based on site-specific information from other areas of the Savannah River Site.

The hydraulic gradient in the TZ was also assumed to be a log-normally distributed stochastic variable with a geometric mean of 0.019 and a standard deviation of 1.05. Distribution coefficients were assigned two values for each element, a best estimate and a "conservative" value. The best-estimate and conservative values are used to define normally distributed inputs. The best-estimate was used as the mean and the standard deviation was calculated from one half of the difference between the mean and conservative value. The values used were based on site-specific values developed for other assessments or generic values from the literature.

Table 14. Example Stochastic Material Properties for P Reactor Risk Assessment (Council 2008). USDOE-RL 2001b)

	Mean (default)	Distribution	Std Deviation
Concrete			
Porosity	0.168	Normal	0.02
Initial Hydraulic Conductivity	3.5×10^{-8} cm/s	Log-normal	10
Grout			
Porosity	0.266	Normal	0.02
Initial Hydraulic Conductivity	3.6×10^{-8} cm/s	Log-normal	10
Stainless Steel			
Corrosion rate	0.0006 lb/yr/ft2 (0.0007 lb/yr/ft2)	Log-normal	2.9
A/AA Horizon			
Porosity	0.3	Normal	0.0275
Vertical Hydraulic Conductivity	0.04 ft/d, truncated at 0.0003 ft/d and	Log-Normal	0.03 ft/d
Transmissive Zone			
Porosity	0.25	Normal	0.06
Horizontal Hydraulic Conductivity	20 ft/d	Log-Normal	9 ft/d

Note: Mean and standard deviation are geometric for the lognormal distribution. Default value for deterministic case is shown in parentheses if different from mean.

5.2.2.3 Sensitivity and Uncertainty Analysis Approach

A deterministic case using best estimate inputs was used as the primary basis for decision making. One-factor-at-a-time sensitivity analyses were also conducted for the infiltration rate, corrosion rate, and distribution coefficient to provide additional information regarding the relative sensitivity of the results to those variables. The results of the sensitivity analysis for the corrosion rate are shown in Figure 17.

One thousand Monte Carlo realizations were modeled for the uncertainty analysis using the stochastic parameters. The results of the uncertainty analysis were used to illustrate the possible range of results for each alternative considered in the analysis. Regression based sensitivity analyses were also conducted based on the Monte Carlo simulations. The steel corrosion rate was shown to be the most important variable to the results based on the sensitivity analysis.

5.3 Hanford Site

5.3.1 221-U Facility Remedial Actions Under CERCLA and NEPA

The Hanford 221-U Facility was placed in standby in 1958 and subsequently retired. The Washington State Department of Ecology established that the CERCLA Remedial Investigation/Feasibility Study process would be used to evaluate potential remedial actions and identify preferred remedial alternatives for the 221-U Facility (DOE-RL 2005)[30]. The selected remedy for the facility included waste removal from vessels and equipment, removal and treatment of liquids, grouting of internal vessel spaces, demolition of various structures followed by stabilization

[30] Consistent with past practices at the USDOE Hanford Site (Thompson 1991), a traditional remedial investigation including a baseline risk assessment was not performed for the 221-U Facility so that additional resources could be focused on the remedial action phase (USDOE-RL 2001b). Instead risk analyses for baseline and closure conditions and to define preliminary remediation goals (PRGs) were provided in the final feasibility study report for the 221-U Facility (USDOE-RL 2001b).

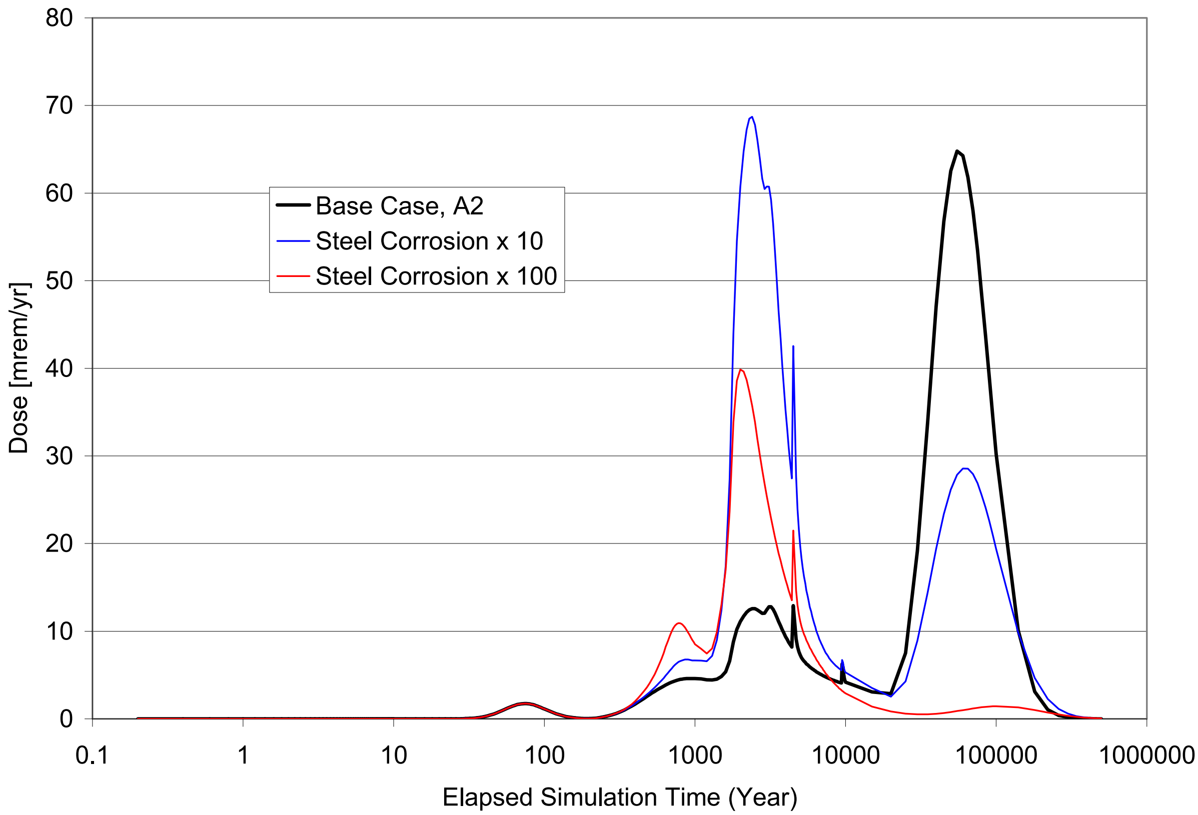

Figure 17. Sensitivity Results for Different Corrosion Rates in P Reactor Assessment (Council 2008)

to support an engineered barrier, construction of the barrier, institutional controls, barrier inspection and maintenance, and barrier performance and groundwater monitoring. The risk assessment performed to support the CERCLA process was performed using the RESidual RADioactivity (RESRAD®) code (Yu et al. 2001) for radionuclide doses and the Hanford Site Risk Assessment Methodology (HSRAM) (USDOE-RL 1995) for non-carcinogenic impacts. These calculations were supported by evaluating the ranges of risks corresponding to the range of contaminant concentrations in the facility. The calculations indicated that baseline conditions posed unacceptable risks, but the selected remedial actions would protect human health and the environment based on an industrial use scenario (USDOE-RL 2001b; USDOE-RL 2005).

5.3.1.1 Modeling Approach

The conceptual site model for the Hanford 221-U Facility is provided in Figure 18 which illustrates the linkages among the contaminant source, release mechanisms, exposure media and routes, and receptors for the facility. This conceptualization of the facility was implemented in the RESRAD® model to estimate doses from radionuclides via external gamma exposure, inhalation, and ingestion using an industrial use scenario. RESRAD® is typically used to (1) estimate doses and risks from residual radioactive materials (2) calculate operational guidelines for soil contamination (Yu et al. 2001). Conceptual diagrams for RESRAD® were provided in Figure 12 and Figure 13.

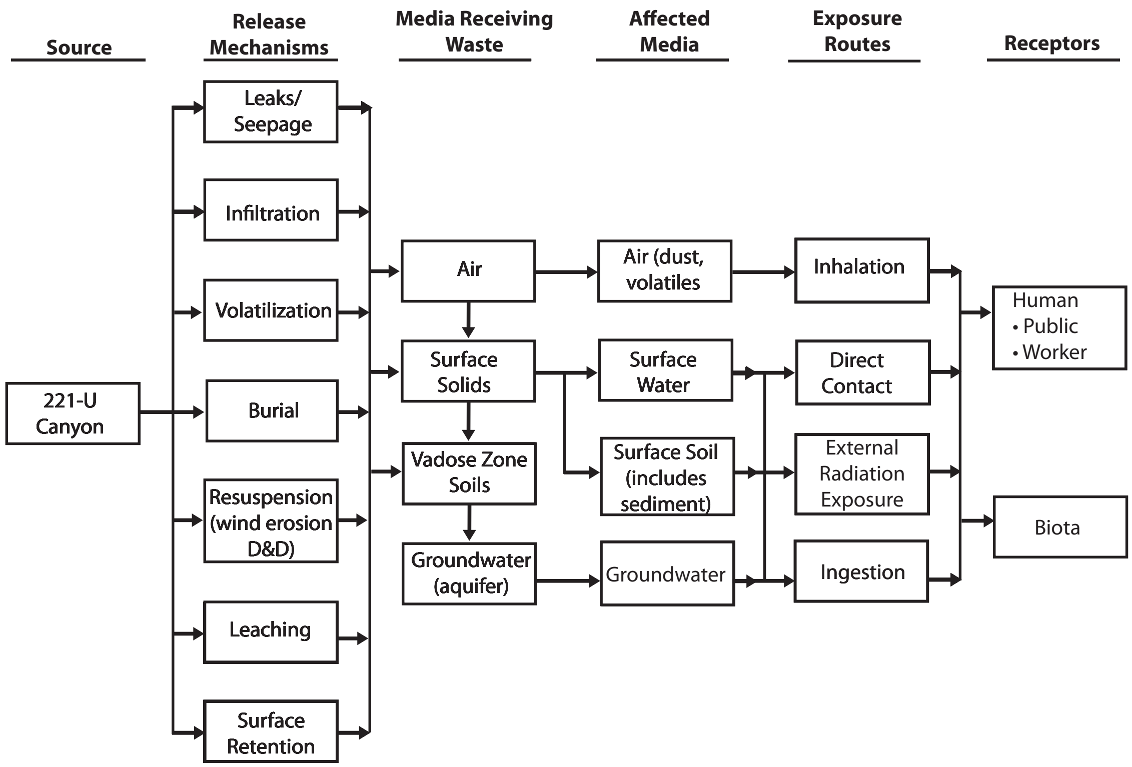

Figure 18. Conceptual Site Model for the Hanford 221-U Facility (reproduced from USDOE-RL 200b)

HSRAM was used to estimate non-carcinogenic impacts. HSRAM is a specifically tailored risk assessment approach (using USEPA and State of Washington guidance) which supports CERCLA risk assessments by focusing conservatively on probable human health impacts (USDOE-RL 1995).

5.3.1.2 Parameter Assumptions and Distributions

Two scenarios were evaluated using the RESRAD® model: industrial use and groundwater protection. The maximum baseline risks for the 221-U Facility were predicted based on the industrial use scenario. Using the RESRAD® model, the 221-U Facility was found to pose unacceptable baseline risks based on the industrial use scenario and remedial actions are necessary.

An evaluation of risks to the groundwater pathway was also performed using the RESRAD® model developed for the facility. The model was used to predict whether residual contaminants would be likely to reach the groundwater within 1,000 years after cleanup, and if so, also estimate the corresponding groundwater concentrations, doses, and risks (USDOE-RL 2001b). Important parameters and those that vary between the industrial use and groundwater protection scenarios are described in Table 15. Some parameters in the model that might impact the results (e.g., erosion rate and hydraulic gradient.) are set to RESRAD® default values without parameter sensitivity analyses being performed. The values for the parameters were selected to provide higher than expected results. The risks posed by residual contamination at the 221-U Facility were found to be unacceptable without a surface barrier to limit infiltration into the site.

Table 15. RESRAD Input Parameters for the Hanford 221-U Facility Model (USDOE-RL 2001b)

Category	Parameter	Units	Industrial Scenario	Groundwater Protection	Rationale
Exposure Pathways*	External Gamma Inhalation Soil Ingestion Drinking Water Ing.		Active Active Active Suppressed	Suppressed Suppressed Suppressed Active	200 Area Industrial-Exclusive scenario includes only external gamma, inhalation, and soil ingestion pathways.
Contaminated Zone (CZ)	Thickness of CZ	m	4.6		WAC 173-340 (2007)
	Dose Limit	mrem/yr	15 and 50	4	200 Area industrial scenario and groundwater protection
CZ Hydrological Data**	Density	g/cm^3	1.6		USDOE-RL (2001a)
	Erosion Rate	m/yr	0.001		RESRAD Default
	Total/Effective Porosity		0.34/0.25		USDOE-RL (2001a)
	Hydraulic Conductivity	m/yr	300		USDOE-RL (2001a)
	ET Coefficient		0.91		WDOH (1997)
	Wind Speed	m/s	3.4		Missing reference
	Precipitation	m/yr	0.16		Average annual rainfall (missing reference)
Saturated Zone (SZ) Hydrological Data**	Density	g/cm^3	1.9		USDOE-RL (2001a)
	Total/Effective Porosity		0.27/0.23		USDOE-RL (2001a)
	Hydraulic Conductivity	m/yr	365000		USDOE-RL (2001a)
	Hydraulic Gradient		0.0001		RESRAD Default
	Water Table Drop Rate	m/yr	0.001		RESRAD Default
	Well Pump Intake Depth	m	4.6		Typical RCRA well screen depth
	Well Pumping Rate	m^3/yr	250		RESRAD Default
Unsaturated Zone (SZ) Hydrological Data**	Thickness	m	50		Generic 200-Area site model
	Density (Soil)	g/cm^3	1.9		USDOE-RL (2001a)
	Total/Effective Porosity		0.27/0.23		USDOE-RL (2001a)
	Hydraulic Conductivity	m/yr	700		USDOE-RL (2001a)
Occupancy, Inhalation, and External Gamma	Inhalation Rate	m^3/yr	7300		WDOH (1997)
	Mass Loading (Inhalation)	g/m^3	0.0001		WDOH (1997)
	Exposure Duration	yr	30		WDOH (1997)
	Indoor Dust Filtration Factor		0.4		RESRAD Default
	External Gamma Shielding Factor		0.8		WDOH (1997)
	Indoor Time Factor		0.137		200 Area industrial scenario (60% indoors)
	Outdoor Time Factor		0.091		200 Area industrial scenario (40% outdoors)
Ingestion Pathway Data, Dietary	Soil Ingestion	g/yr	36.5	0	WDOH (1997)
	Drinking Water Intake	L/yr	0	730	WDOH (1997)
Ingestion Pathway Data, Nondietary	Groundwater Fractional Use (Drinking Water)		0	1	WDOH (1997)
	Depth of Soil Mixing Layer	m	0.15		RESRAD Default

*These pathways are suppressed in both scenarios: plant, meat, milk, and aquatic food ingestion and radon.
**Site-specific partition coefficients (K_d's) were used (USDOE-RL 2001b).

5.3.1.3 Sensitivity and Uncertainty Analysis Approach

Point-value predictions using "conservative" inputs over the expected ranges of contaminant concentrations were used as the primary basis for decision-making for the Hanford 221-U Facility. In fact the only consideration of uncertainty taken into account was in the contaminant concentrations. Uncertainties in the other parameters listed in Table 15 were not evaluated because neither baseline nor residual contaminant levels would be protective without an engineered cap over the 221-U Facility after closure. The assessment results indicated protectiveness for the selected remedial alternative were based primarily on the long-term effectiveness of the engineered cap that will be placed on the facility after the structure is demolished and vessels are grouted in-place (USDOE-RL 2001b)[31]. No credit was taken for cementitious materials in the modeling performed to support the ROD for the 221-U Facility. The only credit that was taken for cementitious materials (i.e., grouting) in the selected remedial alternative for the Hanford 221-U Facility was as a "defense-in-depth" measure if the engineered barrier fails during the 1,000-years simulation period (USDOE-RL 2005).

5.3.2 Tank Waste Remediation System Final Environmental Impact Statement under NEPA

The proposed action analyzed is the management and ultimate disposal of wastes in the Hanford Tank Waste Remediation System (TWRS) (USDOE-RL 1996). From 1943 to 1989, the principal mission of the Hanford Site was the production of weapons-grade plutonium and the corresponding chemical separations processes. Large volumes of radioactive wastes were generated and stored in 177 large underground tanks in the Hanford 200 Areas (including 28 double-shell tanks and 149 single-shell tanks) and 60 smaller active and inactive underground tanks. Past practices have resulted in extensive contamination in the soils beneath the 200 Areas especially near waste management facilities and locations of unplanned releases. Contaminants have migrated to the groundwater and toward the Columbia River (USDOE-RL 1996).

As a result of the NEPA process at Hanford, an EIS was prepared to address safe storage and disposal alternatives for the tank wastes. The focus of the EIS is the alternatives analysis. Alternatives were selected to represent the wide range of possibilities for Hanford tank wastes and were grouped into four categories based on the extent of waste retrieval as illustrated in Figure 19 (USDOE-RL 1996). One potential option for treating low-activity tank wastes upon retrieval is grouting; another is vitrification. Grouting of low-activity wastes was removed from consideration in the TWRS EIS (USDOE-RL 1996):

> "… as a result of concerns with the adequacy of disposal of low-activity waste using grout to immobilize the waste . The concerns involved the ability of grout to adequately inhibit contaminants leaching from the grouted waste and the ability to safely retrieve the waste from the grout vaults in the future, if retrieval became necessary for some reason."

5.3.2.1 Modeling Approach

Various assessments were performed to evaluate baseline, remedial, and post-remedial-action conditions to workers and the general public for actions related to the TWRS. Short-term and long-term baseline and post-remediation risks to residential and industrial receptors associated with the Hanford waste tanks were evaluated using the VAM2D model for the groundwater pathway (Huyakorn, Kool & Robertson 1989).

[31]A separate RESRAD® study was performed to evaluate groundwater risks after placement of a cap; however, known uncertainties in the parameters describing cap performance were not addressed.

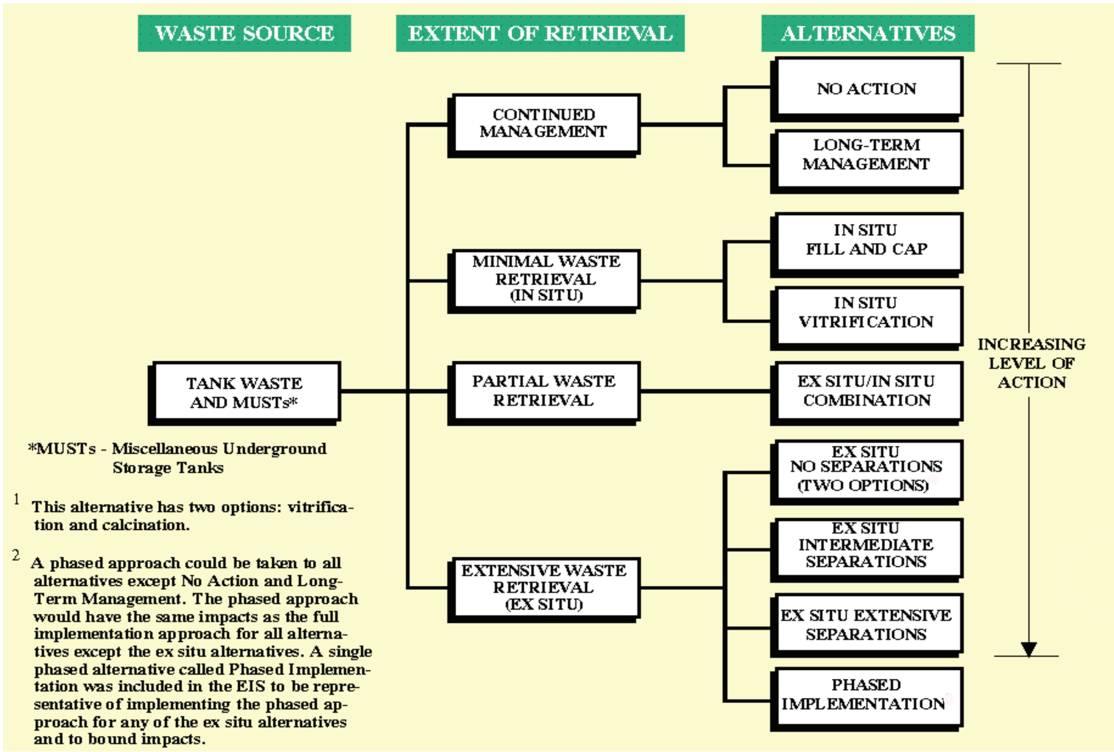

Figure 19. Tank Waste Remedial Alternatives (reproduced from USDOE-RL 1996)

VAM2D is a 2D finite element model simulating fluid flow and solute transport in variably saturated porous media[32].

Although risks evaluated in an EIS are typically intended to be bounding, one-at-a-time sensitivity analyses supplemented with Monte Carlo probabilistic analysis were used to assess the impacts of uncertainties in: remedial alternatives, source, fate and transport, and health effects .

Because of the nature of the TWRS and its disposition, ten complex scenarios were evaluated for the TWRS EIS for a large number of potential public and industrial receptors. The groundwater impacts were modeled using the VAM2D model (Huyakorn, Kool & Robertson 1989). A typical problem modeled with VAM2D is illustrated in Figure 20 which shows the typical information needed for two-dimensional (2D) modeling of transient flow and transport in variably saturated porous media. A 2D analysis was deemed appropriate because of subsurface conditions and availablity of sufficient data to develop a three-dimensional (3D) flow and transport model (USDOE-RL 1996). The VAM2D code only included single-phase flow (i.e., of water) and ignored other phases (e.g., air or other non-aqueous phase). Kinetic sorption effects were not addressed and evaluated groundwater flow was evaluated under steady-state conditions (USDOE-RL 1996).

[32] Intruder risks for the areas associated with the TRWS were evaluated under a previous Hanford performance assessment and will not be discussed here (Rittmann 1994).

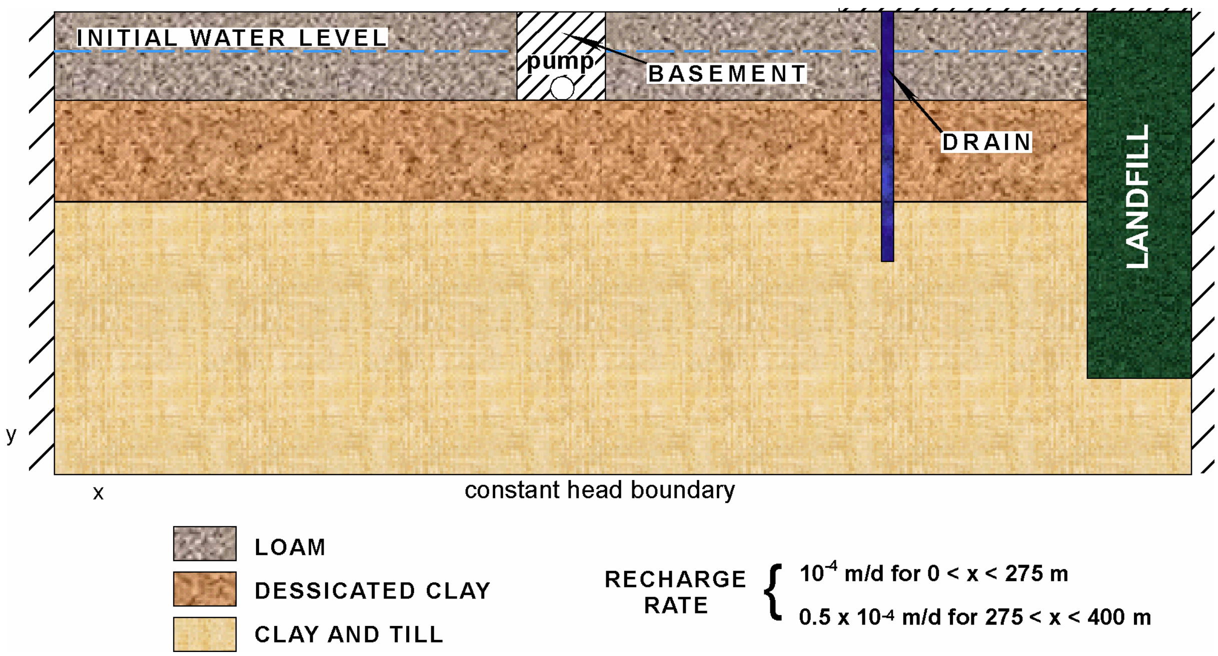

Figure 20. Conceptual Model for a Transient Flow Problem In An Unconfined Groundwater System Adjacent to A Landfill (adapted from Huyakorn et al. 1989)

5.3.2.2 Parameter Assumptions and Distributions

Often the first step in modeling risks is evaluating the contaminant source. The Hanford tanks and the proposed low-activity waste vaults were grouped based on proximity and inventory into nine source areas for groundwater pathway analysis as illustrated in Table 16. The contaminant source concentrations were evaluated for each of the disposal alternatives and baseline conditions. For retrieval alternatives, a 99% recovery was assumed leaving 1% of the initial contaminants including those that are water soluble, which is likely conservative. The *ex situ* treatment

Table 16. Source Area Designations and Description for the TWRS FEIS (USDOE-RL 1996)

Source Area Designation	Location	Single-shell Tanks	Double-shell Tanks	Vaults	Equivalent Area (m^2)
1WSS	200 West	40	--	--	15000
2WSS	200 West	43	--	--	16000
3WDS	200 West	--	3	--	1200
1ESS	200 East	40	--	--	15000
2ESS	200 East	16	--	--	5000
3EDS	200 East	--	11	--	4500
4ESS	200 East	10	--	--	4100
5EDS	200 East	--	14	--	5700
LAW vaults (proposed)	200 East	--	--	TBD	TBD
Total	--	149	28	TBD	--

alternatives were based on varying separation efficiencies between high-level and low-activity waste streams.

Modeling the groundwater pathway requires understanding contaminant releases and source term. For the single- and double-shell tanks, release is assumed to begin at the end of the institutional control period (100 years)[33]. Contaminant release is conservatively based on a congruent dissolution model where constituents are released in proportion to the most abundant constituent, in this case, nitrate. Thus the product of the rate of nitrate dissolution (360 g/L), water flux (source area × infiltration rate of 5.0 cm/yr), and initial mass of nitrate in a tank controls release for all contaminants (i.e., assumed proportionality). These releases are conservative because many of the releases are solubility-limited. (In the Hanford model, releases to the vadose zone from the tanks are controlled by the amounts of contaminants remaining in the tanks).

The other primary inputs required for modeling the TWRS using VAM2D include: the infiltration rate, porous media properties, constitutive relationships, and boundary conditions. The infiltration rate for Hanford is assumed to be 5 cm/yr but may vary between 0 and 10 cm/yr based on precipitation rates and vegetative cover (USDOE-RL 1996). The infiltration rate for the alternative cases will be impacted by any *in situ* filling or treatment and the Hanford surface barrier. For example, placement of the Hanford barrier is assumed to decrease the infiltration rate to 0.05 cm/yr. The Hanford barrier is assumed to lose some integrity after 1,000 years causing the infiltration rate to double throughout the remainder of the 10,000-yr simulation period.

Examples of the properties used for the porous materials represented in the Hanford fate and transport model are provided in Table 17. No cementitious materials were represented in the model because the grout option for *ex situ* treatment was abandoned and TWRS tank closure options (which will likely include grouting as an alternative) are being addressed under a separate NEPA study.

Sensitivity analyses were performed to help characterize the impacts of uncertainties in the alternatives, source term modeling, and flow and transport parameters on the risk results. A Monte Carlo analysis was performed to better characterize the impact of uncertainties on the exposures resulting from the exposure media concentrations. A representative set of the distributions used is provided in Table 18. The results indicated that the exposures were most sensitive to exposure duration and frequency and ingestion and/or inhalation rates.

The fate and transport modeling for the TWRS, which is intended to provide bounding exposure concentrations for risk estimation in the EIS, appears to be based on a combination of expected and bounding assumptions. For example, infiltration rate is a primary driver for contaminant release and migration. The infiltration rate of 5 cm/yr used in the model can vary between 0 and 10 cm/yr; however, no indication was given that a bounding infiltration rate was used[34] For the source term evaluation, a 99% recovery was assumed for tank retrieval operations in which 1% of the original contaminant levels would remain including highly soluble species. Furthermore, the release was assumed to be controlled by that of nitrate, which should likely produce highly conservative release estimates of many constituents. Thus it appears that the source term model provides bounding estimates and the infiltration rate may be closer to an expected value.

[33] Intruder risks for the areas associated with the TRWS were evaluated under a previous Hanford performance assessment and will not be discussed here (Rittmann 1994).
[34] The tank facilities are assumed to be maintained in their current condition during the institutional control period.

Table 17. Properties for Porous Materials Represented in the TWRS FEIS

Material	Type/ Area	Saturated hydraulic conductivity, Ks (m/day)	Saturated water content, θ_s	Residual water content, θ_r	Residual saturation, S_{wr}	Longitudinal Dispersivity, α_L (m)	Van Genuchten		
							α (1/m)	β	γ
Hanford Sandy Sequence	1E	4.330	0.420	0.023	0.055	0.500	19.43	1.868	0.465
Hanford Upper/Lower Gravels	2E	1.320	0.358	0.021	0.059	0.101	2.90	1.613	0.380
Ringold	3E	0.660	0.32	0.025	0.078	0.060	1.76	1.338	0.253
Hanford Formation	1W	10.36	0.30	0.001	0.0033	0.250	9.45	1.25	0.20
Early Palouse Soil	2W	1.42	0.39	0.056	0.14	0.150	0.90	2.09	0.52
Pliocene/Pleistocene Unit	3W	5.18	0.46	0.13	0.28	0.046	4.86	1.35	0.26
Ringold	4W	1.73	0.32	0.025	0.078	0.060	9.16	1.81	0.45

5.3.2.3 Sensitivity and Uncertainty Analysis Approach

Point-value estimates that were intended to be bounding for risk were used as the primary basis for comparison of remedial alternatives in the Hanford TWRS EIS. The approach to uncertainty analysis in the TWRS EIS was to first provide bounding estimates of risk to account for recognized uncertainties in the alternatives (resulting from assumptions concerning inventories, composition, and remedial actions) and risk analyses (using assumptions about source release, fate and transport, future land uses, etc). Sensitivity analyses were also conducted to help identify the impacts on predicted risks of uncertainties. The initial infiltration rate, partition coefficients, and performance period were highly influential on predicted exposure concentrations and risk.

A Monte Carlo analysis was performed to better characterize the impacts of uncertainties on the predicted exposures corresponding to exposure media concentrations. Probabilistic exposures were computed for the concentrations obtained from the VAM2D model using the Crystal Ball add-in to Microsoft Excel. Site-specific probability distributions were used when possible (Table 17). The results of the Monte Carlo analysis indicated that factors such as, exposure duration and frequency and intake factors such as, ingestion and inhalation were primary drivers for uncertainties in exposure. The results also indicated that exposure to given media concentration might be an order of magnitude higher than expected.

Finally, a nominal risk analysis was also performed based on expected values that helped characterize the impacts of the conservative assumptions used in the bounding case risk analyses in the EIS (USDOE-RL 1996). The impact of reducing the uncertainties in the bounding case tended to reduce resulting predicted risks, with reductions varying according to exposure scenario, remedial alternative, and time. Some nominal risks at certain times were found to be greater than the corresponding risks (i.e., not "conservative"); however, this result has more to do with shifting risks in time and not necessarily the magnitude of the risks.

The Hanford Tri-Party Agreement specified vitrification as the preferred treatment method for low-activity wastes at Hanford based on uncertainties associated with grouting. Thus the modeling performed to support the Hanford EIS did not include cementitious materials. Any impacts of these materials on potential remedial alternatives for the Hanford TWRS can only be made qualitatively and would depend on whether these materials were used to close tanks or treat retrieved wastes.

Grouting has been identified as the preferred method for closing high-level waste tanks at both the Savannah River and Hanford sites. *Ex situ* treatment (grouting) of retrieved Hanford tank waste was removed from consideration because of uncertainty that the grout could perform over the long time periods even though one may argue cementitious grout is a "reasonable" alternative for *ex situ* LAW treatment (under CEQ NEPA Regulations 40 CFR §1502.14 and §1505.2)[35, 36]. (However, cementitious grout has been used at SRS to treat low-activity tank waste for disposal in the onsite Saltstone facility.)

The CBP goal of providing more accurate predictions to be made when cementitious barriers are used in disposal could have a large impact in the future, safe and more economic treatment of retrieved wastes possibly including low-activity waste from Hanford.

5.4 Commerical Nuclear Power Facilities

5.4.1 Big Rock Point Decommissioning under the USNRC License Termination Rule and Environmental Assessment

The Big Rock Point Nuclear Power Plant is being decommissioned using a "Greenfield" approach (EPRI 2004)[37]. Before the plant was dismantled, the contaminated areas and components were decontaminated (Tompkins 2006). The spent fuel was removed to the spent fuel pool allowing dismantlement to begin including the spent fuel pool storage racks and liner. The reactor vessel was removed whole, placed in an approved transportation cask, grouted using a low-density cellular concrete, and transported to the Chem-Nuclear Systems, L.L.C., Barnwell, SC low-level waste disposal facility for disposal. The steam drum was removed and shipped by rail to the Envirocare facility in Utah. By April 2006, the containment sphere and turbine building were also demolished.

The company holding a reactor license must seek USNRC permission to decommission a facility including demonstration that the requirements of the License Termination Rule (LTR) (10 CFR §20.1401 *et seq.)* will be satisfied including meeting the 0.25 mSv/yr (25 mrem/yr) LTR dose limit for unrestricted use. The RESidual RADioactivity (RESRAD®) code (Yu et al. 2001) was used to perform the dose analyses needed to support the unrestricted release of the Big Rock Point site (BRPRP 2005; CEC 2004)[38]. Both point-value and probabilistic computations were performed using RESRAD® to support the development of Derived Concentration Guideline Levels (DCGLs) for the Final Status Survey. The probabilistic analyses were primarily used for parameter-sensitivity analysis to identify those parameters important to the assessment. This section provides a brief summary of the uncertainty approach adopted to demonstrate performance with the LTR.

5.4.1.1 Modeling Approach

The predicted doses from soils and groundwater from residual contamination at the Big Rock Point Nuclear Power Plant site were used to develop Derived Concentration Guideline Levels (DCGLs) for final site survey. These radionuclide concentration limits are the basis for evaluating the results of the final status survey for release of the site. A resident farmer scenario was used as the basis for assessing

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[35] The selection was justified based on a noted lack of sensitivity of the risk results to initial infiltration rate when a cap is installed (USDOE-RL 1996).

[36] *Ex situ* treatment of the Hanford LAW waste was not mentioned in the "Alternatives Considered but Dismissed" section of the TWRS EIS (USDOE-RL 1996).

[37] In a "Greenfield" approach, all structures including those below grade (e.g., foundations, basements, etc.) are demolished and disposed of off-site.

[38] However, because contaminated concrete and other building debris obtained after dismantling and demolition was shipped off-site for disposal, these cementitious materials were not considered in the dose modeling using RESRAD®. The only area where cementitious materials impacted the analyses to support decommissioning of the Big Rock Point facility is for the dose assessment for transportation of the reactor pressure vessel to the Barnwell low-level disposal facility.

dose using the RESidual RADioactivity (RESRAD®) code (Yu et al. 2001). RESRAD® is typically used to estimate doses and risks from residual radioactive materials and to calculate operational guidelines for soil contamination. The simple conceptual models that form the basis for the RESRAD® dose and risk analyses for residual contamination at the Big Rock Point Nuclear Power Plant site were illustrated in Figure 12 and Figure 13.

5.4.1.2 Parameter Assumptions and Distributions

In defining DCGLs, site-specific values were determined by direct measurement whenever possible. If a physical parameter value could not be determined by measurement, a value was derived using a probabilistic sensitivity analysis in RESRAD® as described in NUREG/CR-6697 (Yu et al. 2000). For high-priority parameters, distributions were assigned from NUREG/CR-6697 and a probabilistic sensitivity analysis was run using RESRAD®. Parameters were declared "sensitive" if the absolute value of the partial-ranked correlation coefficient (PRCC) was greater than 0.25 mSv based on total expected dose equivalent (TEDE) correlation. For sensitive parameters, a value of either the 75% quartile or the 25% quartile was selected based on whether the correlation was positive or negative, respectively. Nonsensitive parameters were assigned the 50% quartile value. Values were assigned to 55 of the hundreds of parameters used to define DCGLs for the release of the Big Rock Point site in this manner. Samples of the sensitivity results and assigned values are provided in Table 18. However, none of the parameters used in the RESRAD® model pertain to cementitious materials.

The only area where cementitious materials impacted the analyses to support decommissioning the Big Rock Point facility was in the dose assessment for transporting the reactor pressure vessel to the Chem-Nuclear Systems, L.L.C., Barnwell disposal facility. The pressure vessel was removed as a unit and placed in a new transportation cask, which was filled with a low density cellular concrete and welded shut. A series of dose calculations were performed using the Microshield and ISOSHLD-PC codes[39] to demonstrate that the cask complies with all of the 10 CFR 71 criteria for a Type B package (BNFL 2001). Point-value analyses were used as the primary basis for decision-making. Uncertainties in the analysis were managed by making conservative assumptions for the material properties and radionuclide inventory and distribution[40].

5.4.1.3 Sensitivity and Uncertainty Analysis Approach

The inputs to decision-making for decommissioning of the Big Rock Point Nuclear Power Plant site were derived from both point-value and probabilistic analyses. To determine Derived Concentration Guideline Levels (DCGLs) for the Final Status Survey for the site, a probabilistic sensitivity analysis was performed in RESRAD® to identify important physical parameters that were not measured and to assign them values. The values were then used in point-value calculations (using RESRAD®) to define DCGLs as the basis for unrestricted release of the site. Much of the site has since been released for unrestricted public use.

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[39] The most recent version of the Microshield code can be found at http://www.radiationsoftware.com/ (accessed March 20, 2009). The ISOSHLD code is described at http://www.nea.fr/abs/html/ccc-0079.html (accessed March 20, 2009). ISOSHLD can model complex geometries and thus provide more accurate dose rates than Microshield, which was used to verify the ISOSHLD output (BNFL 2001).

[40] For example, the Co-60 inventory, which is the primary driver of dose, is assigned a higher value from another pressure vessel. The annular region between the vessel and the transport cask steel shielding is assumed filled with low density cellular concrete with a minimum density of 800 kg/m_3 (50 lb/ft_3). The concrete in the vessel will have a minimum density of 480 kg/m_3 (30 lb/ft_3). Gamma dose rates are inversely proportional to the shield material density so the use of denser concrete would result in lower dose rates than those obtained in this assessment (BNFL 2001).

Table 18. Selected RESRAD® Sensitivity Analysis Distributions and Results for Big Rock Point DCGL Definition (adapted from CEC 2004)

Parameter	Priority[1]	Distribution	Distribution Parameters[2]				PRCC[3]	50% Quartile	25% or 75% Quartile	Assigned Parameter Value
			1	2	3	4				
Density (Saturated Zone – SZ)	1	Truncated normal	1.52	0.230	0.001	0.999	0.03	1.52	—	1.52
Total porosity (SZ)	1	Truncated normal	0.425	0.0867	0.001	0.999	-0.07	0.424	—	0.424
Effective porosity (SZ)	1	Truncated normal	0.355	0.0906	0.001	0.999	-0.09	0.355	—	0.355
Soil-specific b parameter (SZ)	2	Bounded lognormal-n	1.06	0.66	0.5	30	0.06	2.88	—	2.88
Root depth	1	Uniform	0.3	4.0	—	—	-0.48	—	1.22	1.22
Plant transfer factor for H	1	Truncated lognormal-n	1.57	1.1	0.001	0.999	-0.11	4.80	—	4.80
Mn	1	Truncated lognormal-n	-1.20	0.9	0.001	0.999	-0.01	0.299	—	0.299
Fe	1	Truncated lognormal-n	-6.91	0.9	0.001	0.999	-0.03	0.001	—	0.001
Co	1	Truncated lognormal-n	-2.53	0.9	0.001	0.999	-0.04	0.079	—	0.079
Sr	1	Truncated lognormal-n	-1.20	1.0	0.001	0.999	0.54	—	0.589	0.589
Cs	1	Truncated lognormal-n	-3.22	1.0	0.001	0.999	0.07	0.040	—	0.040
Eu	1	Truncated lognormal-n	-6.21	1.1	0.001	0.999	-0.09	0.002	—	0.002
Gd	1	Truncated lognormal-n	-6.21	1.1	0.001	0.999	0.11	0.002	—	0.002
Erosion rate (Contaminated Zone)	2	Continuous logarithmic	Default[4]				-0.09	0.001	—	0.001
Well-pump intake depth (below water table)	2	Triangular	6	30	10	—	0.03	14.5	—	14.5
Evapotranspiration coefficient	2	Uniform	0.5	0.75	—	—	0.05	0.624	—	0.624
Runoff coefficient	2	Uniform	0.1	0.8	—	—	0.00	0.449	—	0.449
Fruit, vegetable, and grain consumption rate	2	Triangular	135	318	178	—	-0.05	205	—	205
Aquatic food contaminated fraction	2	Triangular	0	1	0.39	—	-0.09	0.448	—	0.448
Soil ingestion rate	2	Triangular	0	36.5	18.3	—	0.06	18.2	—	18.2
Drinking water intake	2	Truncated lognormal-n	6.015	0.489	0.001	0.999	0.06	409	—	409
Depth of soil mixing layer	2	Triangular	0.0	0.6	0.15	—	-0.06	0.232	—	0.232
Wet weight crop yield (non-leafy plants)	2	Truncated lognormal-n	0.56	0.48	0.001	0.999	0.00	1.75	—	1.75
Weathering removal constant	2	Triangular	5.1	84	18	—	-0.05	32.8	—	32.8
Wet foliar interception fraction (leafy vegetables)	2	Triangular	0.06	0.95	0.67	—	-0.07	0.581	—	0.581
Meat transfer factor for H	2	Truncated lognormal-n	-4.42	1.0	0.001	0.999	0.13	0.012	—	0.012
Mn	2	Truncated lognormal-n	-6.91	0.7	—	—	0.03	0.001	—	0.001
Fe	2	Truncated lognormal-n	-3.51	0.4	—	—	0.04	0.030	—	0.030
Co	2	Truncated lognormal-n	-3.51	1.0	—	—	-0.12	0.030	—	0.030
Sr	2	Truncated lognormal-n	-4.61	0.4	—	—	0.03	0.010	—	0.010
Cs	2	Truncated lognormal-n	-3.00	0.4	—	—	0.01	0.050	—	0.050
Eu	2	Truncated lognormal-n	-6.21	1.0	—	—	-0.13	0.002	—	0.002
Gd	2	Truncated lognormal-n	-6.21	1.0	—	—	0.05	0.002	—	0.002

[1] 1 – high priority parameter or 2 – medium priority parameter based on four criteria: (1) relevance of the parameter in dose calculations, (2) variability of the radiation dose as a result of changes in the parameter value, (3) parameter type (physical, behavioral, or metabolic), and (4) availability of data in the literature (Yu et al. 2000)

[2] Parameters for distribution:

Lognormal-n: 1 – mean, 2 – standard deviation
Bounded lognormal-n: 1 – underlying mean value, 2 – underlying standard deviation, 3 – lower limit, 4 – upper limit
Truncated lognormal-n: 1 – underlying mean value, 2 – underlying standard deviation, 3 – lower quantile, 4 – upper quantile
Triangular: 1 – minimum, 2 – maximum, 3 – most likely
Uniform: 1 – minimum, 2 – maximum

[3] PRCC – Partial ranked correlation coefficient for peak all-pathways dose

[4] Default RESRAD v6.21 distribution parameters were used

Because of the "Greenfield" approach taken to decommissioning including removal of all contaminated cementitious materials for off-site disposal, the properties and performance of cementitious materials were not involved in the site release decision. However, the removal and transport of the reactor pressure vessel for disposal at the Barnwell low-level waste site involved dose modeling that took shielding credit for the cementitious materials used to fill the transport cask and pressure vessel for disposal. The approach to uncertainty in this case was to make assumptions for material properties and radionuclide inventory and distributions that assured "conservative" doses would be predicted. Therefore, different uncertainty approaches were used in different areas for the dose assessment modeling to support license termination and unrestricted release of the Big Rock Point Nuclear Power Plant.

5.4.2 Spent Fuel Pool Operations

5.4.2.1 Containment Performance for Spent Nuclear Fuel Pools

Commercial nuclear power reactors in the U.S. are of two basic types: boiling water or pressurized water reactors. The spent fuel pools tend to be located in different areas for the two reactor types. For boiling water reactors, pools tend to be located above ground near the reactor. Pools tend to be located in external structures on or partially embedded in the ground for pressurized water reactors. Regardless of reactor type or location, the storage pools must be constructed to USNRC requirements to protect the public against radiation exposure.

The decommissioning of the Big Rock Point nuclear facility (as described in Section 3.2.8) provides an example of how a spent nuclear fuel pool may be decommissioned as part of the overall strategy for the facility. In this case, the storage racks and pool liner were completely removed as part of the overall plan and the site was released by the NRC for unrestricted use under a "Greenfield" approach to decommissioning. Any small impacts and uncertainties, however large, due to the presence of contaminated materials could thus be ignored without significant consequence. However, it may also be possible to decommission a spent fuel pool separately from the remainder of the nuclear facility.

The Unit 1 Spent Fuel Pool at the Dresden Nuclear Power Station in Grundy County, Illinois was decommissioned using an innovative underwater coating technique developed by the Idaho National Laboratory (INL) for spent fuels pools on the Idaho Site (Demmer et al. 2006). Dresden Station Unit 1 was retired in 1978 and has been declared a Nuclear Historic Landmark[41]. Unit 1 is a boiling water reactor with a spent fuel pool in an area of the facility that makes a "Greenfield" approach to decommissioning the fuel pool impossible. The INL method was successfully used to decommission the Dresden Unit 1 Spent Fuel Pool. Because decommissioning of the Dresden Unit 1 Spent Fuel Pool involved the application of an epoxy-based coating to the walls and floor while underwater, there was no role to be played in the dose or hazard assessments for the cementitious materials comprising the storage pool.

From a cursory examination of the dose assessments that have been performed to support decommissioning activities for commercial power reactor spent nuclear fuel pools, it appears that including the cementitious components and the uncertainties in their properties and performance would not significantly impact the decisions made. However, when alternatives are considered that may leave contaminated cementitious materials onsite analogous to the

[41] Decontamination of the primary system was completed in 1984 and spent fuel and storage equipment were removed from the pool with the remainder of the decommissioning work until the other two operating units at the Dresden Station have reached the end of their licenses. See http://www.nrc.gov/info-finder/decommissioning/power-reactor/dresden-nuclear-power-station-unit-1.html (accessed March 20, 2009).

entombment activities at the Idaho and Hanford Sites (Section 3.2.3 and Section 3.2.6, respectively), the explicit and accurate consideration of cementitious materials may become critical factors in the decision-making process. This consideration must include an evaluation of the uncertainties of the properties and performance of the cementitious materials used.

5.4.3 Spent Fuel Pool Containment during Operations

Apart from decommissioning considerations, cementitious materials may also be considered when assessing the risks and doses posed to the general public from the reactor facility, and in this case, the spent fuel storage facilities. The two primary sources of potential exposures to the general public from a commercial nuclear facility are the reactor core and the spent nuclear fuel storage facility (e.g., dry cask or pool storage). Historically, the probabilistic risk assessments performed for commercial reactors have concentrated on loss-of-coolant accidents (LOCA) because these accidents have a higher probability and would result in the most catastrophic consequences (USNRC 1975).

However, probabilistic risk assessments for commercial nuclear reactors have considered the consequences of accidents involving the spent nuclear fuel storage pools (especially those involving a loss of water in the pool). Improvements in the ability to characterize the uncertainties in the structural and thermal properties of the cementitious materials (structural concrete) used will improve the transparency and acceptance of the assessment of these types of accidents events. However, the likelihood of these events are typically very low and thus the ability to more accurately assess the likelihood and magnitude of contaminant releases associated with the occurrence of an accident appears limited in affecting decisions concerning spent fuel pools. On the other hand, since releases and impacts to the general public from spent fuel storage pools in aging facilities may occur, periodic structural performance evaluations are necessary.

The ability to make more accurate predictions of the properties and performance of cementitious materials may help improve decisions made concerning spent fuel storage facilities.

6.0 SUMMARY OF MODELING APPROACHES

The cornerstones of the DOE authority to manage and regulate radioactive wastes are the Atomic Energy Act (AEA) and Nuclear Waste Policy Act (NWPA). However, these laws are not the sole applicable federal statutes (NAS 2006). Additional legislation including CERCLA, RCRA, and the NEPA and correlative state and local laws may also play important roles. The relevant considerations under these additional statutes often go well beyond and adopt different practices than the AEA or NWPA, and more importantly are not administered by the DOE but instead by the EPA and the states (NAS 2006). Whereas performance assessments are required under DOE 435.1 and the AEA, the other laws require different sorts of assessments, which although are often similar to PAs in basic structure, are termed PA-like in this report. Because the License Termination Rule (LTR; 10 CFR Part 20 Subpart E), which is administered by the NRC, also does not require a performance assessment, this law was examined in this chapter.

Although none of the laws referred to in this chapter have requirements for how uncertainty analyses should be performed, it has been recognized that the analysis of uncertainty is a necessary additional dimension of risk. There are different ways to analyze uncertainties.

For example, the typical DOE practice when performing assessments to support CERCLA and RCRA cleanup activities has been to base decisions on bounding estimates of concentrations and risks supported by limited sensitivity analyses based on recognized uncertainties. Performance assessments have been also performed in a manner similar to the approach for DOE CERCLA and RCRA processes. However, typically there is more detail associated with the modeling and, more recently, greater use of probabilistic techniques either individually or in conjunction with deterministic approaches to characterize uncertainties in a more comprehensive manner. The requirements for managing uncertainties for those laws that do not require a formal performance assessment are summarized in Table 19. Note that no

Table 19. Summary of Uncertainty Requirements in Regulations Requiring Other Types of Risk Assessments

Regulation	Uncertainty-Related Requirements	Guidance for Cementitious Barriers and Uncertainty	Frequency of Modeling Cementitious Barriers
Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA)	No specific requirements or recommendations in CERCLA. USEPA guidance for baseline and other assessments contains general suggestions but not specific methodology.	No specific requirements or recommendations in the law. Credit may be taken per guidance documents but resulting model uncertainties must be accounted for in decision-making process.	Remedial actions using cementitious barriers frequently modeled, but much less frequently selected for action. Often not included in the risk or uncertainty analysis.
Resource Conservation and Recovery Act (RCRA) (Subtitle C)	No specific requirements or recommendations in RCRA. USEPA guidance for CERCLA baseline and other assessments often used and contains general suggestions but not specific methodology. At USDOE sites, CERCLA and RCRA often integrated.	No specific requirements or recommendations in the law. Credit may be taken per guidance documents but resulting model uncertainties must be accounted for in decision-making process.	Often not included in the risk or uncertainty analysis or as defense-in-depth. If included, bounding or conservative assumptions are often made to account for uncertainties from lack of property and performance data.
National Environmental Policy Act (NEPA)	No specific requirements in NEPA. Uncertainty approach is at discretion of the lead agency although risks are often meant to be bounding. US Office of Management and Budget proposed uncertainty characterized for major findings and sensitivity analyses. At USDOE sites, often NEPA values are integrated into CERCLA/RCRA process.	No specific requirements or recommendations. Requires all "reasonable" alternatives be considered for EIS including those involving cementitious materials. Like other uncertainties, approach is at the discretion of the lead agency.	Review of EISs from SRS, Hanford, and Idaho did not reveal trend although approaches provided bounding risks using bounding assumptions including those for cementitious materials. Probabilistic techniques rarely used because of lack of property/performance data.

Table 19. Summary of Uncertainty Requirements in Regulations Requiring Other Types of Risk Assessments (2004) (contd)

Regulation	Uncertainty-Related Requirements	Guidance for Cementitious Barriers and Uncertainty	Frequency of Modeling Cementitious Barriers
License Termination Rule (10 CFR Part 20 Subpart E)	No legal requirements for uncertainty analysis. USNRC guidance requires discussion of the effect of uncertainties on dose results. Also discusses use of uncertainty/ sensitivity analyses to focus on important parameters.	No specific requirements or recommendations for cementitious barriers although there are requirements for engineered barriers including uncertainties in design and functionality especially those that have to perform for very long times. For complex sites involving long-lived radionuclides, a probabilistic analysis is suggested.	Cementitious materials are likely considered in every case in either contaminated concrete disposal, assessing residual contamination, reactor components disposal, etc. Consideration of uncertainties (especially those for cementitious materials) reduced by using a "Greenfield" approach to decommissioning.

distinction is made for handling uncertainties associated with cementitious barriers as opposed to other aspects of the analysis.

The assessment and uncertainty analysis methods were evaluated and summarized for four regulations that do not require formal performance assessments to assess risks and doses for with waste disposal activities at USDOE and other facilities that produce, store, and manage radioactive and hazardous wastes. Because several laws (including CERCLA, RCRA, and NEPA) may be applicable to the same contaminated site, policies have been adopted on the USDOE Complex level as well as the operating site level for integrating these laws and their assessments (Cook 2002; Shedrow, Gaughan & Moore-Shedrow 1993; DOE 1994c). Because of the integrated nature of these assessments, consistent guidance has been developed by the USEPA to manage uncertainties in the assessments. There are no specific requirements in CERCLA, RCRA, or NEPA for uncertainty analysis methods. The USEPA guidance provides a tiered, iterative framework for uncertainty analysis.

For commercial nuclear facilities liensed by the USNRC, the performance of NEPA environmental assessments and impact statements are part of the decommissioning process and demonstration of compliance with the LTR. The uncertainty analyses supporting the NEPA process may follow the typical "bounding assessment supported by limited sensitivity analysis" framework often followed by the DOE or they may be probabilistic in nature following the customary practice for commercial nuclear reactors including those assessments to support license termination.

For the three laws administered by the EPA, there are no legal requirements regarding the approaches that must be used for assessments or uncertainty analyses when cementitious barriers are present. NEPA requires that all "reasonable" alternatives be considered during the Environmental Impact Statement (EIS) process (including those using cementitious materials). However, the EIS process tends to focus on bounding risks estimates which are often supported by sensitivity analyses.

Demonstration of compliance with the NRC LTR requires a dose assessment for either *unrestricted* release (i.e., dose < 0.25 mSv/yr per 10 CFR §20.1402) or for *restricted* release when meeting certain conditions (10 CFR §20.1403(a)-(e)). Although there are no specific requirements for cementitious materials when performing the LTR dose assessment and uncertainty to determine site release characteristics, there are requirements for engineered barriers (often involving cementitious materials) that include consideration of uncertainties in the design and functionality of the barriers especially those that have to perform for very long times. For complex sites involving long-lived radionuclides, a probabilistic analysis of uncertainties is suggested.

6.1 Comparison of Examples

A typical analysis common to DOE Order 435.1, 10 CFR Part 61, IAEA, CERCLA, RCRA, NEPA, and the LTR can be conceptualized as an exposure assessment over various pathways from which either the dose or risk to a critical receptor (or receptors) is estimated with some degree of uncertainty. Because the conversions from exposure or intake dose to response (e.g., cancer risk, total effective dose equivalent, etc.) are determined by regulatory fiat, the uncertainties for an estimated dose or risk are actually associated with the exposures themselves. Assumptions made to model exposure will introduce uncertainties as will the uncertain input parameters used in the exposure model including the source term and release characteristics, fate and transport, and exposure scenario factors for selected receptors (e.g., resident, intruder, etc.). It is interesting to note that as shown in Table 1, the USEPA tends to focus on sampling uncertainties, parameters such as intakes and bioavailability, and chemical toxicity uncertainties rather than uncertainties associated with modeling engineered features and the natural environment. This reflects a more typical focus on exposure and toxicity assessment rather than fate and transport. Thus, for cementitious materials, uncertainties from assumptions and input parameters for the source term and release and near field transport will likely be important and have traditionally been considered in more detail in PAs conducted for LLW disposal as opposed to PA-like analyses. The key assumptions are summarized in Table 20.

For applications incorporating cementitious materials, key assumptions that introduce uncertainties tend to be related to the physical and chemical aspects of the source release and near field transport. In the examples provided in the previous chapter, the credit taken for cementitious materials ranged from none for the RWMC CERCLA assessment and the PA for the Nevada Test Site to considerable for the detailed assessment performed for the Idaho and Savannah Rive Site Tank Closure PA and for the Idaho RCRA Landfill Closure of the Waste Calcining Facility. Notably, even for the cases that took credit for cementitious materials, these materials were often represented in the models with gross conservatisms to provide bounding estimates. Other examples took various levels of credit for cementitious materials in modeling.

The examples provided demonstrate how assessments and uncertainty analyses have been performed to support decision-making for contaminated sites at DOE and other facilities. The assessments vary in terms of source and release assumptions, transport pathways modeled, exposure scenarios, and whether dose orrisk limits are mandated. These different assumptions and models used to predict risk result in varying levels of uncertainty in the endpoints for decision-making. It is likely that credit taken for cementitious materials in the modeling performed to support the assessments will typically impact the source term and release and near field transport. Of the various approaches represented in this chapter, those involving a graded or

Table 20. Summary of Examples of Uncertainty Analysis

Example	Description	Modeling Approach	Parameter Assumptions and Distributions	Sensitivity and Uncertainty Approach
Area 5 Radioactive Waste Management Site (Nevada)	Low-level waste disposal facility managed under DOE Order 435.1	Graded and iterative approach to modeling with no groundwater pathway. Surface processes addressed in a probabilistic manner using the GoldSim modeling platform. Probabilistic results were used for the compliance calculations. No credit for cementitious materials.	Detailed approach used to develop input parameter distributions including the use of expert elicitation. Distributions were developed for many of the input parameters, including the timing for intrusion into the site and loss of institutional memory of the site.	A combination of deterministic point value assessments and a probabilistic model was used. The deterministic runs were used during development of the models for intercomparisons and benchmarking. Latin Hypercube sampling and Monte Carlo simulation were used. Deterministic and probabilistic sensitivity cases were considered.
Radioactive Waste Management Complex Active Disposal Facility (Idaho)	Low-level waste disposal facility managed under DOE Order 435.1	Graded and iterative approach using multiple screening steps and a combination of deterministic and probabilistic calculations for the sensitivity and uncertainty analysis. Parallel groundwater modeling with TETRAD for detailed calculations and MCM and GWSCREEN for the probabilistic calculations. Deterministic calculations used for final decision. Cementitious materials considered in a limited manner.	Distributions were developed for fifteen parameters deemed important for the assessment, such as inventory/source term, infiltration, aquifer velocity and dispersivity, and geochemistry for key radionuclides. The distributions were based on a combination of site-specific data, literature reviews and expert judgment.	A combination of deterministic and probabilistic Monte Carlo calculations were used in a hybrid manner. Point values sensitivity cases were conducted to illustrate key assumptions, generally to highlight conservatism built into the model. Probabilistic calculations were conducted to illustrate the range of potential results and a comparison of deterministic results within that range. Sensitivity analysis was conducted using the probabilistic results for different times to identify key inputs for the different peaks in the results.

Table 20. Summary of Examples of Uncertainty Analysis (contd)

Example	Description	Modeling Approach	Parameter Assumptions and Distributions	Sensitivity and Uncertainty Approach
F Tank Farm (Savannah River Site)	Tank Closure under Section 3116	Graded and iterative approach, using screening followed by a combination of detailed deterministic assessments using HELP and PORFLOW and a probabilistic assessment using GoldSim in parallel for the sensitivity and uncertainty analysis. Deterministic calculations used as basis for decisions. Cementitious materials considered in significant detail.	Input distributions were developed for many different inputs into the model, including contaminant inventories, physical properties of barriers and the natural environment, geohydrology, geochemistry, failure scenarios and exposure assumptions. The distributions were developed using site-specific data, targeted research activities, literature reviews, and expert judgment.	A combination of deterministic and probabilistic calculations were used in a hybrid manner. Point values sensitivity cases were conducted to illustrate key assumptions and for benchmarking. Probabilistic calculations were conducted to illustrate the range of potential results. Sensitivity analysis was conducted using the probabilistic results for different times and well locations to identify key inputs for the different peaks in the results.
Integrated Disposal Facility (Hanford)	Low-level and mixed low-level waste disposal cells managed under DOE Order 435.1 and RCRA, respectively	Graded and iterative approach, using screening followed by deterministic analyses using STORM, VAM3DF, and CFEST for the source term and engineered features. An analytical model was also used for selected source term modeling. For the cementitious waste form, diffusion was assumed to control migration.	Distributions were not developed for any parameters in the deterministic approach, but detailed data packages were developed to document the basis for the parameters that were used. Ranges of values were specified for the parameters considered in the sensitivity cases. Emphasis was placed on developing the technical justification for realistic and defensible values in lieu of a probabilistic uncertainty analysis.	Deterministic point value sensitivity cases were used to illustrate the influence of changes in key parameters on the results. A variety of different parameters were considered, including geochemistry, diffusion coefficients for cement-based waste form, infiltration rates, and different well pumping rate scenarios.

Table 20. Summary of Examples of Uncertainty Analysis (contd)

Example	Description	Modeling Approach	Parameter Assumptions and Distributions	Sensitivity and Uncertainty Approach
Engineering Test Reactor (Idaho Site)	Decommissioning under a non-time-critical CERCLA removal action. ETR reactor vessel removed and disposed on-site	Graded approach to groundwater modeling using 1) NCRP method to screen for COPCs, 2) GWSCREEN to identify COCs, and 3) detailed GWSCREEN to better characterize impacts. Separate analysis examined whether reactor vessel could be left in-place or disposed using standard USEPA calculations.	Only point-value screening analyses performed. For groundwater analysis, simple models and bounding parameter values used first to define COPCs and less conservative models and values used in second and third phases. Cementitious materials not considered.	Only point-value, bounding doses and risks estimated for screening analyses. Approach was to "err" on the conservative side to likely over predict actual risks. Graded approach for groundwater pathway. Cementitious materials not considered because bounding risks were less than USEPA action limit.
Radioactive Waste Management Complex (Idaho Site)	Closure under the CERCLA remedial investigation/feasibility study (RI/FS) process	Modular approach to exposure and risk modeling. Complex, individual models are linked to estimate risks. WILD provides inventories, DUST-MS models source release, TETRAD models fate and transport. Standard USEPA methods for exposure to risk.	Only point-value exposures and risks estimated due to complexity of site and models. No credit was taken for cementitious materials affecting sources (concrete vaults or waste forms) or treatment. Cementitious waste forms were treated as soil.	Point-value exposures and risks estimated supported by one-at-a-time sensitivity analyses for inventory, infiltration, and subsurface. Credit for cementitious materials in risk and uncertainty anslysis unlikely to change conclusion that site poses unacceptable risk but might impact COCs.

Table 20. Summary of Examples of Uncertainty Analysis (contd)

Example	Description	Modeling Approach	Parameter Assumptions and Distributions	Sensitivity and Uncertainty Approach
Waste Calcining Facility (Idaho Site)	Landfill closure under RCRA supported by NEPA Environmental Assessment (EA)	Model based on conservative assumptions to provide bounding residual levels. Graded approach using GWSCREEN, RESRAD, and PORFLOW to identify and refine COCs. Detailed PORFLOW screening model took credit for cementitious materials including cracking.	Exposure parameters for receptors same for all phases. The simple GWSCREEN groundwater used general information, whereas, the PORFLOW model used site-specific hydraulic transport parameters and a simple cracking model because of importance of this process.	Point-value exposures and risks estimated. Approach was to "err" on the conservative side to likely over predict actual risks. Graded approach for groundwater pathway. More accurate models for cementitious properties and cracking not considered because bounding risks were less than USEPA action limit.
Tanks 17-F and 20-F (Savannah River Site)	Operational closure under SCDHEC industrial wastewater permits supported by NEPA Environmental Impact Statement (EIS)	Relatively simple release and saturated zone transport model using MEPAS to estimate concentrations, doses, and lifetime cancer risks for radioactive and hazardous contaminants.	Conservative estimate of inventory. Transport model is K_d-based although site-specific values used. Impact of REDOX on K_d's included. Instantaneous failure at 1,000 years increasing basemat hydraulic conductivity and infiltration rate.	Point-value doses and risks supported by one-at-a-time sensitivity analyses for inventory, K_ds, hydraulic properties, etc. Even bounding results indicated no exceedances for 10,000 yrs. Additional credit for cementitious materials could have only provided additional assurance.
P Reactor (Savannah River Site)	*In-Situ* Decommissioning under CERCLA	Simple GoldSim model for reactor portion of facility whose results summed with those from models for other parts of facility to give comprehensive risk. Reactor modeled a 1D system of five materials.	Six different materials were modeled with site-specific probabilistic distributions for soil and cementitious materials and others taken from literature. Vadose zone not modeled.	Point-value analyses based on best-estimate inputs supported by one-at-a-time sensitivity analyses were used for decision-making. A Monte Carlo simulation was performed and indicated that steel corrosion rate was most important.

Table 20. Summary of Examples of Uncertainty Analysis (contd)

Example	Description	Modeling Approach	Parameter Assumptions and Distributions	Sensitivity and Uncertainty Approach
221-U Facility (Hanford Site)	CERCLA RI/FS process used to evaluate potential actions and identify preferred alternatives supported by inclusion of NEPA values in process	Conceptual site model linking sources to receptors implemented in RESRAD for external exposure, ingestion, and inhalation for industrial use and groundwater protection scenarios. HSRAM used to evaluate noncarcinogenic impacts.	Maximum baseline risks estimated using bounding input parameter values intended to bound risk predictions. Some default RESRAD parameters used without sensitivity analysis. Only soil ingestion and drinking water intake values changed in scenarios.	Point-value predictions based on "conservative" inputs used to err on high-risk side. Only uncertainties in inventory considered. Uncertainties in other parameters not considered because protectiveness derived from cap. Cementitious materials provide "defense-in-depth."
Tank Waste Remediation System (Hanford Site)	NEPA EIS needed because of potential environmental impacts for proposed actions concerning the management and disposal of Hanford tank wastes	Groundwater impacts for 10 complex remedial scenarios and numerous receptors modeled using VAM2D. Groundwater flow evaluated under steady-state conditions. Cementitious materials not modeled because removed as alternatives per Tri-Party Agreement.	99% recovery assumed for retrieval scenarios. 1% (including water soluble species assumed left in tank to be conservative). *Ex situ* treatments have varying efficiencies. Releases from tanks begin at end of IC period using conservative congruent dissolution model. Cap assumed to lose integrity at 1,000 yrs. Probabilistic distributions developed for exposure factors.	Point-values were primary basis for decision. Bounding values used to represent uncertainties in alternatives and risk factors. Sensitivity analyses used to characterize impacts of uncertainties with infiltration rate and K_ds as important. Monte Carlo analysis performed and showed predicted exposures might be 10x high. Nominal analysis values indicated that risks are overpredicted (and shifted in time). Better modeling of cementitious materials might make them attractive for TWRS actions.

Table 20. Summary of Examples of Uncertainty Analysis (contd)

Example	Description	Modeling Approach	Parameter Assumptions and Distributions	Sensitivity and Uncertainty Approach
Big Rock Point Nuclear Power Plant	Decommissioned using a "Greenfield" approach under a license termination plan and demonstrating compliance with License Termination Rule supported by NEPA EA.	RESRAD model used to predict doses from soils and groundwater from residual contamination to define DCGLs for final survey. The DCGLs were used for the unrestricted release of site.	Parameters in RESRAD based on measurements when possible. Probabilistic sensitivity analysis in RESRAD used to define other parameters. No parameters used to develop DCGLs pertain to cementitious materials.	The inputs used to define DCGLs used based on both measurements and parameter sensitivity analyses. Point-value estimates performed using these parameters were the basis for defining DCGLs for unrestricted release.

COC – Contaminant of Concern
DCGL – Derived Concentration Guideline Levels
HSRAM – Hanford Site Risk Assessment Methodology (USDOE-RL 1995)
MCM – Mixing Cell Model (Rood 2005)
MEPAS – Multimedia Environmental Pollutant Assessment System (Strenge & Chamberlain 1995)
NCRP – National Council on Radiation Protection and Measurements
RESRAD – RESidual RADioactivity (Yu et al. 2001)
VAM2D – Variably Saturated Analysis Model in Two Dimensions (Huyakorn, Kool & Robertson 1989)
WILD – Waste Inventory and Location Database (McKenzie et al. 2005)

tiered iterative approach to both PA, risk assessment and uncertainty analysis are consistent with CERCLA guidance (USEPA 1989). Such an approach is similar to the basic remmendations provided by the USDOE, USNRC, IAEA, and NCRP (Brown 2008).

One consistent theme running through the various dose and risk assessments performed in the example cases was that often gross simplifying assumptions were made when cementitious materials were considered in the assessment process, especially for physical performance. These assumptions were typically needed because of lack of material property data and information and/or a lack of willingness or need to make the effort to defend the assumptions.

Often physical performance is only important for a short, easily defended time frame, so there is not a significant need to take additional credit. For example, in the Idaho RWMC CERCLA assessment (Section 3.2.2), the cement-based waste forms were treated like soil in the source term model because specific waste form to water partitioning coefficients were not available. In the end, it did not impact the decisions, although it did perpetuate over-conservatisms that could impact future decisions. This uncertainty due to the "unknowable" is very difficult to manage in an assessment and often the decision is made to ignore the cementitious materials, which allows little or no credit for their participation in the disposition process and can significantly bias an alternatives analysis.

The ability to provide data and more accurate models for the cementitious materials used in nuclear application offers the potential for more sensible credit to be taken for these materials. One reason that vitrification was selected for immobilization of low-activity wastes (LAW) at the Hanford Site was the relative durability and certainty of glass waste forms when compared to cementitious forms. Cementitious waste forms may actually be adequate for Hanford LAW ; however, the extensive work performed on vitrified high-level waste forms provided the certainty needed for stakeholders to rely on these waste forms for both Hanford HLW and LAW. One goal of the CBP is to provide needed data and more accurate models for cementitious materials used in nuclear application to ultimately support this type of assurance for future applications of cementitious materials.

Because cementitious barriers/materials function as diffusion barriers to contaminant releases, cracking is critical because it alters the mechanistic transport of water and vapor through the material (increasing the potential for leaching)[42]. Failure due to cracking was modeled for F-Tank Farm PA, the of the Idaho Waste Calcining Facility landfill closure[43] and the operational closures for the 17-F and 20-F Tanks at SRS. The manner in which cracking was introduced into the SRS tank closure assessments was fairly typical. The grout and concerte were assumed to remain intact for a given, long period of time (i.e., 1,000 years) with low hydraulic conductivities. They were then assumed to fail instantaneously and completely resulting in a material with several order of greater hydraulic conductivities. These assumptions have significant impacts on the release properties for the materials. The F-Tank Farm PA (Section 3.1.3) involved a more rigorous assessment of the timing of failure of the cementitious barriers and included distributions of failure times considering both chemical and physical aspects of the barrier. Nevertheless, the assumptions about changes in permeability due to cracks were simplified.

Uncertainties and temporal degradation and other effects on physical and chemical properties for the cementitious materials appear to be rarely taken into account even if they can have significant impacts on the endpoint predictions used to characterize doses and risks for decision-making purposes. Longer-term credit is generally taken for chemical performance than for physical performance and can have a significant influence on performance for long-lived radionuclides. Improvements in the characterization and modeling of the phenomena and properties related to the cementitious materials used in disposal will provide more accurate predictions and support continued use in future disposal activities.

One goal of the CBP is to provide more accurate models for cementitious materials used in nuclear application to ultimately provide this type of assurance for future applications of cementitious materials.

7.0 CONCLUSIONS AND NEEDS

Cementitious materials have been used in numerous waste management applications (e.g., waste processing, soil and groundwater, and decommissioning) regulated under various federal regulations including DOE, IAEA and NRC requirements related to waste disposal and CERCLA, RCRA, and NEPA, which are administered by the USEPA. Nuclear reactor and licensed material facilities have been decommissioned under the License Termination Rule (10 CFR Part 20 Subpart E). Unlike assessment processes regulated under the Atomic Energy Act (AEA) and USDOE 435.1, the risk and dose assessments performed under the laws administered by EPA and the LTR do not require performance assessments, but do require deration of long-term performance. Although, there may be different goals and frameworks for these different applications, there are many similarities and experiences that can be shared. There is a critical

[42] For example, Walton (1992) concluded that cracking is the “Achilles heel” of cementitious barrier performance. Furthermore, high quality concrete (without cracks) will typically perform acceptably well in the isolation of contaminants because of its “low permeability and high available surface area for sorption.” When cracked, concrete cannot be relied upon for contaminant isolation.

[43] This assumption was assumed to be conservative under RCRA closure requirements (Demmer et al. 1999).

need to create a means to share information regarding the lessons learned and good practices associated with all these different applications and to identify specific aspects that may be beneficial from one application to the next.

When considering PA-like assessments for applications outside of the radioactive waste disposal realm, cementitious barriers have traditionally not been considered or been considered in a simplified manner. Furthermore, there is typically minimal guidance related to treatment of cementitious barriers in any of the regulations and associated guidance and especially related to uncertainty analysis. There are more guidance documents beginning to be developed, primarily by the USNRC that address cementitious materials and sensitivity and uncertainty analysis and recommendations have also been published by the NCRP. There is an additional challenge associated with moving towards probabilistic approaches. If more detailed models are developed, there will be a need to be able to represent them in a manner that is amenable to probabilistic analysis (i.e., may be a need for an abstracted or simplified manner suitable for hundreds or thousands of realizations in a probabilistic model).

A significant area of need is to update existing guidance to account for the latest developments in modeling of cementitious materials and to make that guidance useful across the spectrum of different types of assessments that are being conducted. Guidance on the development of distributions for key parameters to be considered in an uncertainty analysis is also needed as this is an area that is routinely subject to comments from reviewers (Seitz et al. 2008).

With the variety of applications taking advantage of cementitious materials continually increasing, a larger population of modelers is getting involved in assessments. The lack of taking credit for cementitious barriers can often be the result of a lack of awareness of information regarding the properties and performance of these materials for the specific conditions under analysis. This highlights a need for improved sharing of information regarding models and data that are needed to assess the performance of cementitious barriers. This applies as well to the sharing of approaches that are being used for development of input distributions and modeling approaches that are needed to properly conduct probabilistic sensitivity and uncertainty analyses. In this specific area, there are lessons that can be learned from the deep geologic repository programs, where there is substantial experience in applying probabilistic approaches.

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