OVERVIEW OF THE U.S. DEPARTMENT OF ENERGY AND NUCLEAR REGULATORY COMMISSION PERFORMANCE ASSESSMENT APPROACHES

Cementitious Barriers Partnership

May 2009

CBP-TR-2009-001, Rev. 0

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and

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An electronic copy of this document is also available through links on the following websites: http://srnl.doe.gov/ and http://cementbarriers.org/

FOREWORD

The Cementitious Barriers Partnership (CBP) is a multi-disciplinary, multi-institutional collaboration sponsored by the United States Department of Energy (US DOE) Office of Waste Processing. The objective of the CBP project is to develop a set of tools to improve understanding and prediction of the long-term structural, hydraulic, and chemical performance of cementitious barriers used in nuclear applications.

A multi-disciplinary partnership of federal, academic, private sector, and international expertise has been formed to accomplish the project objective. In addition to the US DOE, the CBP partners are the United States Nuclear Regulatory Commission (NRC), the National Institute of Standards and Technology (NIST), the Savannah River National Laboratory (SRNL), Vanderbilt University (VU) / Consortium for Risk Evaluation with Stakeholder Participation (CRESP), Energy Research Center of the Netherlands (ECN), and SIMCO Technologies, Inc.

The periods of cementitious performance being evaluated are >100 years for operating facilities and > 1000 years for waste management. The set of simulation tools and data developed under this project will be used to evaluate and predict the behavior of cementitious barriers used in near-surface engineered waste disposal systems, e.g., waste forms, containment structures, entombments, and environmental remediation, including decontamination and decommissioning (D&D) activities. The simulation tools also will support analysis of structural concrete components of nuclear facilities (spent-fuel pools, dry spent-fuel storage units, and recycling facilities such as fuel fabrication, separations processes). Simulation parameters will be obtained from prior literature and will be experimentally measured under this project, as necessary, to demonstrate application of the simulation tools for three prototype applications (waste form in concrete vault, high-level waste tank grouting, and spent-fuel pool). Test methods and data needs to support use of the simulation tools for future applications will be defined.

The CBP project is a five-year effort focused on reducing the uncertainties of current methodologies for assessing cementitious barrier performance and increasing the consistency and transparency of the assessment process. The results of this project will enable improved risk-informed, performance-based decision-making and support several of the strategic initiatives in the DOE Office of Environmental Management Engineering & Technology Roadmap. Those strategic initiatives include 1) enhanced tank closure processes; 2) enhanced stabilization technologies; 3) advanced predictive capabilities; 4) enhanced remediation methods; 5) adapted technologies for site-specific and complex-wide D&D applications; 6) improved SNF storage, stabilization and disposal preparation; 7) enhanced storage, monitoring and stabilization systems; and 8) enhanced long-term performance evaluation and monitoring.

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1.0 INTRODUCTION

Engineered barriers including cementitious barriers are used at sites disposing or contaminated with low-level radioactive waste to enhance performance of the natural environment with respect to controlling the potential spread of contaminants. Drivers for using cementitious barriers include: high radionuclide inventory, radionuclide characteristics (e.g., long half-live, high mobility due to chemical form / speciation, waste matrix properties, shallow water table, and humid climate that provides water for leaching the waste).

This document comprises the first in a series of reports being prepared for the Cementitious Barriers Partnership. The document is divided into two parts which provide a summary of: 1) existing experience in the assessment of performance of cementitious materials used for radioactive waste management and disposal and 2) sensitivity and uncertainty analysis approaches that have been applied for assessments. Each chapter is organized into five parts: Introduction, Regulatory Considerations, Specific Examples, Summary of Modeling Approaches and Conclusions and Needs.

The objective of the report is to provide perspective on the state of the practice for conducting assessments for facilities involving cementitious barriers and to identify opportunities for improvements to the existing approaches. Examples are provided in two contexts: (1) performance assessments conducted for waste disposal facilities and (2) performance assessment-like analyses (e.g., risk assessments) conducted under other regulatory regimes.

The introductory sections of each section provide a perspective on the purpose of performance assessments and different roles of cementitious materials for radioactive waste management. Significant experience with assessments of cementitious materials associated with radioactive waste disposal concepts exists in the US Department of Energy Complex and the commercial nuclear sector. Recently, the desire to close legacy facilities has created a need to assess the behavior of cementitious materials for applications in environmental remediation and decontamination and decommissioning (D&D) applications. The ability to assess the use and benefits of cementitious materials for these applications can significantly affect decisions related to cleanup activities. For example the need for costly remedial actions may not be necessary if existing or new cementitious barriers were adequately represented.

The sections dealing with regulatory considerations include summaries of the different regulations that are relevant for various applications involving cementitious materials. A summary of regulatory guidance and/or policies pertaining to performance assessment of cementitious materials and sensitivity and uncertainty analyses is also provided in the following chapters.

Numerous examples of specific applications are provided in each report. The examples are organized into traditional waste disposal applications (performance assessments), applications related to environmental remediation and D&D, and reactor and spent fuel related assessments.

Sections that discuss specific facilities or sites contain: (1) descriptions of the role of the cementitious barriers or sensitivity/uncertainty analysis, (2) parameter assumptions and conceptual models, and (3) a relative discussion of the significance in the context of the assessment. Examples from both the U.S. Department of Energy Sites and the U.S. Nuclear Regulatory Commission are provided to illustrate the variety of applications and approaches that have been used.

In many cases, minimal credit was taken for cementitious barriers. However, in some of those cases, benefits of being able to take credit for barriers were identified. The examples included: (1) disposal facilities (vaults, trenches, tank closures, cementitious waste forms and containers, etc.), (2) environmental remediation (old disposal facilities), (3) reactor and large structure decommissioning, and (4) spent fuel pools. These examples were selected to provide a perspective on the various needs, capabilities to model cementitious barriers, and use of sensitivity and uncertainty analysis and were not intended to include all cementitious barriers used in all low-level waste related PAs.

The summary section in each report/chapter provides an overview of important considerations for the examples and compares and contrasts the different approaches that have been used. For example, specific time dependent physical processes (changes in hydraulic conductivity) and chemical processes (partitioning coefficients, and solubility coefficients) are identified and compared. The summary section also identifies key needs for future assessments.

The Cementitious Barriers Partnership was established to address the key needs related to the use of cementitious barriers (waste forms, containment structures, physical stabilization fill materials. These needs are identified in the conclusions sections of each report/chapter and include:

- Improved information and technology transfer sharing of information (e.g., data, models) relevant for designing and assessing cementitious barriers encountered in radioactive waste disposals.
- Accepted approaches and guidance for developing input parameter distributions and for performing sensitivity and uncertainty analysis for concepts and scenarios involving cementitious barriers
- Updated guidance on approaches for assessing as-built cementitious barriers including sensitivity and uncertainty analysis
- Improved representation of temporal changes in the physical properties of cementitious materials and barrier structures to water and gas flow
- Improved coupling of the multi physics phenomena and processes that affect the physical and chemical evolution of cementitious materials over long and short time frames.

2.0 BACKGROUND

Performance assessments (PAs) of radioactive waste disposal systems are iterative processes involving site specific, prospective modeling evaluations of the post closure time phase (see Figure 1). Performance assessments have two primary objectives:

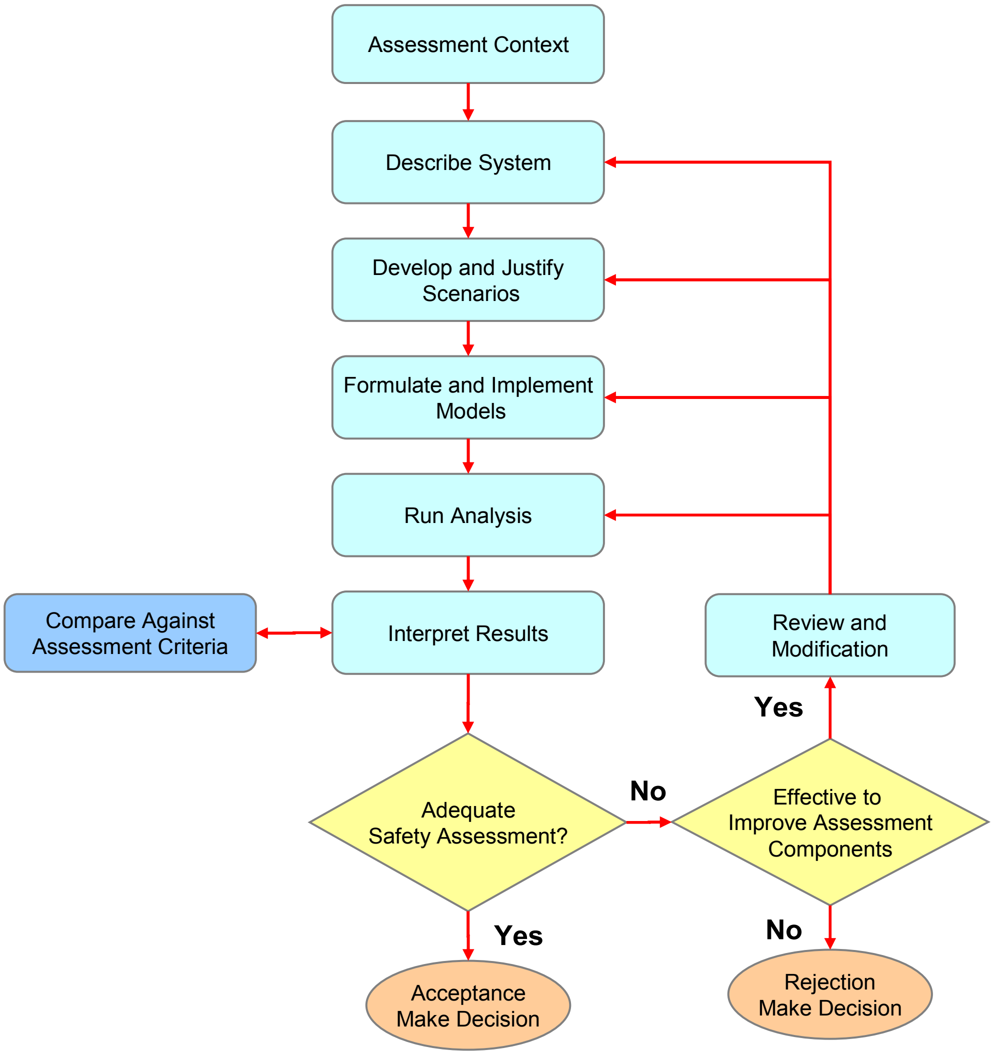

Figure 1. Example Performance Assessment Methodology (after IAEA 2004)

- To determine whether reasonable assurance of compliance with quantitative performance objectives for protection of human health can be demonstrated; and
- To identify critical data, facility design and information needs and model development needs for:
  1) Defensible and cost-effective licensing decisions and
  2) Developing and maintaining operating limits, such as, waste acceptance criteria[1].

The modeling evaluations conducted for PAs include assessments of contaminant migration through environmental pathways (e.g., air, groundwater, and surface water) and potential human exposures to the contaminants in various exposure media (e.g., soil,

[1] Based on the definition provided in U.S. National Council for Radiation Protection and Measurements Report No. 152, Performance Assessment of Near-Surface Facilities for Disposal of Low-Level Radioactive Waste (2006).

drinking water, crops, and livestock). The potential for inadvertent human intrusion into the waste as an accidental exposure pathway is also addressed.

Substantial experience exists in conducting PAs for radioactive waste disposal facilities owned or regulated by the U.S. Department of Energy, US Nuclear Regulatory Commission. PAs have been completed for several dedicated radioactive waste disposal facilities in the U.S. Department of Energy Complex (e.g., Hanford, Idaho, and Savannah River Sites, Nevada Test Site, and the Los Alamos National Laboratory) and for commercially operated facilities in South Carolina, Washington State, Utah and Texas. Current U.S. Department of Energy Performance Assessments, Composite Analyses, and Special Analyses are listed in Table 1.

2.1 Role of Performance Assessments

Practical experience has proven that PAs and PA-like analyses can provide useful input at many different points throughout the lifecycle of a variety of different waste management facilities (e.g., siting, design decisions, operational limits, monitoring programs, closure options, remediation of contaminated areas, in-situ decommissioning).

Internationally, the concept of a safety case has been introduced and includes disposal facility PAs as well as many other activities that contribute to the safety basis for these facilities (waste acceptance criteria, monitoring, facility design, operating procedures, research and development). The Safety Case is a package of information that supports safe operation and closure of a facility. The safety case approach highlights the role of the PA as a (1) management tool to guide many safety-related activities associated with waste management and (2) tool for demonstrating compliance with performance objectives.

2.2 PA Complexity

Early PAs conducted for near-surface disposal facilities had a large influence from health physics and were oriented towards the use of deterministic calculations to demonstrate compliance using relatively simplified modeling approaches. As PAs have evolved, the concept of the iterative approach and the broader view of PA as a management tool in addition to a compliance tool has become the norm. This broader view has resulted in the use of increasingly sophisticated approaches capable of representing physicochemical processes that can be used to guide improved designs for waste forms, containers and facilities. There has also been an increase in the use of stochastic modeling approaches to better capture the uncertainty in model results. Benefits of these improvements have been recognized both in defensibility with stakeholders and also improved efficiency of waste management.

2.3 Sensitivity and Uncertainty Analysis

Increasing use of PAs as design and management tools resulted in an increased emphasis on quantification of uncertainty, including significant use of sensitivity analysis to better understand the processes and design features that have the most influence on the conclusions of the assessment. This information is used to focus activities such as design, site characterization and model improvements on the areas expected to provide the most benefit. As mentioned above, stochastic or probabilistic approaches are becoming more commonly used for sensitivity and uncertainty analysis. However, some of the most recent PAs conducted for the USDOE have used what is being termed a “hybrid” approach, which includes the use of a combination of deterministic (point source) and probabilistic approaches to provide multiple lines of reasoning in support of a decision. Figure 2 is an illustration of “hybrid” results including a deterministic base case and sensitivity results overlain on the range of results generated from the probabilistic simulations.

Table 1. Current US Department of Energy Performance Assessments, Composite Analyses, and Special Analyses (reproduced from US DOE, courtesy of L. C. Suttora, DOE-EM 41, 2009)

Site	Facility	Document Title	Date	Document Number
Idaho	RWMC PA	Performance Assessment for the RWMC Active Low-Level Waste Disposal Facility at the Idaho National Laboratory Site	9/07	DOE/NE-ID-11243
	RWMC CA	Composite Analysis for the RWMC Active Low-Level Waste Disposal Facility at the Idaho National Laboratory Site	9/08	DOE/NE-ID-11244
	ICDF PA	Idaho CERCLA Disposal Facility Performance Assessment	--	--
LANL	Area G PA/CA	Performance Assessment and Composite Analysis for Los Alamos National Laboratory Technical Area 54, Area G, Revision 4	10/08	LA-UR-08-06764
NTS	Area 3 PA	Performance Assessment/Composite Analysis for the Area 3 Radioactive Waste Management Site, Nevada Test Site, Nye County, Nevada (Revision 2.1)	10/00	DOE/NV--491-REV 2.1
	Area 5 PA	Performance Assessment for the Area 5 Radioactive Waste Management Site at the Nevada Test Site, Nye County, Nevada (Rev. 2.1)	1/98	DOE/NV/11718-176, UC-721
		Addendum 1, Performance Assessment for the Radioactive Waste Management Site at the Nevada Test Site, Nye County, Nevada	11/01	DOE/NV/11718—176-ADD1
		Addendum 2 to the Performance Assessment for the Radioactive Waste Management Site at the Nevada Test Site, Nye County, Nevada	6/06	DOE/NV/11718—176-ADD2
	Area 5 CA	Composite Analysis for the Area 5 Radioactive Waste Management Site at the Nevada Test Site, Nye County, Nevada	9/01	DOE/NV--594
		Addendum 1, Composite Analysis for the Area 5 Radioactive Waste Management Site at the Nevada Test Site, Nye County, Nevada	11/01	DOE/NV--594-ADD1
	Area 5 SA	Special Analysis of Transuranic Waste in Trench T04C at the Area 5 Radioactive Waste Management Site, Nevada Test Site, Nye County, Nevada, Rev.1.0	5/08	DOE/NV/25946--470
	GCD Boreholes PA	Compliance Assessment Document for the Transuranic Wastes in the Greater Confinement Disposal Boreholes at the Nevada Test Site, Volume 2: Performance Assessment, Revision 2.0	7/21/00	SAND2001-2977

Table 1. Current US Department of Energy Performance Assessments, Composite Analyses, and Special Analyses (reproduced from US DOE, courtesy of L. C. Suttora, DOE-EM 41, 2009) (contd)

Site	Facility	Document Title	Date	Document Number
RL	200 East PA	Performance Assessment for the Disposal of Low-Level Waste in the 200 East Area Burial Grounds	8/15/96	WHC-SD-WM-TI-730, Rev. 0
		Addendum to the Performance Assessment Analysis for Low-Level Waste Disposal in the 200 East Area Active Burial Grounds	Undated	HNF-2005, Rev. 0
	200 West PA	Performance Assessment for the Disposal of Low-Level Waste in the 200 West Area Burial Grounds	6/95	WHC-EP-0645
		Addendum to the Performance Assessment Analysis for the Low-Level Waste Disposal in the 200 West Area Active Burial Grounds	12/20/96	HNF-SD-WM-TI-798, Rev. 0
	200 Area CA	Composite Analysis for Low-Level Waste Disposal in the 200 Area Plateau of the Hanford Site	3/98	PNNL-11800
		Addendum to Composite Analysis for Low-Level Waste Disposal in the 200 Area Plateau of the Hanford Site	9/01	PNNL-11800, Addendum 1
	IDF PA	Hanford Integrated Disposal Facility Performance Assessment: 2005 Version	1/21/05	ORP-25439, Revision 0
SRS	E Area PA	E-Area Low-Level Waste Facility DOE 435.1. Performance Assessment	7/08	WSRC-STI-2007-00306, Revision 0
	Z Area PA	Radiological Performance Assessment for the Z-Area Saltstone Disposal Facility (U)	12/18/92	WSRC-RP-92-1360
	Z Area SA	Special Analysis: Revision of Saltstone Vault 4 Disposal Limits (U)	5/26/05	WSRC-TR-2005-00074, Revision 0
	E and Z Area CA	Composite Analysis, E-Area Vaults and Saltstone Disposal Facilities, Rev. 0	9/97	WSRC-RP-97-311
	F-Tank Farm PA	Performance Assessment for the F-Tank Farm at the Savannah River Site	6/27/08	SRS-REG-2007-00002, Revision 0

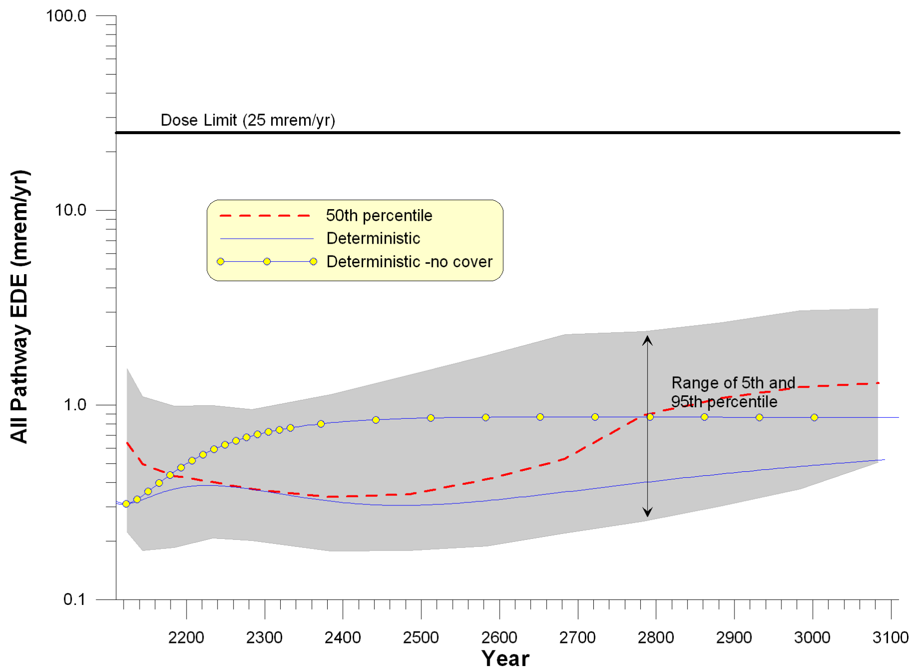

Figure 2. Dose Results for a "Hybrid" PA (DOE-NE/ID 2007)

2.4 Role of Cementitious Materials

PAs that include analyses of cementitious barriers have traditionally been associated with waste disposal. More recently, challenging environmental remediation and decommissioning activities also require PA-like analyses that are capable of taking credit for the performance of cementitious materials. This broader category of problems involves numerous applications at many different locations (e.g., in-situ decommissioning of reactors and large facilities, tank closures, remediation of contaminated sites and old burial grounds) (see Figure 3).

Many of these PAs include consideration of designs involving cementitious barriers and/or waste forms (see Table 2). Cementitious materials can be expected to perform some physicochemical function for waste isolation for time frames of thousands or possibly tens of thousands of years. Although, it is generally accepted that cementitious materials can be engineered to perform over long time frames, a set of accepted tools for modeling the coupled behavior of the processes that affect aging of cementitious materials over those time frames is not available. Improved representation of cementitious barriers in PAs can help reduce conservatism in PAs, potentially decrease near term and life-cycle disposal costs, and in some cases support disposal of increased radionuclide inventories.

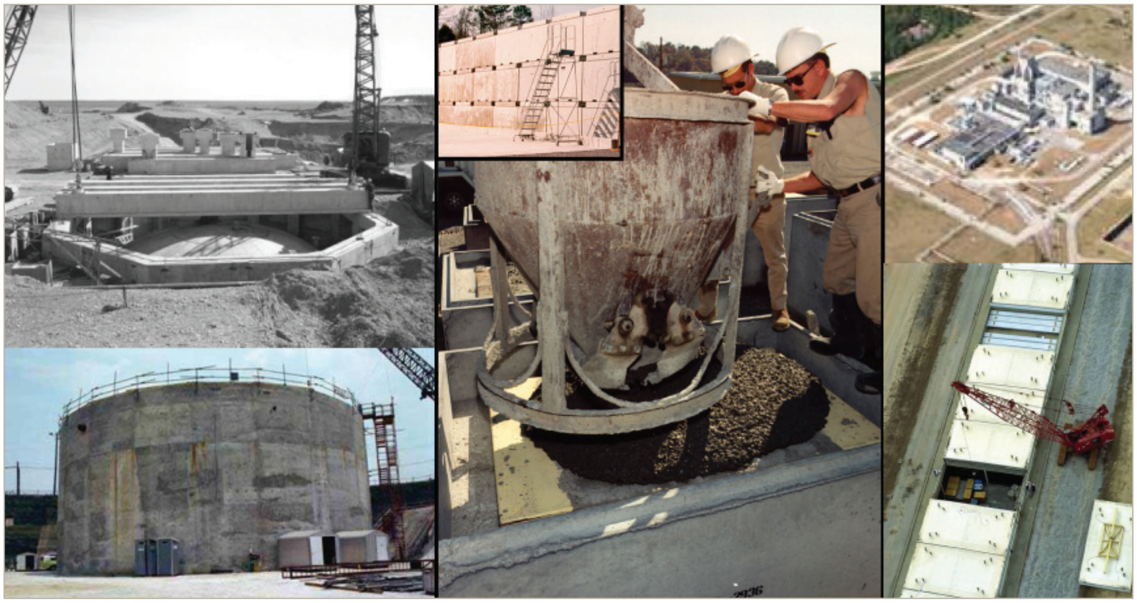

Figure 3. Examples of Cementitious Materials Considered in PAs (left to right: INL and SRS reinforced concrete HLW tank vaults, grouted containerized waste, reinforced concrete reactor facility, and concrete waste disposal vaults)

Table 2. Cementitious Barriers Considered for DOE Low-Level Waste Disposal Facilities

Site	US DOE Facility	Cementitious Barrier	PA Analysis
Idaho (semi arid, deep ground water)	RWMC PA	Present and assumed to have properties of soil	No credit for hydraulic properties - Treated as soil; Credit for diffusion barrier for selected containers
	RWMC CA	Present and assumed to have properties of soil	
LANL (arid, deep ground water)	Area G PA/CA	Present as small amounts of waste forms and rubble	No physical, hydraulic, or chemical credit – Treated as soil
NTS (arid, no ground water pathway)	Area 3 PA	No	Not considered No credit
	Area 5 PA		
	Area 5 CA		
	Area 5 SA		
	GCD Boreholes PA		
RL (semi arid, moderately deep ground water)	200 East PA	Containers Grouted Waste (waste forms)	Primarily physical stabilization, Chemical stabilization for uranium, A few waste streams credited as diffusion barriers
	200 West PA		
	200 Area CA		
	IDF PA	Potential for Containerized Grouted Waste / Waste Form	Potential hydraulic, chemical and physical barrier
SRS (humid, shallow groundwater)	E Area PA	Concrete Vaults Components in Grout Grout Fill	Diffusion barrier Chemical stabilization Physical stabilization
	Z Area PA	Cementitious Salt Waste Form Concrete Vaults	Diffusion barrier Chemical stabilization Physical stabilization
	Z Area SA	Cementitious Salt Waste Form Concrete Vaults	Diffusion barrier Chemical stabilization Physical stabilization
	E and Z Area CA	Concrete Vaults Components in Grout Grout Fill Cementitious Salt Waste Form	Diffusion barrier Chemical stabilization Physical stabilization
	F-Tank Farm PA	Concrete Vaults Grout Fill Concrete Base Mat	Diffusion barrier Chemical stabilization Physical stabilization

3.0 REFERENCES

IAEA 2004, Safety Assessment for Near Surface Disposal of Radioactive Waste – Safety Guide, IAEA Safety Standards Series No. WS-G-1.1, International Atomic Energy Agency, Vienna, Austria.

USDOE-NE/ID 2007, Performance Assessment for the RWMC Active Low-Level Waste Disposal Facility at the Idaho National Laboratory Site, DOE/NE-ID-11243, US Department of Energy, Idaho Falls, ID.

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http://www.gc.energy.gov/NEPA/environmental_assessments.htm

OVERVIEW OF PERFORMANCE ASSESSMENTS AND MODELING OF CEMENTITIOUS BARRIERS

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LIST OF ACRONYMS AND ABBREVIATIONS

AEA	Atomic Energy Act
ALARA	as low as reasonably achievable
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
CRESP	Consortium for Risk Evaluation with Stakeholder Participation
CMI	Corrective Measures Implementation
CMS	Corrective Measures Study
CWI	CH2M-WG Idaho, LLC
D&D	Decontamination and Decommissioning
DCGL	Derived Concentration Guideline Limits
DP	Decommissioning Plan
DOE	United States Department of Energy
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
ETR	Engineering Test Reactor
FFA	Federal Facility Agreement
FONSI	Finding of No Significant Impact
FTF	F-Tank Farm
GAO	Government Accounting Office
ha	hectare
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
ICDF	INEEL CERCLA Disposal Facility
IDEQ	Idaho Division of Environmental Quality
IDF	Integrated Disposal Facility
ILV	Intermediate Level Vault

LIST OF ACRONYMS AND ABBREVIATIONS (contd)

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
LOCA	loss-of-coolant accidents
LTP	license termination plan
LTR	License Termination Rule
MCL	maximum contaminant level
MEPAS	Multimedia Environmental Pollutant Assessment System
mrem	millirem
MWMF	Mixed Waste Management Facility
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	U.S. Office of Management and Budget
ORNL	Oak Ridge National Laboratory
PA	Performance Assessment
PET	potential evapotranspiration
PRG	preliminary remediation goal
PSDAR	Post-Shutdown Decommissioning Activities Report
Pu	plutonium
PUREX	Plutonium Recovery and Extraction
PWR	Pressurized Water Reactor
RAGS	Risk Assessment Guidance for Superfund
RCRA	Resource Conservation and Recovery Act
RDs/RAs	remedial designs/remedial actions
REM	mrem
RESRAD	RESidual RADioactivity
RFA/RFI	RCRA Facility Assessment/RCRA Facility Investigation

LIST OF ACRONYMS AND ABBREVIATIONS (contd)

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 Acts
SCDHEC	South Carolina Department of Health and Environmental Control
SDA	Subsurface Disposal Area
Sr	Strontium
SER	Safety Evaluation Report
SRNL	Savannah River National Laboratory
SWMU	solid waste management unit
SWSA	Solid Waste Storage Area
Tc	Technetium
TEDE	total expected dose equivalent
TFF	Tank Farm Facility
TBP	tributyl phosphate
TRU	transuranic
TSDF	treatment, storage, and disposal facility
TWRS	Tank Waste Remediation System
USDOE	United States Department of Energy
USEPA	United States Environmental Protection Agency
USGAO	United State Government Accounting Office
USNRC	United States Nuclear Regulatory Commission
UST	underground storage tank
WCF	Waste Calcining Facility
WSRC	Washington Savannah River Company

Overview of Performance Assessments and Modeling of Cementitious Barriers

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1.0 INTRODUCTION

Performance assessments (PA) and PA-like analyses are conducted to provide an assessment 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. Although there may be different goals and fundamental approaches to conducting such assessments for waste management activities that need to be maintained, there are a number of similarities. 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 similarities associated with specific aspects of the different approaches that can and should be shared from the perspective of consistency and continuous improvement.

The most rigorous consideration of cementitious barriers has traditionally been associated with disposal of low-level radioactive waste. More recently, as more difficult facility decommissionings and closures and remediation activities are being undertaken, there has been an increased need to be able to take credit for cementitious barriers as part of a broader range of waste management activities. Figure 1 illustrates examples of cementitious barriers encountered in radioactive waste management and disposal activities. This expanded use has highlighted two issues: the need for improved methods for assessing cementitious barriers and the need for improved sharing of information

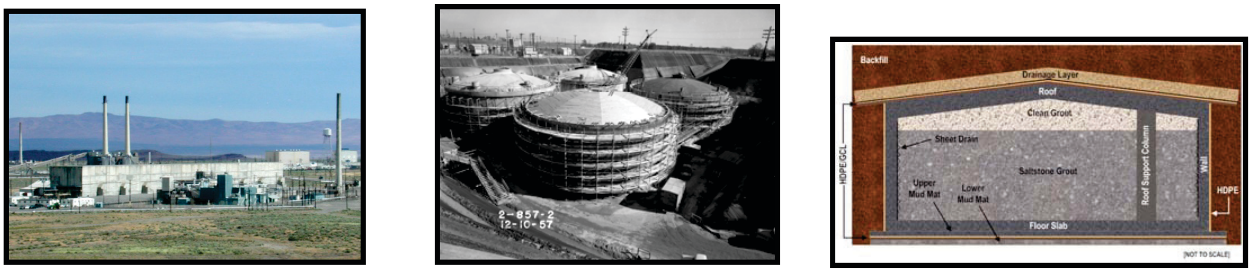


Figure 1. Examples of Cementitious Barriers
(left to right, Reactor Facility, High-Level Waste Under Construction, Schematic of Waste Form in a Concrete Vault-Post Closure)

between analysts conducting assessments in support of these different regulatory activities. Approaches for uncertainty analyses are an important aspect of any PA exercise and are summarized in a separate chapter of this report.

Cementitious materials are often used as engineered barriers in waste disposal and other facilities as a means to contain the radioactive waste and/or to limit the migration of radionuclides into the accessible environment. One common form of barrier is the cementitious material as the waste form, that is, the waste is intimately mingled with the cementitious material. Another common form is that of containment, something intended to isolate the waste from the environment, such as a container or a vault. In this form, the waste is segregated from the cementitious material. In either case, the release of radioactive waste can be controlled as a function of the rate at which the cementitious material is assumed to degrade and lose its effectiveness as a chemical and physical barrier.

“Degrade” is often used synonymously with “aging”. The aging of the cementitious barrier is the parameter of interest in the development of a PA, and in most cases the barrier’s performance is seen to decrease with time, hence the use of “degrade”. There are two aspects to be considered in aging, the effect on hydraulic properties and the effect on chemical properties. While these two phenomena are in actuality closely coupled, in the PA arena they are often modeled independently. In order to take credit for the benefits of cementitious materials in a PA or PA-like analysis, it is necessary to have models and data sufficient to stand up to external review. This concern has often resulted in overly-conservative assumptions being made regarding barrier degradation. Although expedient in the short-term, such approaches could result in decisions being made that are more costly or over-restrictive over the long term.

This document is primarily directed at an overview of PA and PA-like analyses used by the US Department of Energy (USDOE) with some examples from the U.S. Nuclear Regulatory Commission (USNRC). The focus of this report is on summarizing the regulatory expectations and providing some illustrative example applications of PAs and PA-like analyses and their approaches for modeling cementitious materials both as waste forms and barriers for disposal facilities, remediation, and decommissioning.

Approaches are not described in detail, but enough information is provided to allow the reader to determine the type of credit that has been taken for cementitious materials and some perspective on processes considered. The reader is expected to consult the original references for detailed information about the models used. Furthermore, the intent of the document is not to pass judgment on the approaches that have been used. The purpose is to survey approaches that have been used, identify similarities and differences and make recommendations regarding future needs.

2.0 REGULATORY DRIVERS

Performance assessments and PA-like analyses are conducted within a number of different regulatory frameworks. This diversity of regulatory environments often involves multiple different regulators and analysts conducting assessments for projects for a single 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 any guidance or recommendations related to modeling of cementitious barriers.

2.1 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 radioactive low-level waste disposal facility. PAs are also a means to make decisions with regard to siting, design, operation and development of closure plans for a disposal facility. Different regulators can be involved depending on the purpose 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 U.S. activities.

2.1.1 DOE Order 435.1 and Supporting Manuals: DOE LLW Disposal

2.1.1.1 Assessment Related Requirements

The US Department of Energy's (USDOE's) authority to manage and regulate radioactive wastes is promulgated in the Atomic Energy Act (AEA) (AEA 1954) and the Nuclear Waste Policy Act (NWPA) (NAS 2006). DOE Order 435.1, Radioactive Waste Management (USDOE 2001) implements regulatory guidance 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 (USDOE 2001b). 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 Section IV.P include performance objectives for all pathways, the air pathway, and for release of radon. The requirements related to performance assessments include the need to: (1) demonstrate compliance with the performance objectives, (2) establish limits on waste concentrations based on the intruder performance measures, (3) identify a baseline point of compliance, (4) conduct a sensitivity/uncertainty analysis, and (5) address requirements related to protection of water resources.

There is also a requirement to conduct a Composite Analysis that includes contributions from the disposal facility and any other collocated sources that could contribute to a composite dose to a member of the public. The composite analysis is used to ensure that the total dose associated with the facility and any other sources remains within levels allowed for exposure to the general public.

2.1.1.2 Guidance Related to Assessment of Cementitious Barriers

There are no specific requirements or recommendations in DOE O 435.1 or DOE M 435.1-1 regarding specific approaches to be used for the assessment of cementitious barriers. Thus, there is no prescribed approach. There is guidance in Chapter IV of DOE Guide 435.1-1 (USDOE 1999) that suggests that credit may be taken for use of intruder barriers and durable waste forms when considering the potential for intrusion into specific wastes. Thus, it is possible to develop barriers or waste forms involving cementitious materials that could serve as a means to delay the consideration of intrusion while the integrity of the barrier or waste form is intact. There is no specific discussion of how to consider cementitious materials in assessments for the groundwater or air pathways.

2.1.2 NRC 10 CFR Part 61: Commercial LLW Disposal

2.1.2.1 Assessment Related Requirements

NRC regulated LLW disposal facilities must comply with 10 CFR Part 61, which was promulgated in 1982. State regulators responsible for LLW disposal also use 10 CFR Part 61 as a basis for their regulations. 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 from inadvertent intruders, and minimizing the need for active maintenance after closure.

2.1.2.2 Guidance Related to Assessment of Cementitious Barriers

There are no specific requirements or recommendations in Part 61 regarding specific approaches to be used for the assessment of cementitious barriers. It is specified in 61.7 that intruder barriers for Class C waste must have an effective life of at least 500 years. There is no specific discussion of how to consider cementitious materials in assessments for the groundwater or air pathways. However, there is guidance on assessing the stability of cementitious waste forms in the NRC Branch Technical Position on Waste Forms issued in January 1991 (USNRC 1991).

NRC Staff also 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. The role of engineered barriers was flagged as one of five key issues in the document. In Section 3.2.2 of NUREG-1573, NRC Staff concluded that cementitious barriers can remain effective as intruder barriers for more than 500 years but any such assumptions must be defended on a case-by-case basis. Information must also be provided regarding the expected degraded condition of the barrier in respect of its designed physical and chemical functions.

Section 3.3.4 of NUREG-1573 includes more detailed suggestions for addressing performance of engineered barriers. The importance of addressing interactions between different materials is emphasized along with verification of construction quality. Section 3.3.4.4

includes additional information about addressing performance of engineered barriers. The emphasis of the suggestions is on general characteristics to be considered for intact, degrading and degraded performance (e.g., need to address cracking when considering hydraulic conductivity of a cementitious barrier).

2.1.3 NDAA Section 3116: HLW Tanks and Facility Closure

2.1.3.1 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) (NAS 2006). 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.

2.1.3.2 Guidance Related to Assessment of Cementitious Barriers

There is no specific guidance in Section 3116 for the consideration of cementitious barriers. However, NRC Staff prepared Draft Final NUREG-1854, "NRC Staff Guidance for Activities Related to U.S. Department of Energy Waste Determinations (USNRC 2007)". NUREG-1854 includes recommendations for reviews of PAs conducted for Section 3116 issues. Engineered barriers are addressed in Section 4.3.2 of NUREG-1854.

NUREG-1854 discusses considerations for reviews of modeling degradation of chemical performance of cementitious barriers. The importance of redox conditions and pH in terms of chemical performance are highlights, but it is also recommended to address the impacts of physical changes in a cementitious barrier and the associated impacts on changes in a barrier's effectiveness from a chemical perspective. NRC Staff also refer to NUREG-1573 as a source of information and similar to NUREG-1573 re-emphasized the importance of considering interactions of different materials and also construction quality. Section 4.3.2.2 of NUREG-1854 includes a relatively detailed list of review considerations for assessments of engineered barriers.

2.1.4 NCRP Guidance on PA: LLW Disposal

In 2005, the National Council on Radiation Protection and Measurements (NCRP) completed NCRP Report number 152, "Performance Assessment of Near-Surface Facilities for Disposal of Low-Level Radioactive Waste" (NCRP 2005). This report includes relatively detailed discussions regarding approaches that can be used for specific aspects of modeling associated with PAs and PA-like analyses.

2.1.4.1 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. The document is intended to serve as a resource for those conducting PAs rather than as a requirement for how the modeling should be done.

2.1.4.2 Guidance Related to Assessment of Cementitious Barriers

The NCRP guidance includes a section on the performance of concrete barriers. This section focuses on water flow through concrete and mechanisms for degradation of concrete. The report addresses approaches that have been used in the past and some approaches that have been proposed for use. The document includes a number of references for more detailed information.

2.1.5 International Atomic Energy Agency

2.1.5.1 Assessment Related Requirements

The International Atomic Energy Agency (IAEA) publishes non-binding requirements related to radioactive waste safety as well as guidance for implementation. 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 (IAEA 1999a and b, respectively). Internationally, the term Safety Assessment is used rather than PA.

The Safety Requirement is intended to establish requirements that must be met to ensure safety. 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. There are also statements regarding credit for institutional controls and how to address human behavior in addition to a recommendation to use current human habits as the basis for projections of doses in the future.

The IAEA has sponsored several projects addressing Safety Assessment approaches. Although these projects are not intended to represent guidance or requirements, there have been specific ideas provided that can be considered good practices. For example, the project on Improvement of Safety Assessment Methodologies (ISAM) resulted in development of a basic methodology for the conduct of safety assessments that is often cited. See Figure 2.

2.1.5.2 Guidance Related to Assessment of Cementitious Barriers

The Safety Requirement described above 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 modeling of cementitious barriers. The IAEA Safety Guide on Safety Assessment discusses the need to address degradation of barriers and the associated changes in performance but does not include any specific guidance (IAEA 1999b).

2.2 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, the AEA and NWPA are not the sole applicable federal statutes (NAS 2006). Additional legislation including the: Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA); Resource Conservation and Recovery Act (RCRA); National Environmental Policy Act (NEPA); and correlative state and local laws also play critical regulatory roles. The relevant considerations under these additional statutes often go well beyond and adopt different practices than 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 USEPA and by the states (although often through delegated authority) (NAS 2006).

At the USDOE Savannah River Site (SRS) in Aiken, South Carolina, the two primary federal laws that drive cleanup are RCRA, which establishes a system for tracking and managing hazardous wastes from generation to disposal, and CERCLA, which addresses protection and cleanup from known waste sites (WSRC 2008). The SRS is satisfying the requirements of these laws via a Federal Facility Agreement (FFA) (WSRC 1993) between the USDOE, USEPA Region 4, and the South Carolina Department of Health and Environmental Control (SCDHEC). The SRS FFA, which is required under CERCLA, specifies how contamination or potential contamination will be addressed in accordance with both RCRA and CERCLA requirements. NEPA evaluations of alternative closure options often serve as inputs to the RCRA and CERCLA feasibility studies (Shedrow et al. 1993).

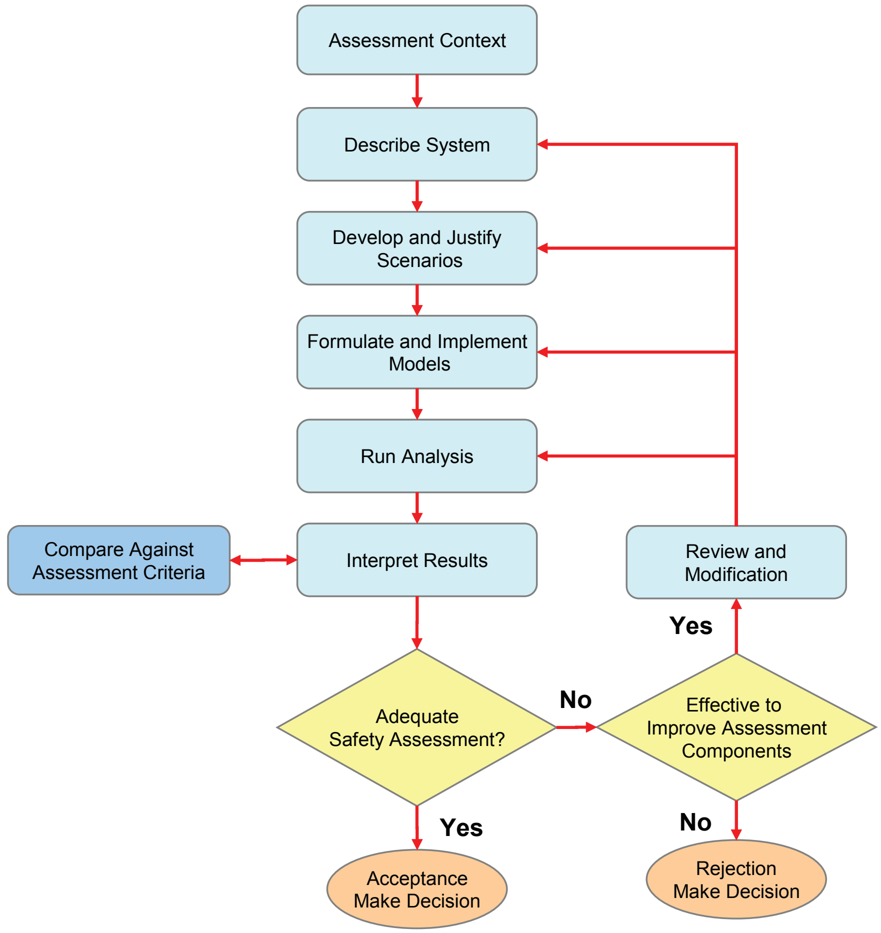


Figure 2. Example of Safety Assessment Methodology (after IAEA 2004)

Sites were identified in 1986 within the Idaho Site that could pose unacceptable risks. DOE-ID entered into a Consent Order and Compliance Agreement (COCA) with the USEPA (USDOE-ID 1986) calling for remediation of active and inactive waste disposal sites under RCRA. In 1989, the USEPA added the Idaho National Engineering Laboratory (INEL) to the National Priorities List (NPL) under CERCLA. The Idaho Site Federal Facilities Agreement (USDOE-ID 1991), which superseded parts of the COCA, was adopted by USDOE-ID, USEPA, and the Idaho Division of Environmental Quality (IDEQ) in 1991 to implement the INEEL remedial actions under CERCLA. The Energy Secretary's policy statement on NEPA stipulated that the USDOE will rely on the CERCLA process for review of actions to be taken under CERCLA (USDOE 1994a).

The USDOE, which operates the Hanford Site in Washington State, the USEPA, and the State of Washington Department of Ecology, signed a comprehensive cleanup and compliance agreement on May 15, 1989[1]. The Hanford Federal Facility Agreement and Consent Order (or Tri-Party Agreement) is an agreement for achieving compliance with the CERCLA remedial action provisions and with the RCRA treatment, storage, and disposal unit regulations and corrective action provisions (USDOE

[1] From the Hanford Site Tri-Party Agreement available at http://www.hanford.gov/?page=91&parent=0 (accessed on February 27, 2009).

1989a). More specifically, the agreement 1) defines and ranks CERCLA and RCRA remedial commitments, 2) establishes responsibilities, 3) provides a basis for budgeting, and 4) indicates the goal of achieving full compliance and remediation. The agreement is legally binding and consists of two main parts: 1) the legal agreement itself which describes the roles, responsibilities and authority of the three agencies in the cleanup, compliance, and permitting processes and 2) the action plan to implement the cleanup and permitting efforts[2].

2.2.1 CERCLA

In 1990 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[3] (USDOE 1994b). The primary difference between CERCLA and RCRA is that CERCLA addresses uncontrolled releases of hazardous substances from facilities no longer in operation where contamination resulted from past practices; by contrast, RCRA focuses on prevention and remediation of releases from currently operating facilities.

CERCLA applies to all Federal agencies (USDOE 1994b). Section § 120(a)(1) states that each U.S. department, agency, and instrumentality shall be subject to, and comply with, the Act in the same manner and to the same extent, both procedurally and substantively, as any non-governmental entity. This intent for Federal agencies is continued in Section § 120(a)(2), which requires that all guidelines, rules, regulations, and criteria that are applicable to assessments, evaluations under the National Contingency Plan (NCP) (40 CFR Part 300), inclusion on the National Priorities List (NPL), or remedial actions shall also be applicable to facilities which are owned or operated by a U.S. department, agency, or instrumentality in the same manner and to the same extent as are applicable to other facilities. Section 120 also includes many requirements applicable only to Federal agencies including (USDOE 1994b):

- All potential Federal CERCLA sites be listed on the Federal Agency Hazardous Waste Compliance Docket.
- The responsible Federal agency completes a preliminary assessment for each site listed on the Docket.
- National Priorities List (NPL) listing decisions are made for those sites on the Docket. For Federal sites on the NPL, the responsible Federal agency, in consultation with the USEPA commence remedial investigation/feasibility study (RI/FS) within 6 months of NPL listing.
- The responsible Federal agency enter into an Inter-Agency Agreement with USEPA to conduct a remedial action within 180 days of the completion of the RI/FS.
- There is "substantial progress" in conducting the remedial action within 15 months of completion of the RI/FS.

Executive Order 12580 (EO12580 1987), Superfund Implementation, delegated the responsibility for CERCLA compliance at Federal facilities to each responsible official (i.e., Secretaries of Defense and Energy, and heads of other Executive Branch departments or agencies) (USDOE 1994b). The USDOE issued Order 5400.4 (USDOE 1989b), Comprehensive Environmental Response, Compensation, and Liability Act Requirements, establishing their policy regarding CERCLA compliance and included (USDOE 1994b):

[2] Additionally, a "Community Relations Plan" describes how the general public will be informed and involved throughout the process.
[3] CERCLA was amended by the Superfund Amendments and Reauthorization Act of 1986 (SARA) (Pub. L. No. 99-499).

- Responding to releases of hazardous substances from USDOE facilities,
- Entering into Federal Facility Agreements (FFAs) with USEPA and the State at both NPL and non-NPL sites for the purpose of conducting RIs/FSs and remedial designs/remedial actions (RDs/RAs),
- Where appropriate, integrating RCRA Corrective Action with CERCLA remedial actions to ensure that the RCRA Corrective Action is not inconsistent with the NCP, and
- Conducting natural resource damage assessments as required for resources under USDOE trustees.

2.2.1.1 Assessment Related Requirements

The overview diagram in Figure 3 illustrates the similarities between the discovery, assessment, and action phases of the CERCLA remedial action and the RCRA corrective action (USDOE 1994b). As indicated in the diagram, there are a number of distinct assessments required under CERCLA including the preliminary site assessment and remedial investigation/feasibility study (RI/FS).

If a removal action is not required, then a preliminary site assessment is performed as outlined in the NCP (40 CFR Part 300) at § 300.420. USDOE first conducts a remedial preliminary assessment, which involves collecting demographic and physical characteristics. Those sites not posing sufficient risk to human health or the environment to warrant response are screened out. A remedial site inspection, a more detailed investigation of site conditions often employing sampling, may be required to more fully evaluate site conditions. The information obtained from the preliminary assessment and site inspection is used to score the site using the Hazard Ranking System (HRS) (40 CFR § 300.425) (USEPA 1994b)[4]. If the site scores 28.5 or more, it may be placed on the NPL, which requires that a RI/FS be performed[5].

The RI/FS (as described in 40 CFR § 300.430) is used to characterize site risks and evaluate potential remedial actions. Sufficiently detailed information must be collected during the RI (often in a staged process) to characterize site conditions, determine the nature and estimate the extent of contamination, evaluate risks posed by the site, and assess the performance of potential remedial options to make an informed risk management decision (USDOE 1994b). The FS involves developing, screening[6], and evaluating each proposed remedial option. The RI and FS phases are conducted concurrently and interactively as illustrated in Figure 3[7]. The stages in the RI/FS assessment process that are of interest in terms conceptual and/or mathematical modeling include:

- **RI/FS Scoping:** Development of the conceptual model, which is a brief description of the site including suspected sources, contaminant pathways, and potential receptors to help identify decisions that must be made and deficiencies in existing information (USDOE 1987).
- **RI:** Site characterization is conducted to assess the threat a site poses to human health and the environment. The physical characteristics of the site are investigated and the sources of contamination and nature and extent of contamination are determined. Although these steps are primarily based on field activities, modeling activities may also play an important part.

[4] The CERCLA process differs from RCRA in that the RCRA Corrective Action does not employ a site-ranking model (USDOE 1994a).
[5] For sites that are not listed, USDOE's policy is to remedy contaminated sites under CERCLA or, when appropriate, other authorities such as RCRA. Within 6 months of listing, USDOE policy further requires that the facility enter into an agreement with USEPA and the State to establish the requirements for conducting the RI/FS (USDOE 1994a).
[6] One method for selecting acceptable remedial alternatives is based on a screening analysis using the effectiveness, implementability, and cost criteria per the National Oil and Hazardous Substances Pollution Contingency Plan (NCP) 40 CFR 300).
[7] This is another difference between CERCLA and RCRA—under RCRA, the RFI and CMS are not necessarily carried out concurrently (USDOE 1994a).

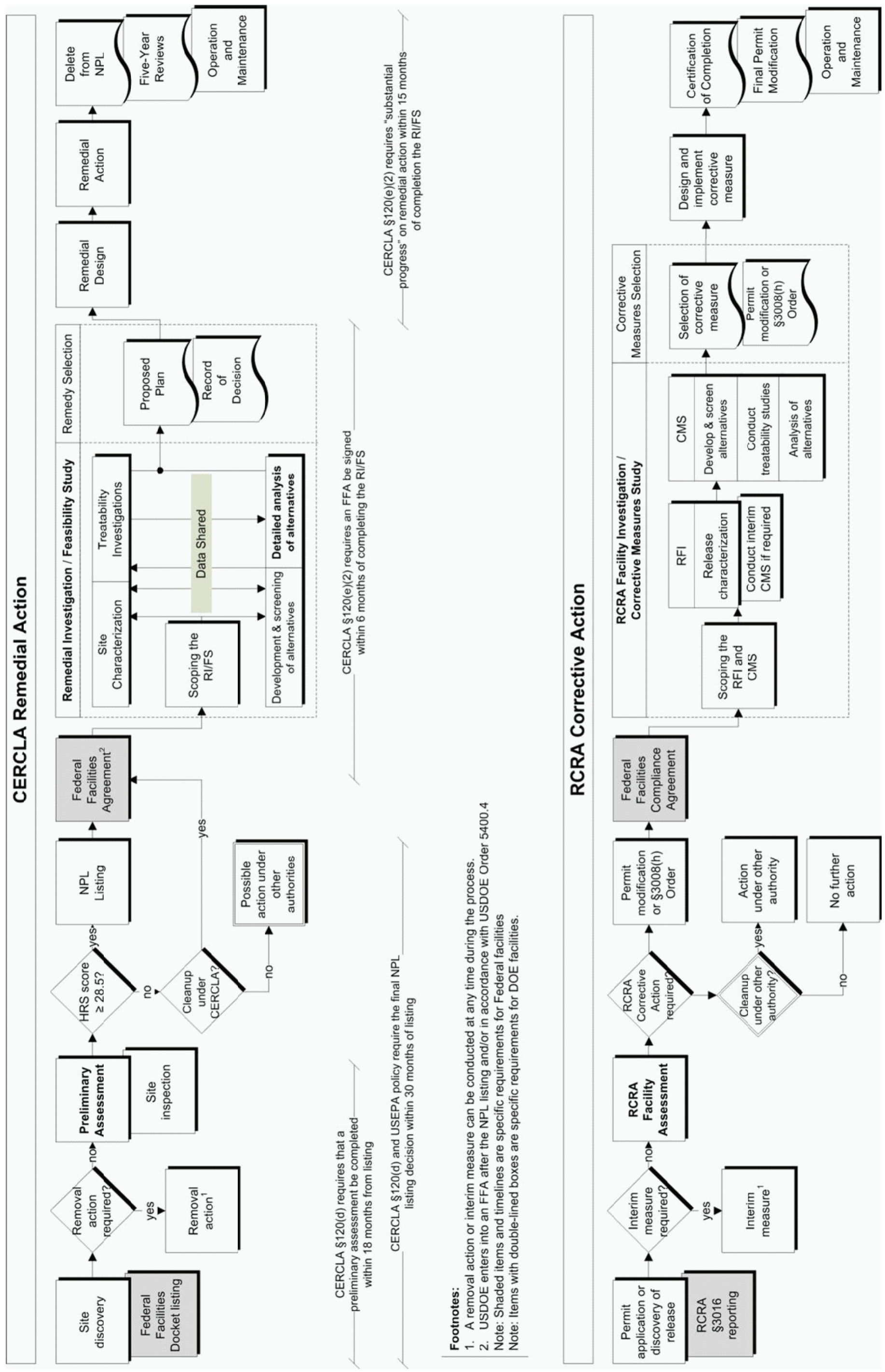


Figure 3. Overview of the Similarities between the CERCLA Remedial Action and RCRA Corrective Action (USDOE 1994a)

- **RI:** Baseline Risk Assessment (BRA) is used to evaluate the potential threat to human health and the environment posed by the site, which is an important element in making an informed risk management decision. USEPA published a detailed guidance document on conducting baseline risk assessments entitled Risk Assessment Guidance for Superfund (RAGS) (USEPA 1989a; USEPA 1989b; USEPA 1991a; USEPA 1991b; USEPA 1998; USEPA 2004).
- **FS:** Development and screening of remedial alternatives is used to develop a preliminary list of remedial alternatives. Often modeling is needed to assess the practicality of proposed alternatives given site conditions, which is related to one of the screening criteria. This step is needed to reduce the possible alternatives to a preliminary list of remedial alternatives, which may include the "no action" alternative.
- **FS:** Detailed analysis of remedial alternatives consists of examining the information needed to make an informed remedial action selection. Each alternative is assessed against the nine evaluation criteria found in the NCP (40 CFR §300.430(e)(9)(iii)) and the results are then compared with the other alternatives.

The RI/FS process results in the selection of a remedial option, a proposed plan for implementation, and a Record of Decision (ROD). The signing of the final ROD signifies the completion of the RI/FS phases (USDOE 1994b).

2.2.1.2 Requirements Related to Assessment of Cementitious Barriers

There are no specific requirements or recommendations in CERCLA or the Superfund Amendments and Reauthorization Act (SARA) regarding the approaches that must be used for the assessment of cementitious barriers. However, there is information in the Risk Assessment Guidance for Superfund (RAGS) that takes credit for engineered barriers (which may include cementitious barriers) when estimating the external radiation exposure risk (USEPA 1991a).

The risk calculations in RAGS require estimation of exposure media concentrations either using sampling results, model predictions, or a combination (USEPA 1989a; USEPA 1989b; USEPA 1991a). Credit may be taken for waste forms and barriers when projecting exposure media concentrations and risk into the future. However, this credit adds complexity and modeling uncertainty to the situation, which must be accounted for in the decision-making process (USEPA 1989a). One goal of the CBP is to help provide the basis for taking credit for this additional complexity and modeling uncertainty.

2.2.2 RCRA

The Resource Conservation and Recovery Act (RCRA) (Pub. L. 94-580) was signed into law in 1976. The purpose of RCRA is to protect human health and the environment via a comprehensive approach to hazardous and solid waste management at operating facilities (USDOE 1994b). This section focuses primarily on RCRA Subtitle C, entitled Hazardous Waste Management[8]. This subtitle established: methods for classifying wastes as hazardous, a "cradle-to-grave" tracking system, standards for generators and transporters, a permitting program and standards for the design and operation of hazardous waste treatment, storage, or disposal facilities (TSDFs), and requirements for facilities to implement hazardous waste minimization programs.

In 1984, Congress amended RCRA with the Hazardous and Solid Waste Amendments (HSWA) (Pub. L. 98-616). Key provisions included (USDOE 1994b):

[8] Two other important subtitles are Subtitle D, Solid Waste Management, and Subtitle I, Underground Storage Tanks.

- Regulation of small-quantity generators of hazardous waste,
- Requirements for the cleanup of releases of hazardous waste or hazardous waste constituents from solid waste management unit (SWMU) at TSDFS,
- Restrictions on land disposal of hazardous wastes, and
- Regulation of underground storage tanks (USTs) (Subtitle l).

These elements were intended to help reduce the total quantity of hazardous waste generated and to help prevent releases of such wastes into the environment.

RCRA Section 6001 indicates that it applies to Federal agencies by stating:

- "Each department, agency, and instrumentality of the Federal Government (1) having jurisdiction over any solid waste management facility or disposal site, or (2) engaged in any activity resulting in, or which may result in, the disposal or management of solid waste or hazardous waste shall be subject to, and comply with, all Federal, State, interstate, and local requirements."

Thus Federal agencies must comply with RCRA, including §3008(h) Corrective Action Orders, and the terms of permits issued under RCRA authority.

Federal agencies are also required to comply with RCRA under Executive Order 12088, Federal Compliance with Pollution Control Standards (EO12088 1978). Under this executive order, all Federal agencies must submit pollution control plans and request funding to implement and support pollution control activities. Under USDOE Order 5400.3, Hazardous and Radioactive Mixed Waste Program all USDOE facilities are also required to (USDOE 1989c):

- Comply with the requirements of RCRA and the AEA for the management of hazardous and radioactive mixed wastes generated by operations;
- Protect the environment and the safety of the public, DOE, and contractor employees through safe handling, transportation, treatment storage, and disposal of hazardous and radioactive mixed wastes generated through DOE operations; and
- Implement waste minimization procedures as specified in RCRA for hazardous and radioactive mixed wastes.

2.2.2.1 Assessment Related Requirements

In 1990 USEPA issued a proposed rule (55 FR 30798) establishing the procedural and technical requirements for conducting corrective actions under RCRA (USDOE 1994b)[9]. Through this proposed rule, EPA encouraged its use as guidance for conducting corrective actions by creating a four-phased approach: (1) RCRA Facility Assessment (RFA); (2) RCRA Facility Investigation (RFI); (3) Corrective Measures Study (CMS) and selection of the corrective measure; and (4) Corrective Measures Implementation (CMI).

The overview diagram in Figure 3 illustrates the similarities between the discovery, assessment, and action phases of the proposed RCRA corrective action approach and the CERCLA remedial action (USDOE 1994b). As indicated in the diagram, there are various assessments required under the RCRA proposed rule including the RFA, RFI, and CMS. Although many provisions of the Subtitle C proposal have been withdrawn and replaced by a results-based approach (USEPA 2003), the four phases described above still represent the assessments performed for the examples that will be described in this report and are still promulgated by the States programs for corrective actions[10].

[9] The proposed rule (55 FR 30798) created 40 CFR Part 264 Subpart S, Corrective Action for Solid Waste Management Units at Hazardous Waste Management Facilities.

[10] The proposed rule (55 FR 30798) created 40 CFR Part 264 Subpart S, Corrective Action for Solid Waste Management Units at Hazardous Waste Management Facilities.

Facilities may be required to begin corrective action: (1) when applying for a permit to treat, store, or dispose of hazardous waste; (2) upon discovering hazardous waste release from a Solid Waste Management Unit (SWMU) at a permitted or interim status facility; or (3) upon discovering additional SWMUs or hazardous waste releases from SWMUs at a facility already conducting a corrective action (USDOE 1994b). When a hazardous waste release is discovered, a corrective action is required through modification of the facility's permit or through a RCRA §3008(h) Corrective Action Order (USEPA 2008).

The RFA is the first phase in the RCRA corrective action process. The USEPA will conduct (or require the permittee to conduct under RCRA) the RFA (USDOE 1994b). The RFA consists of a review of existing information about a facility, a visit to the facility, and, if warranted, sampling of environmental media to determine if there is a hazardous waste release from SWMUs at the facility. If the RFA finds that hazardous wastes have been released, the facility permit will require modification or issuance of a RCRA §3008(h) Corrective Action Order to require an RFI for an interim facility[11]. If RCRA is not the correct legal vehicle for addressing the site, the USDOE will examine the requirements for remediation under other legal authorities (e.g., CERCLA). If no remediation is required, a "Determination of No Further Action" is issued by the USEPA (USDOE 1994b)[12].

As indicated in Figure 3, the RFI (40 CFR §264.510-13) is the second phase of the corrective action process. The RFI is a detailed investigation to determine the nature, extent, and migration rate of any releases and to provide the information necessary to develop a strategy for addressing contamination (USDOE 1994b). While similar to the CERCLA RI, the RFI is often more focused than the RI in that it pertains to characterization of releases from SWMUs rather than characterization of the entire facility for the RI (USDOE 1994b).

The third phase of the RCRA corrective action process is a Corrective Measures Study (CMS) (40 CFR §264.520-24). If the RFI finds that a corrective measure is required, the CMS, which corresponds to the CERCLA FS[13], is used to examine alternatives for the corrective measure. The stages in the corrective action process that are of interest in terms conceptual and/or mathematical modeling include:

RFA: The RFA is a screening process to determine if there is a hazardous waste release or threat at a Treatment, Storage, and Disposal Facility (TSDF). Information collected during this phase identifies those SWMUs, environmental media, or parts of a facility requiring further investigation; modeling may be used to supplement sampling information during this phase.

RFI: The RFI has three elements: information gathering and sampling activities, sample analysis and data verification, and periodic progress assessments (USEPA 1989c; USEPA 1989d; USEPA 1989e; USEPA 1989f). The first phase in the RFI is to collect and review available information on the release and the facility for information including the characteristics of the release, the environmental setting, evaluations of the threats posed to human health and the environment, and those actions taken to control or minimize threats. Predictive models may be useful for refining conceptualizations of the environmental setting (USEPA 1989c).

[11] If hazardous wastes have been released, a Federal Facility Compliance Agreement (FFCA) between DOE and EPA will be developed to require the facility to conduct further studies (USDOE 1994a).
[12] Interim measures taken to mitigate actual or potential threats may be conducted during any phase of the corrective action process.
[13] One difference between CERCLA and RCRA is that the RFI and CMS phases are not necessarily carried out concurrently; whereas the CERCLA RI and FS are (USDOE 1994a).

Use of predictive models during the RFI may also be appropriate for guiding the general development of monitoring networks. Models may be used in media-specific situations. For example, surface water models may be used to determine the extent of a monitoring system necessary for a stream. In general, model results are not acceptable for estimating release concentrations in an RFI; however, one exception is air. Atmospheric dispersion models are suggested for use with emission-rate monitoring or modeling to predict downwind release concentrations and to define the extent of a release (USEPA 1989c).

CMS: The CMS (USDOE 1993a) involves evaluating the likely effectiveness of proposed alternatives and analyzing and evaluating any testing results. USEPA has the authority to require testing, typically in the form of treatability studies, to occur concurrently with the RFI to prevent a delay in conducting the corrective measure (USDOE 1993a). Predictive models may be particularly useful in designing corrective measures (e.g., pumping and treating contaminated ground water) (USEPA 1989c).

Following the CMS, a permit modification or RCRA §3008(h) Order and an inter-agency agreement are developed to select the technology to be used as the corrective measure at the facility (USDOE 1994b).

2.2.2.2 Requirements Related to Assessment of Cementitious Barriers

Like CERCLA there are no specific requirements or recommendations in RCRA or HSWA regarding the approaches that must be used for the assessment of cementitious barriers. However, the guidance (USEPA 1989a; USEPA 1991a; USEPA 1991b; USEPA 1998; USEPA 2004) developed for human health risk assessment under CERCLA is generally used for RCRA[14]. Thus credit can be taken for waste forms and barriers when predicting exposure media concentrations and corresponding risks although any increases in modeling complexity and uncertainty must be taken into account in the decision-making process (USEPA 1989a). One goal of the Cementitious Barriers Partnership (CBP) is to allow more accurate predictions to be made when cementitious barriers are used in disposal.

One interesting distinction that has arisen involves grouting of RCRA wastes during treatment. RCRA specifically prohibits dilution of hazardous waste (40 CFR § 268.3, "Dilution prohibited as a substitute for treatment"). The standard involves dilution of waste in lieu of treatment, and the USEPA recognizes that such dilution (via grouting) that is a necessary part of the treatment process, which otherwise destroys, removes, or immobilizes the hazardous constituents, is normally permissible[15].

2.2.3 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., and its passage stimulated the types of citizen involvement and litigation that have been characteristic of the environmental arena ever since (Bear 1989). Growing concerns about environmental pollution and quality were addressed in NEPA, which was the foundation for inserting environmental considerations into federal decision-making and dramatically increased the amount of information available to the public and boosted the role of the judiciary in federal decisions concerning the environment and its protection (Bear 1989). NEPA established the US National Environmental Policies Council on Environmental Quality (CEQ) (CEQ 2007).

[14] For example, see EPA's *RCRA Risk Assessment* at http://www.epa.gov/oswer/riskassessment/risk_rcra.htm (accessed March 2, 2009).

[15] "ORNL MVST Sludge (SL) Solidification Feasibility Study Overview," Presented at Slurry Retrieval, Pipeline Transport & Plugging and Mixing Workshop, January 14-18, 2008. Available at http://www.em.doe.gov/ (accessed March 2, 2009).

Because various environmental regulations may apply, the USDOE and the various Sites have developed strategies to integrate actions under the various laws including CERCLA, RCRA, and NEPA (Cook 2002; Shedrow et al. 1993). Policy dictates that 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 1994a). For example, a strategy for integrating NEPA requirements and combined RCRA/CERCLA programs for remedial actions at the Savannah River Site (SRS) was developed (Shedrow et al. 1993). The SRS strategy tiers RCRA/CERCLA activities to NEPA reviews and integrates elements of the NEPA and RCRA/CERCLA processes, where applicable.

Under the NEPA/CERCLA policy, USDOE relies on the CERCLA process for review of actions taken under CERCLA—no separate NEPA process is typically required (Cook 2002)[16]. In the CERCLA process, USDOE addresses NEPA values (e.g., analysis of cumulative, off-site, ecological, and socioeconomic impacts), includes a discussion of these impacts in CERCLA or other environmental documents, and takes steps to ensure early public involvement in the process.

The USDOE approach to NEPA review for RCRA corrective actions tends to be project-specific[17]. Most USDOE RCRA actions have fallen within the scope of a categorical exclusion (Cook 2002). In the instances where proposed RCRA actions have not qualified for a categorical exclusion, USDOE has often been able to rely on the CERCLA process when the corrective action was taken under a compliance agreement that largely integrates the CERCLA and NEPA requirements.

2.2.3.1 Assessment Related Requirements under NEPA

Every federal agency in the executive branch has a responsibility to implement NEPA[18]. The Congress directed that the U.S. policies, regulations, and laws shall be interpreted and administered in accordance with the policies set forth in NEPA and prescribed a procedure, commonly referred to as “the NEPA process” for its implementation (CEQ 2007). The typical NEPA process is outlined in Figure 4.

The NEPA process begins when a federal agency needs to take an action; the need may be something the agency identifies itself, or it may be identified by someone outside of the agency (CEQ 2007). Based on the need, the agency develops a proposal for the action (Number 1 in Figure 4). The agency will then enter the initial analytical stage (Number 2 in Figure 4) to help determine whether it will pursue one of the following paths (CEQ 2007):

CATegorical EXclusion (CATEX) (Number 3 in Figure 4) is a category of actions that are deemed to not have a significant effect (either individually or cumulatively) on the quality of the human environment (Bear 1989). This category is primarily based on experience with a particular kind of action and its effects. For example, similar actions may have been studied previously and found to have no significant impact after implementation. When there is uncertainty as to the environmental impacts of the proposed action, the agency should prepare an EA. Although actions are often categorized as CATEX based on

[16] The basis for this policy is a U.S. Department of Justice determination that there is a statutory conflict between NEPA and CERCLA, and that NEPA, as a matter of law, does not apply to CERCLA cleanups (Cook 2002). Whereas NEPA allows judicial review before an agency takes action, CERCLA generally bars such “pre-enforcement” reviews (Cook 2002).

[17] The U.S. Department of Justice has not determined that RCRA corrective actions are not subject to NEPA, so DOE has not been able to establish a broad RCRA/NEPA policy paralleling its CERCLA/NEPA policy (Cook 2002).

[18] However, NEPA does not apply to the President, to Congress, or to the Federal courts (CEQ NEPA Regulations 40 CFR §1508.12).

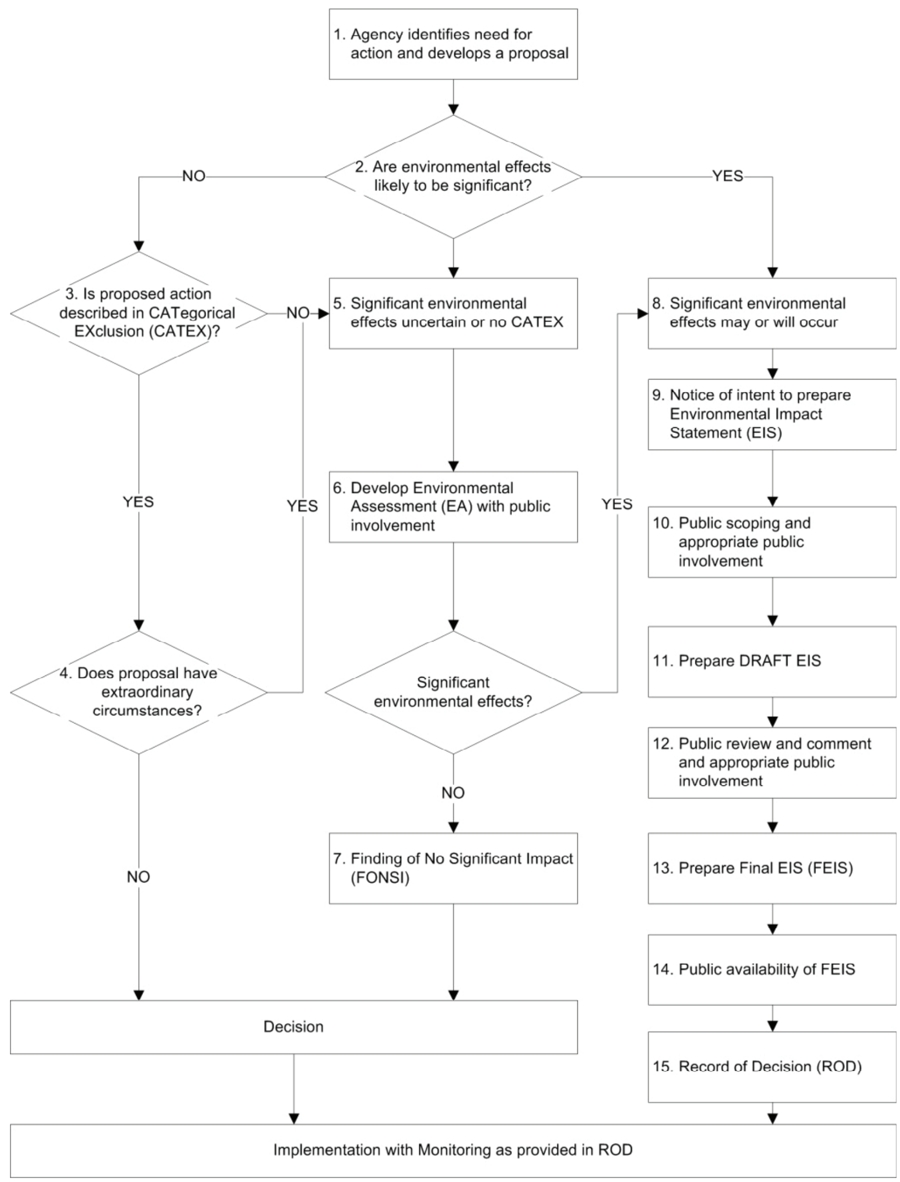


Figure 4. The National Environmental Policy Act (NEPA) Process (CEQ 2007)

experience with previous actions, predictive models can be used in the original CATEX designations for actions or needed for actions for which previous actions may not be pertinent[19].

Environmental Assessment (EA) (Number 6 in Figure 4) is intended to be a brief public document used to determine the significance of the environmental effects and to examine alternative means to achieve stated objectives. The objectives of the EA are to: provide sufficient evidence, which may be in the form of or supported by modeling results, for determining whether to prepare an EIS, aid compliance with NEPA when no EIS is needed (i.e., resulting in a Finding of No Significant Impact (FONSI) (Number 7 in Figure 4)), and facilitate preparation of the EIS (Bear 1989).

Environmental Impact Statement (EIS) (Number 8 in Figure 4) must be prepared if a major federal action is proposed that may significantly affect the quality of the human environment and acts as an action-forcing vehicle to ensure that NEPA policies are integrated into ongoing Federal programs and actions (Bear 1989). Once the decision is made to prepare an EIS (and the Notice of Intent is published), the agency then engages in a "scoping process" (Number 10 in Figure 4) to determine the scope of the EIS and the problems to be addressed. A draft EIS is prepared (Number 11 in Figure 4) for comment with contents set out under the CEQ regulations (40 CFR §1502.10) (Bear 1989). The focal point of the EIS is the alternatives analysis, which includes the determination of which alternatives are analyzed, with the judicial standard being that of "reasonableness". The analysis of alternatives frequently requires the use of computer models, some even quite complex, to predict the impact of the actions being studied. Substantive comments are addressed in the final EIS (Number 13 in Figure 4), which is made available to the public.

The Record of Decision (Number 15 in Figure 4) is the final step in the EIS process. The ROD states the decision, identifies the alternatives that were considered, and discusses mitigation plans (CEQ 2007). Through the NEPA process, Federal agencies are required to determine if their actions may have significant environmental effects and to consider the environmental and related social and economic effects of their actions (CEQ 2007).

2.2.3.2 Requirements Related to Assessment of Cementitious Barriers under NEPA

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. However, NEPA requires that all "reasonable" alternatives be considered during the EIS process; this process is when alternatives including barriers or grouting may be considered for action and evaluation[20]. Credit can be taken for waste forms and barriers when predicting exposure media concentrations and corresponding risks although any increases in modeling complexity and uncertainty must be taken into account in the decision-making process. One goal of the CBP is to allow more accurate predictions to be made when cementitious barriers are used in disposal actions.

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

[19] As an example of the use of models in designating an action as an CATEX, see Town of Marshfield v. Federal Aviation Administration, 2008 U.S. App. LEXIS 25410, 2008 WL 5251104, No. 07-2820 (1st Cir. 12/18/08).

[20] For example, SRS is considering grouting as one alternative (and the preferred alternative at that) for closure of the 49 remaining high-level waste (HLW) tanks on site (USDOE 2002). Two SRS tanks (i.e., 17F and 20F) were operationally closed by filling them with grout under South Carolina Department of Health and Environmental Control (SCDHEC) industrial wastewater permits. These operational tank closures are described in the examples.

other hand, Final EISs, the focal point of which is a detailed analysis of the potential impacts of proposed actions, were examined for the SR, Hanford, and Idaho Sites. Of the Final EISs identified in Table 1, 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).

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

The U.S. Nuclear Regulatory Commission (USNRC), which was established by the Energy Reorganization Act of 1974, grants licenses to companies for the commercial operation of nuclear reactors and radiological facilities. Any company holding such a license must seek USNRC permission to decommission a commercial facility. The general decommissioning process is illustrated in Figure 5. The USNRC does not have regulatory authority over defense nuclear facilities.

For a power reactor, a Post-Shutdown Decommissioning Activities Report (PSDAR) must be submitted either before or within two years following cessation of operations[21]. Among other requirements, the PSDAR must include a discussion describing how environmental impacts from decommissioning activities will be bounded by pertinent environmental impact statements. For a power reactor, the licensee must submit an application for termination of its license for USNRC approval and be accompanied or preceded by a license termination plan (LTP), which must include[21]:

- A site characterization,
- Identification of remaining dismantlement activities and estimate of remaining decommissioning costs,
- Plans for site remediation and for the final radiation survey,
- A description of the end use of the site, if restricted, and
- A supplement to the environmental report describing any new information or significant environmental changes from the proposed termination activities.

The licensee must also demonstrate that the requirements of the License Termination Rule (LTR) (10 CFR §20.1401 et seq.) will be met. For a reactor, decommissioning must be completed within 60 years of the cessation of operations unless otherwise approved.

For a radiological material site licensed by the USNRC, a decommissioning plan (DP) is submitted to the USNRC if required or if the activities have not been previously approved and could increase health and safety impacts. Once the licensee demonstrates compliance with its decommissioning plan, it must then request license termination from the USNRC either for unrestricted or restricted release (where controls remain in place)[22]. For unrestricted release, a full technical review guided by NUREG-1757 (USNRC 2003a; USNRC 2003b; USNRC 2003c)[23] is undertaken with results documented in an Environmental Assessment (EA) and a Safety Evaluation Report (SER). The EA process is carried out as described in the previous section on NEPA.

[21] See Decommissioning Process at http://www.nrc.gov/about-nrc/regulatory/decommissioning/process.html (accessed March 6, 2009).
[22] New Jersey v. USNRC, Nos. 06-5140, 07-1559, 07-1756, JORDAN, Circuit Judge, concurring available at http://www.ca3.uscourts.gov/opinarch/065140p.pdf (accessed March 6, 2009).
[23] The NRC consolidated numerous guidance documents into a single, three-volume document (NUREG-1757) describing how to satisfy the license termination requirements by means acceptable to the NRC (USNRC 2003a; USNRC 2003b; USNRC 2003c).

Table 1. Final Environmental Impact Statements Related to the Savannah River, Hanford, and Idaho Sites

(http://www.gc.doe.gov/NEPA/final_evironmental_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

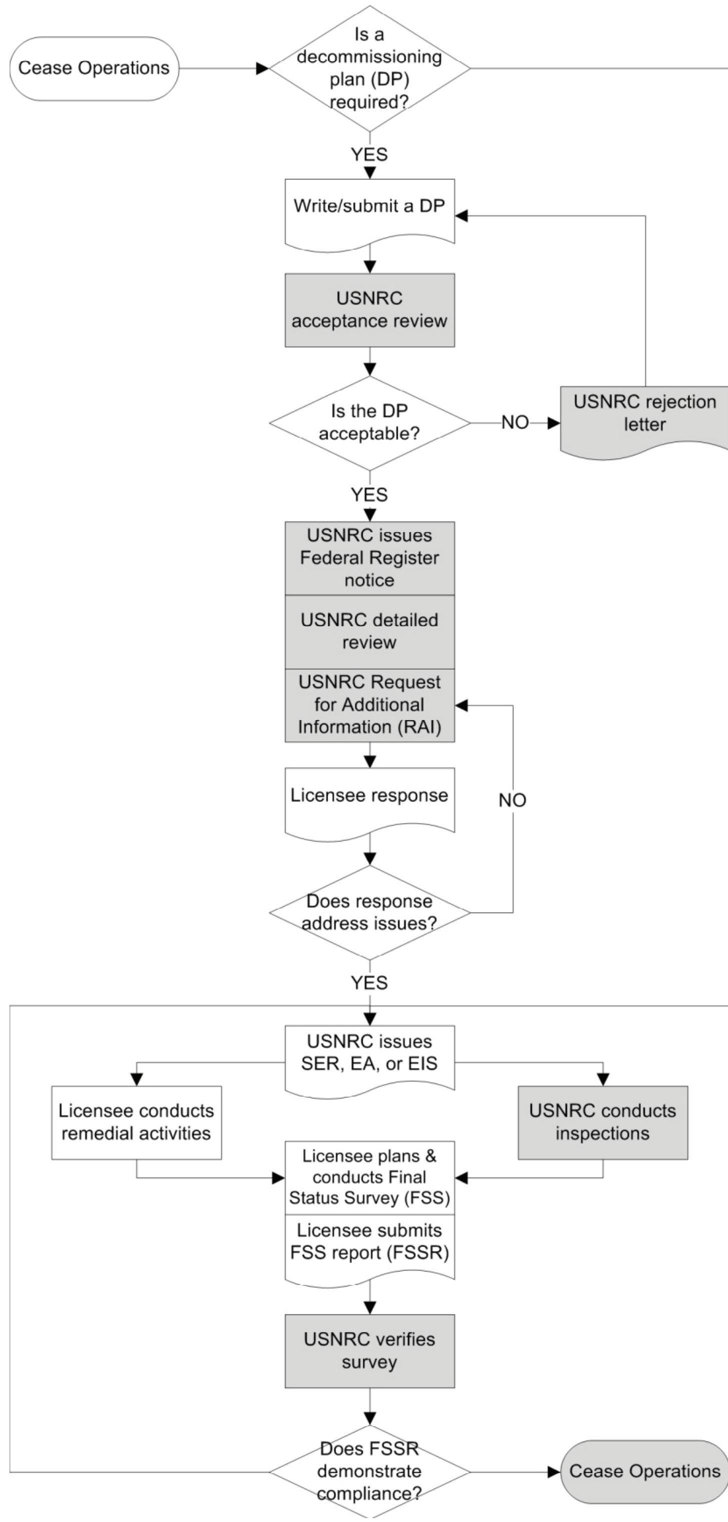


Figure 5. The USNRC Decommissioning Process (after USNRC 2003a)

For plans proposing restricted release for material sites, the review is conducted in two phases[24]. 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. The second phase of the review addresses the rest of the technical review as guided by NUREG-1757 and includes developing an EIS. The EIS process is also carried out as described in the previous section on NEPA. Following approval of the DP by the USNRC, the licensee must complete decommissioning activities within 24 months or apply for an alternate schedule. In general, the decommissioning process illustrated in Figure 5 for fuel cycle facilities is the same as for material sites.

2.2.4.1 Assessment Related Requirements

The primary type of evaluation required under the License Termination Rule (LTR) (10 CFR §20.1401 et seq.) is the assessment of 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)[25]. A site is acceptable for unrestricted release if the residual radioactivity, upon reduction to levels that are as low as reasonably achievable (ALARA)[26], translates to a total expected dose equivalent (TEDE) to an average member of the critical group that does not exceed 0.25 mSv (25 mrem) per year including that from groundwater sources of drinking water (10 CFR §20.1402).

A site will be considered acceptable for restricted release if the licensee meets certain conditions (10 CFR §20.1403(a)-(e)) including provisions of not increasing net public or environmental harm from the proposed actions or "legally enforceable institutional controls" to protect the public by restricting future land use. 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). Typically predictive models are important if not critical to supporting the license termination process in terms of dose assessment calculations.

2.2.4.2 Guidance for Cementitious Barriers

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. However, unlike these laws administered by the USEPA, 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 USNRC suggests employing probabilistic analyses (USNRC 2003a)[27]. Point-value analyses may be inadequate in these cases. For simpler, low-risk sites and those with short-lived radionuclides,

[24] A license can be terminated for restricted release only after the licensee has met certain conditions including "legally enforceable institutional controls" to protect the public (10 CFR §20.1403(a)-(e)). The USNRC License Termination Rule (10 CFR §20.1401 et seq.) sets forth decommissioning requirements.
See New Jersey v. NRC, Nos. 06-5140, 07-1559, 07-1756 described in footnote 22.

[25] As illustrated above, the decommissioning process may involve development of either environmental assessments or environmental impact statements or both.

[26] ALARA determinations must take into account consideration of detriments expected to potentially result from decontamination and waste disposal.

point-value analysis with sensitivity analysis may be sufficient (USNRC 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 (USNRC 2003a). The behavior of the barrier should be considered an evolving component of a larger, dynamic ecosystem (Waugh et al. 1997). Table 2 summarizes selected guidance and reference reports that may have relevance to the application of engineered barriers at decommissioning sites (USNRC 2003a).

The USNRC 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 et al. 1997). Concrete degradation mechanisms (e.g. sulfate attack, chloride corrosion, 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-based materials 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 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.

2.3 Spent Fuel Pools

When removed from a reactor, spent fuel is placed in a spent fuel pool to allow the fuel to cool and decay. Spent fuel pools are typically 40-foot deep, steel-lined, concrete vaults filled with water, which is a natural barrier to radiation (USGAO 2005). Over time, spent fuel in the pools is typically rearranged to accommodate additional fuel while maintaining safety. Some spent fuel has been transferred to dry storage casks to await permanent disposition at a national repository. Spent fuel is cooled for at least five years before it can be moved to dry storage casks (USGAO 2005).

Spent nuclear fuel can be stored in a water filled spent fuel pool as regulated under 10 CFR Part 50 (Domestic Licensing of Production and Utilization Facilities)[28]. Technical challenges include removing the spent fuel decay heat, storing the fuel in an arrangement to avoid criticality, and providing shielding (USNRC 2006a). The USNRC requires in 10 CFR §50.68 that spent fuel pools remain subcritical in an unborated, most adverse moderation condition, but allows credit for fuel burnup when analyzing the storage configuration of the spent fuel (USNRC 2006a). Because burnup can be accounted for in these evaluations, predictive modeling is important to the regulation of spent fuel pools.

SFP structures, systems and components (SSC) are designed to accomplish the following tasks:

- Prevent loss of water from the fuel pool that would lead to water levels that are inadequate for cooling and shielding.
- Protect the fuel from mechanical damage.

[27] Point value methods are suggested for selecting the design basis flood for the development of long-term erosion controls (USNRC 2003a).
[28] The USNRC has promulgated regulations governing spent fuel pools (10 CFR Parts 50 and 70). The pertinent USNRC URL is http://www.nrc.gov/waste/spent-fuel-storage/pools.html.

Table 2. 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.

Table 2. Summary of Selected Reports Related to Engineered Barriers (reproduced from USNRC 2003a) (contd)

Report	Brief Summary
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.
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's waste sites and identifies characteristics and design criteria for effective long-term institutional management.

- Provide the capability to limit potential offsite exposures from a significant release of radioactivity from the fuel or significant leakage of pool coolant.
- Provide adequate cooling to the spent fuel to remove residual heat.

General Design Criterion (GDC) 61, "Fuel Storage Handling and Radioactive Controls", Appendix A to 10 CFR Part 50, requires that fuel storage and handling systems be designed to ensure adequate safety under anticipated operating and accident conditions. These include (i) periodic inspections, (ii) suitable radiation shielding, (iii) appropriate containment, confinement, and filtering systems, (iv) residual heat removal capability consistent with its importance to safety, and (v) prevention of significant reduction in fuel storage inventory under accident conditions.

The SFP design basis is also covered by GDC 2, "Design Basis for Protection against Natural Phenomena", GDC 4, "Environmental and Dynamic Effects Design Basis", and GDC 63, "Prevention of Criticality on Fuel Storage and Handling", as described in Appendix A to 10 CFR Part 50. Regulatory guide 1.26, "Quality Group Classification and Standards for Water, Steam and Radioactive Waste Containing Components of NPP", and regulatory guide 1.29, "Seismic Design classification", detail the quality groups and seismic categories applicable to the design of SFPs.

Applicable sections of the regulations include Appendix S, "Earthquake Engineering Criteria for NPP," to 10 CFR Part 50 and Regulatory Guide 1.92, "Combining Modal responses and Spatial Components in Seismic Response Analysis," that provide guidance on seismic response analysis. The facilities are protected against extreme winds and where tornadoes cause the strongest wind, Regulatory Guide 1.76, "Design Basis Tornado for NPP" provides guidance for design basis tornado characteristics. Regulatory Guide 1.115, "Protection Against Low-Trajectory Turbine Missiles" provides guidance on the protection of spent fuel storage facilities against turbine missiles while Regulatory Guide 1.52, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered Safety Feature Atmosphere Cleanup Systems in Light water Cooled NPP," discusses safety of the SFP filtration systems.

Other design feature requirements include requirements for the control of crane loads acting above the SFP, drainage prevention provisions, instrumentation, additional water to add coolant to the SFP, pool cooling to maintain a temperature below 60 degrees centigrade, the design of gates and weirs to isolate the SFP from adjacent fuel handling areas, enabling fuel cooling for all stored fuel assemblies, providing leakage containment, ensuring pool cleanup to maintain low radiation levels, and provisions to protect high burn-up fuel from mechanical damage.

As suggested above, one analysis required specifically for spent fuel pools is to evaluate the storage configuration to assure that criticality is not a concern (10 CFR §50.68). Assessments for spent fuel pools, especially those for decommissioning, are similar to if not part of those for commercial reactors licensed by the USNRC.

3.0 CEMENTITIOUS BARRIER PA MODELING APPROACHES

A variety of different modeling approaches have been used to address cementitious barriers. Approaches range from taking no credit for the cementitious materials to detailed modeling to support assumptions about the evolution of chemical and physical properties. The USDOE has a need for a better understanding of cementitious barrier performance and better approaches for long term modeling to support decisions in the different regulatory environments described in the previous section. One goal of the examples in this section is to illustrate how modeling has been implemented in the different environments to encourage improved sharing of information.

The emphasis of this section is on examples of PAs from USDOE disposal facilities. The examples in this section are organized by general climate at the DOE Sites of interest (i.e., arid, semi-arid, and temperate). This arrangement reflects the emphasis placed on the groundwater pathway. Thus, the importance of cementitious barriers is related to climate and amount of infiltration at a given site.

Because of arid climates, water infiltration into waste forms is not a concern in some DOE facilities. For example, at the Nevada Test Site (USNTS) the mean annual precipitation of 12 cm is greatly exceeded by the annual potential evapotranspiration, typically about 150 cm/yr (See Figure 6). The migration of groundwater, when there is any because of rain, is down for a small distance and then upward. The depth of the saturated zone is about 240 m. Samples of corings, by way of chloride content, show that no surface water has reached the deep saturated zone in many thousands of years. Because there is no mechanism to transport the contaminants to the groundwater, cementitious barriers are not used at the NTS.

The Los Alamos National Laboratory (LANL) is nearly as dry as the NTS. For many of the same reasons as listed above, LANL has not credited cementitious materials for engineered barriers in its PAs.

Two DOE facilities, the Hanford Site and the Idaho National Laboratory, are as dry as the Nevada and Los Alamos sites, but have water tables somewhat closer to the surface and have some significant waste streams that pose potential risks without consideration of cementitious barriers. Therefore, cementitious barriers are considered in the PAs for these sites.

Figure 7 illustrates a conceptual model used to evaluate degradation of cementitious grout used to physically stabilize and isolate residual waste tanks at the Idaho site and could generally applied to any of the disposal facilities mentioned in this section.

The Oak Ridge and Savannah River Sites are located in more temperate climates with greater infiltration and water tables located much closer to the ground surface. In these environments, the groundwater pathway tends to be a more significant contributor to the PA.

Examples from these DOE sites are provided in the following sections.

3.1 Idaho Site

3.1.1 Tank Farm Facility Performance Assessment (INL)

A Performance Assessment (USDOE-ID 2003) was performed to assess the projected radiological dose impacts associated with the closure of the Tank Farm Facility (TFF) at the Idaho Nuclear Technology and Engineering Center (INTEC) at the Idaho National Laboratory (INL) Site in the southeastern part of the State of Idaho. Section 3116 was the regulatory framework for the assessment. The INL Site is a semi-arid site with roughly 22 cm/yr of precipitation. The water table at INTEC is roughly 450 ft below the ground surface. The TFF is a collection of 15 below-ground stainless-steel tanks. The eleven 300,000-gal tanks are enclosed in belowground concrete vaults (see Fig. 8), while the four 30,000-gal tanks are directly buried in the soil. The tanks were used for storage of HLW from operations at INTEC.

3.1.1.1 Role of Cementitious Barriers and Processes Considered

The concrete and grout are assumed to function as physical and chemical barriers controlled by the assumed hydraulic conductivity and distribution coefficients. These properties are assumed to change with time, in general degrading the performance of the materials as barriers over time. Figure 7 is an illustration of the physical degradation assumptions. Cracks

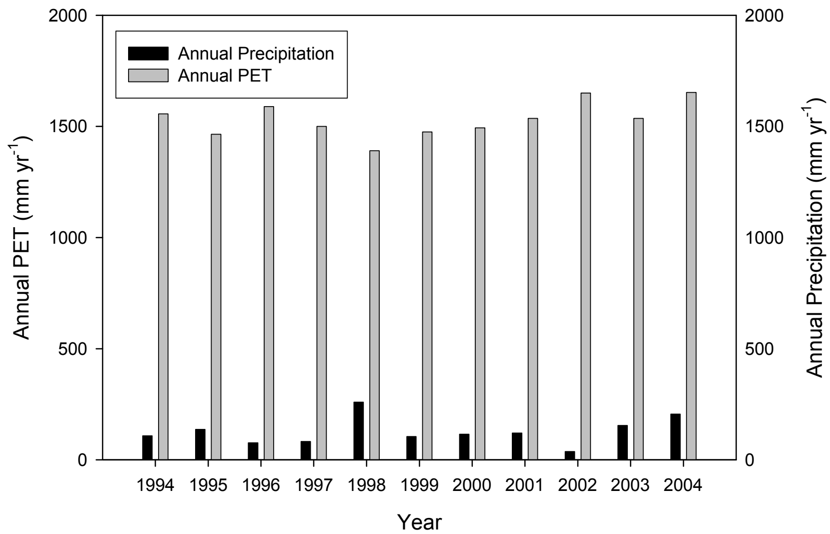


Figure 6. Nevada Test Site Area 5 Annual Precipitation and Potential Evapotranspiration[29]

are not specifically modeled but are represented as a step change in the bulk hydraulic conductivity of the porous media.

3.1.1.2 Parameter Assumptions and Conceptual Models

A detailed analysis of cementitious materials degradation was performed for the INTEC Tank Farm. The detailed analysis included consideration of sulfate and magnesium attack, carbonation, and calcium hydroxide leaching. Reinforcement corrosion in the outer vault concrete and general corrosion of the steel tank were also modeled. The effects of acid attack, alkali-aggregate reaction, and corrosion of the pipes on the concrete and grout degradation were assumed to be insignificant compared to the three modeled chemical attacks. The DUST-MS computer code was used to model releases from the engineered features. The degradation mechanisms were modeled using a number of different algorithms, which are documented in detail in Appendix E of the PA.

The base case degradation model results indicated that maximum degradation, which is from reinforcement corrosion caused cracks in the concrete. The assumption was that the outer vault to turn to rubble after about 500 years. Once the outer vault completely degrades, the grout between the vault and the tank was also assume to be rubble at approximately 5,000 years. The concrete tank and grout fill in the tank was assume to completely degrade and turn to rubble after about 40,000 years. The grout associated with the piping turned to rubble after about 500 years. The predominant chemical attack on the grout was caused by reactions with sulfate and magnesium ions. Sensitivity analyses indicated the times for the four zones to turn to rubble can vary greatly, i.e., from tens of years to tens of thousands of years or beyond.

The result of the detailed analysis was used to provide a basis for a set of conservative assumptions regarding degradation of the material properties. The material initially had properties suitable for intact conditions but was assumed to have properties of the native

[29] From "Special Analysis of Transuranic Waste In Trench T04C at the Area 5 Radioactive Waste Management Site, Nevada Test Site, Nye County, Nevada Revision 1.0, DOE/NV/25946-283, March 2008.

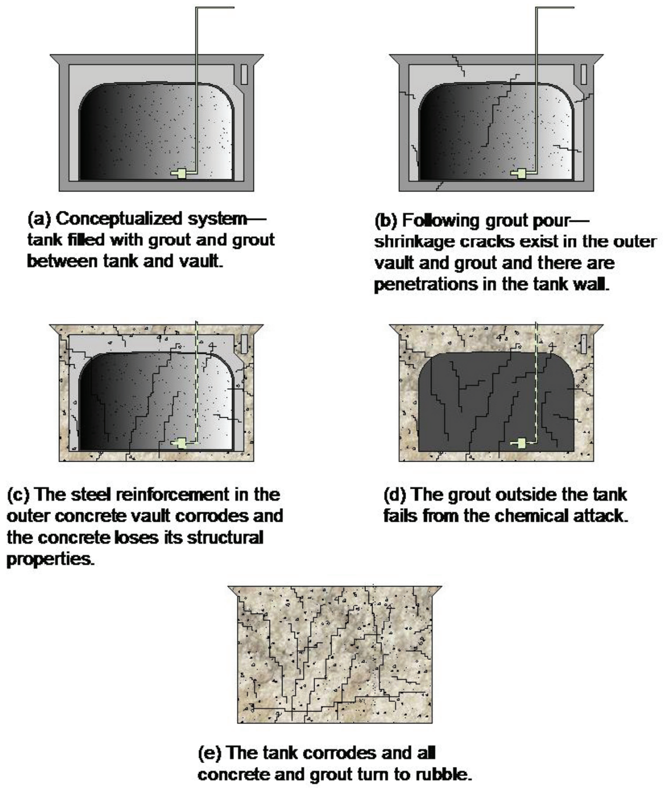


Figure 7. Conceptualization of the Degradation Sequence (a) to (e) for a Closed Tank Farm Facility (USDOE-ID 2003)

Figure 8. INL Tank WM-185 Vault Dome, Support Beams, and Risers (USDOE-ID 2003)

soil when degraded. For instance, the time of the step change in the parameters for the vault, grout between the vault and tank, piping, and the steel tank filled with grout were assumed to be 100, 100, 500, and 500 years, respectively. These degradation step changes were within the ranges predicted by the degradation analysis, even for the minimum degradation case, which indicated 100, 500, 1,750, and 1,750 years, respectively.

Conservative assumptions used in the degradation analyses included: (1) determining the rates of modeled chemical attacks from experiments involving degraded instead of intact concrete and (2) not taking credit for the chemical barrier provided by the chemically-reducing grout, vault, and tank. Only the physical barrier to flow and transport was modeled, while the chemical barrier was represented with a distribution coefficient, which was more significant in affecting radionuclide release rates.

In addition, the vault failure, modeled as occurring from the expansion reaction caused by reinforcement corrosion, was assumed to accure at 100 years after closure, but no credit was taken for the non-aggressive corrosion environment surrounding the vault. Finally, the corrosion rates that were determined for coupons placed in the tank liquid and used in corrosion rate calculations were expected to be greater than corrosion rates in the grouted tank and piping and in the water contacting the vault wall.

From a chemical perspective, reducing conditions were assumed to be maintained for both the grout and the concrete. The K_ds assumed for cementitious materials were:

Sr $K_d = 0.006\ m^3/kg$, range from 0.001 to 0.006
Tc $K_d = 5\ m^3/kg$, range from 1 to 5
I $K_d = 0.03\ m^3/kg$, range from 0.002 to 0.03
C $K_d = 10\ m^3/kg$, range from 1 to 10

The detailed analysis showed that no change in chemical properties was expected until long past the 1,000 years of the compliance period.

3.1.1.3 Relative Importance in Context of Assessment

In context of the assessment, the most important factor was the reducing environment steel tank of the cementitious material, which was used as the basis for assumed K_d values. The assumed failure times were demonstrated to be conservative based on a number of sensitivity cases. Thus, although it was concluded that additional credit could be taken for longer performance of the physical barriers, the conclusions regarding compliance were not sensitive to changes in the timing of the physical degradation.

3.1.2 Radioactive Waste Management Complex (INL)

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 (USDOE/NE-ID 2007). The facility is located within the historic radioactive waste "burial grounds" and thus the inventories are also included in the CERCLA assessment for the RWMC. 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.

3.1.2.1 Role of Cementitious Barriers and Processes Considered

Figure 9 shows concrete vaults at the INL Radioactive Waste Complex, but the PA for this site (USDOE/NE-ID 2007) includes the assumption that a

Figure 9. INL Radioactive Waste Management Complex Active LLW Disposal Facility within the Subsurface Disposal Area (May 2005)

representative elementary volume can be defined that allows the waste to be described as a porous medium. The buried waste forms, such as metal drums, metal, concrete, and wooden boxes, soft-sided containers, and a variety of specialized containers, challenge this assumption. Given the scale of the area being represented in the numerical model (i.e., the entire Active LLW Disposal Facility), it is not practical to consider the waste as having hydrological properties different from the surface sediments.

In general, cementitious barriers are not credited hydraulically in the RWMC performance assessment. However, some credit was taken for diffusion-controlled migration of radionuclides through selected concrete containers. This approach is discussed in the following section.

3.1.2.2 Parameter Assumptions and Conceptual Models

As stated above, at the RWMC cementitious barriers are generally treated hydraulically as being the same as the surrounding porous media. The performance of the concrete as a barrier was treated as a diffusion problem only for cask containers. The chemistry of the cementitious barriers was not considered to change during the course of the simulation. The distribution coefficients were, therefore, considered not to change.

Concrete casks were not modeled with an assumed failure time. Instead, release of contaminant mass from within the casks was modeled as diffusion out of the cask. Casks were modeled as cylinders with a 15-cm (6-in.) wall thickness. Using this thickness assumption allowed the ready release of contamination at the surface of the cask. In addition, a diffusion coefficient of 10^{-6} cm^2/s was used. A diffusion coefficient of 10^{-6} cm^2/s is typical for a metal ion in water and does not account for the possible partitioning of the contaminant within the waste form or the tortuosity of the porous media. Partitioning and travel through a tortuous path would slow the contaminant release.

3.1.2.3 Relative Importance in Context of Assessment

Although, some credit was taken for diffusion through the concrete container, the analysis illustrated that even with conservative diffusion assumptions acceptable performance was obtained. Thus, consideration of the performance of cementitious materials was not an important factor in the RWMC PA. This is largely related to the semi-arid conditions and the depth of the water table at the INL Site.

3.2 Hanford Site

3.2.1 Integrated Disposal Facility

The Hanford Site is located in southeastern Washington State in a semi-arid climate. The Integrated Disposal Facility (IDF) at the Hanford Site consists of a single landfill with two adjacent, expandable cells. One cell is permitted as a RCRA Subtitle C compliant landfill system. The other cell contains waste not governed by RCRA. A performance assessment has been conducted for both cells of the IDF (Mann et al. 2005). The performance assessment addressed a number of different waste forms including glass and grouted wastes. This example focused on the assessment of the grouted waste form.

3.2.2.1 Role of Cementitious Barriers and Processes Considered

The IDF PA assumed that all "treated" waste has been grouted prior to disposal. The grout is assumed to form an effective barrier to infiltrating moisture. Therefore, the dominant release mechanism from a grouted waste form is the diffusion of the contaminant through the grout to the waste package surface. Once the contaminant is available on the package surface, it becomes available for transport in/with the infiltrating water.

3.2.2.2 Parameter Assumptions and Conceptual Models

The near-field numerical model calculations for the grouted waste form assume the contaminant flux into the far-field numerical model can be approximated by an analytical solution for the contaminant release at the bottom of the IDF trench. The waste form release rate for treated solid waste assumed all treated waste is encapsulated in a grouted waste form where the contaminant release mechanism is dominated by diffusion from the waste package. Assumed most probable and conservative diffusion coefficients for key contaminants are summarized in Table 3.

This calculation approach neglects any chemical interactions with the surrounding backfill materials and other waste packages in the trench. The calculation approach also neglects the transport time associated with the recharge through the trench.

3.2.2.3 Relative Importance in Context of Assessment

Similar to the case for the RWMC PA, little credit was taken for the performance of cementitious materials. The cementitious waste forms in the Hanford IDF PA are relatively unimportant when compared to the vitrified waste included in this analysis. It appears that much work had been done on the vitrified waste to define its release mechanisms with some degree of accuracy. The conclusion was made that it was anticipated that improved grout waste forms would be developed and used for the actual disposal, thus the conclusions of the PA should be bounding for the cementitious waste forms.

3.3 Oak Ridge National Laboratory

3.3.1 Solid Waste Storage Area 6

The Solid Waste Storage Area (SWSA) 6 has accepted waste since 1969 (ORNL 1997). It has been the only active waste disposal facility for ORNL-generated wastes since 1973. Approximately 12 hectares (30 acres) of the site is still useable for waste disposal operations, with most of the total site capacity having been used prior to September 26, 1988. Prior to September 1988 a variety of disposal methods were used at SWSA 6, with the bulk of waste materials buried in shallow, unlined trenches. Wastes disposed of since that time have been placed in excavated trenches for biological materials; below-grade, concrete-lined, silos for bulk waste materials;

Table 3. Hanford IDF PA Effective Diffusion Coefficients for Cementitious Waste Forms (after Mann et. al 2005)

		Effective Diffusion Coefficient	
Waste Form Species	**Waste Form Type**	**Most Probable (base case)**	**Conservative**
		cm^2/s	
NO_2^-, NO_3^-	any cement/grout	5×10^{-9}	3×10^{-8}
I^- (free), IO_3^- (free)	any cement/grout	2.6×10^{-9}	1×10^{-8}
TcO_4^-	any cement/grout	5×10^{-10}	1×10^{-8}
Cr(VI)	any cement/grout	5×10^{-11}	5×10^{-10} (guess)
Hg(I) free	any cement/grout	1×10^{-11} (guess)	1×10^{-10} (guess)
U(VI)	aged cement/grout	1×10^{-11} (guess)	1×10^{-10} (guess)

Note: All other contaminants assumed to have effective diffusion coefficient of 5×10^{-9} cm^2/s

engineered wells for fissile and high "range" (>200 mrem/hr) materials; and tumulus disposal units for containerized wastes. ORNL is located in a temperate environment with the water table relatively close to the base of the disposal facility.

3.3.1.1 Role of Cementitious Barriers and Processes Considered

The cementitious barriers are considered as both barriers to infiltration and release and as adsorbing media. Although a relatively detailed approach was adopted for the degradation calculations, a conservative assumption was made, i.e., the cementitious barrier was assumed to lose all capability as a physical barrier at the time through cracks form. Sensitivity analysis results indicated that sulfate attack was the primary degradation mechanism for the cementitious barriers.

3.3.1.2 Parameter Assumptions and Conceptual Models

A rather involved analysis of the cementitious barriers was used in the SWSA PA (ORNL 1997). Figures 10 and 11 illustrate the conceptual approach used for the SOURCE computer programs. The SOURCE programs are used to conduct structural and degradation calculations in a relatively detailed manner. As illustrated in the figures, hydroxide leaching, sulfate attack, reinforcement corrosion, and the associated cracking of the cementitious materials are calculated to determine the onset of cracking.

The chemical aspect of the cementitious barriers performance is based on a linear isotherm K_d model and diffusion coefficients, which are invariant with time but change when the overall barrier is assumed to change conditions. Solubility controls are also applied

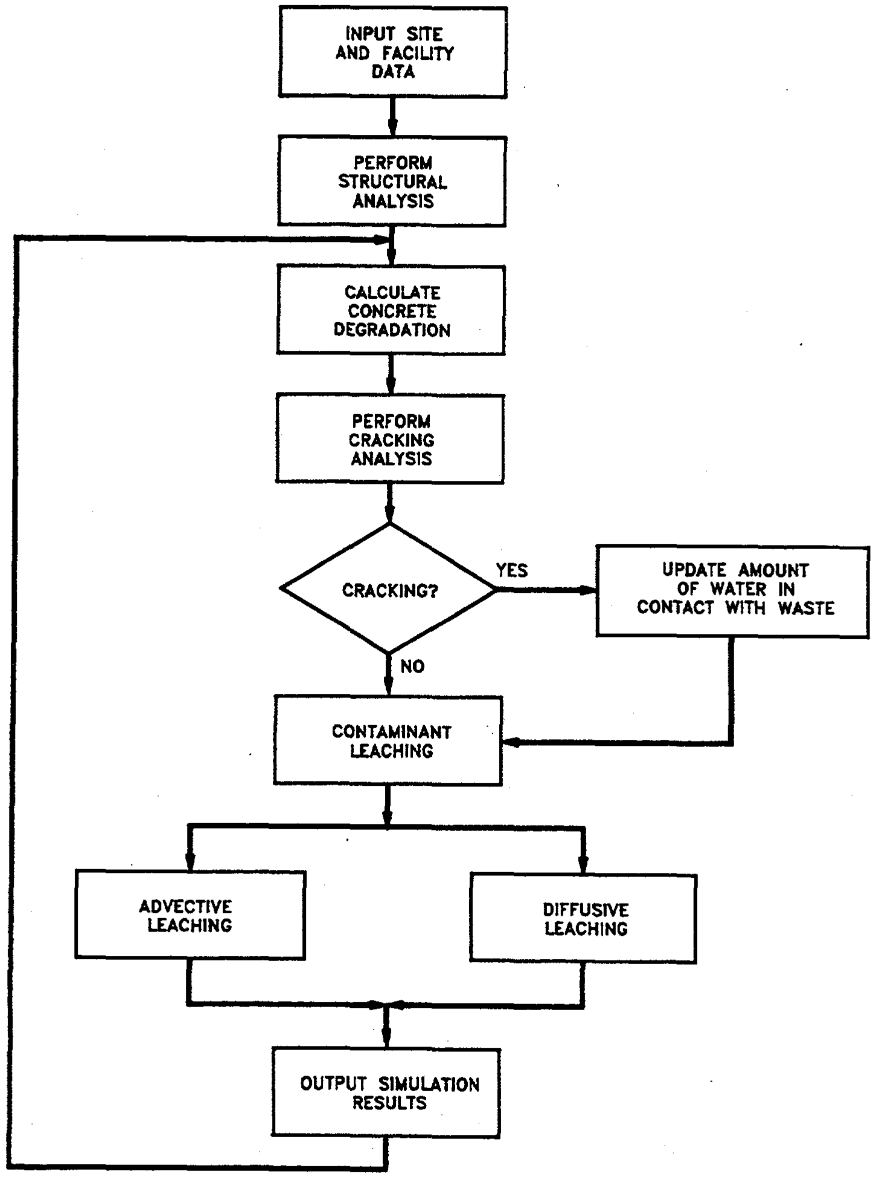


Figure 10. Overall Logic Flow in the Oak Ridge PA SOURCE Computer Codes (ORNL 1997)

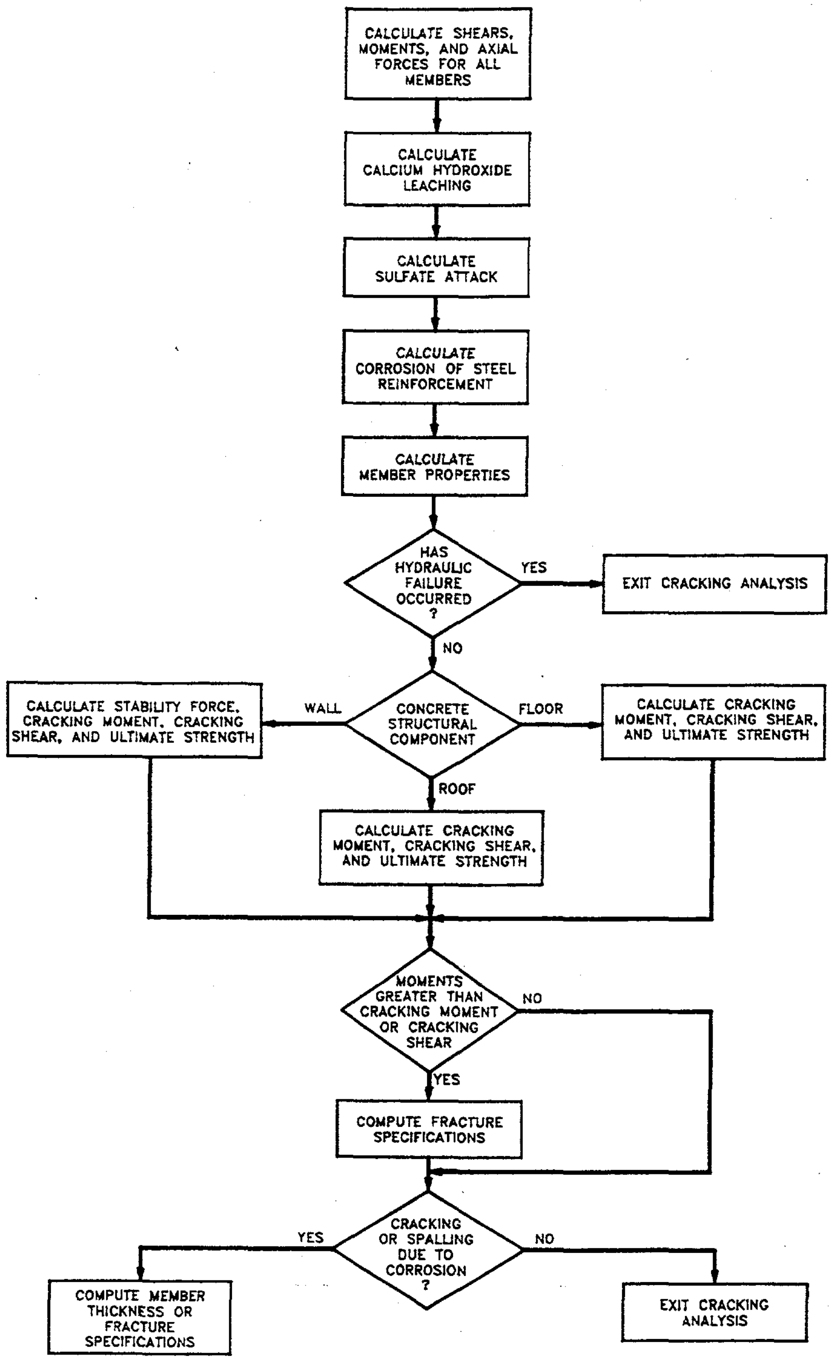


Figure 11. Detailed Logic Flow for the SOURCE1 and SOURCE2 Computer Programs (ORNL 1997)

as appropriate. No credit is taken for additives that could improve the chemical performance of cementitious barriers (e.g., slag or similar additives to create a reducing environment).

3.3.1.3 Relative Importance in Context of Assessment

Cementitious materials were an important feature for the overall performance of the disposal facility. The grout inside of vaults, the thickness of vault walls, and the timing of failure of the concrete pad and collection system all were shown to influence the results of the assessment. This is reflected by the relatively high level of rigor applied to the calculations related to degradation of cementitious materials. Although double thickness walls were considered, only single thickness walls were used as a basis for the final results, which illustrates that even for a site that depends on cementitious barriers, some room for conservatism remains. The authors concluded that further efforts to refine the design and analysis of cementitious barriers could potentially result in improved estimates regarding disposal capacity.

3.4 Savannah River Site

3.4.1 F-Tank Farm

The F-Tank Farm (FTF) is in the north-central portion of the Savannah River Site (SRS) and occupies approximately 22 acres within F-Area. The FTF is an active radioactive waste storage facility consisting of 22 carbon steel waste tanks and ancillary equipment such as transfer lines, evaporators and pump tanks. The FTF stores and processes liquid radioactive waste generated primarily from the Plutonium Recovery and Extraction (PUREX) process. FTF began radioactive operations in 1954. Two of the 22 tanks (Tanks 17 and 20) were operationally closed in 1997 by filling with grout. In accordance with the FFA, industrial wastewater construction and operating permits were obtained from South Carolina Department of Health and Environmental Control (SCDHEC) for the underground waste tanks. The remaining FTF waste tanks will be filled with grout as part of the tank farm closure plan. A performance assessment is being conducted to support closure decisions (SRS 2008).

3.4.1.1 Role of Cementitious Barriers and Processes Considered

In the SRS PAs cementitious materials are assumed to serve roles as physical and chemical barriers. The materials are used in waste forms, infiltration barrier, barriers to releases and as a barrier to steel corrosion. Rather than being a well mixed waste form, the cementitious material, used in the tank closures was poured on top of the thin, residual waste layer. It provides a barrier to infiltration and provides a reducing environment which limits transport. In addition, a layer of “strong” grout was poured on top of the reducing grout and serves as protection against human intrusion by way of drilling. The concrete surrounding the steel tank is the barrier delaying de-passivation of the steel tank. Figure 12 shows the steel tank and Figure 13 shows the completed tank with surrounding concrete before burial. The cementitious materials are an important feature in the PA for the F-Tank Farm.

Figure 12. Type IIIA Primary and Secondary Carbon Steel Liners - Late Tank Construction

Figure 13. Concrete Vault Around Steel Tank - Final Construction of Type III/IIIA Tank

3.4.1.2 Parameter Assumptions and Conceptual Models

A relatively detailed modeling approach was applied for the cementitious materials considered in the SRS F-Tank Farm models. Figure 14 is a list of the phenomena considered in the PA which affect the durability of the cementitious barriers. As noted above, changes in the cementitious materials were not only assumed to impact migration of water and radionuclides, they also delayed the onset of corrosion of the steel tank. Figure 15 is an illustration of the different features considered in the models for one type of tank.

The results from the SRS F-Tank Farm PA indicate that the tank fill grout can begin degrading hydraulically as early as year 800 (Type IV tanks) with full degradation being reached as early as year 13,000 (Type I tanks). The waste tank concrete can begin degrading as early as year 400 after closure (Type IV tank) with full degradation occurring at year 800 (Type IV tank). The grout was assumed to chemically degrade based on the number of pore water flushes, going from reducing to oxidizing. Figure 16 shows the “history” of the degradation model for the various cementitious materials evaluated in the FTF PA.

The concrete, which contains the steel tank, is considered only for its ability to prevent the steel tank from oxidizing. No hydraulic credit is taken, but credit is taken for sorption in this material. A model (Subramanian 2007) was run to determine the penetration rate from the environment to the steel tank for those chemicals which affect the steel passivation.

3.4.1.3 Relative Importance in Context of Assessment

Two assumptions about the behavior of the cementitious barriers were important to this PA. The more important related to the penetration of the environmental chemicals through the concrete and their affect on the steel tank. No waste was released as long as the steel tank was considered intact. The chemical behavior of the grout in limiting radionuclide migration became more important once the steel tank failed since the reducing properties of the grout helped hold the radionuclides in place.

3.4.2 E-Area Low-level Waste Facility

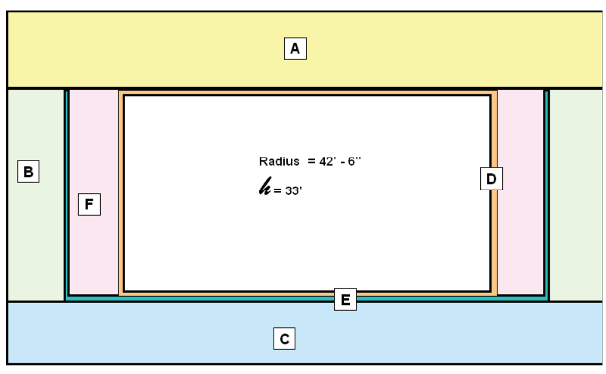
The SRS is located in a temperate climate with the water table relatively close to the ground surface, such that the groundwater pathway is a significant contributor to potential doses. A revised performance assessment was recently completed for the E-Area (SRS 2007).

The E-area Low-level Waste Facility (ELLWF) is located in the central region of the SRS known as the General Separations Area (GSA). It is an elbow-shaped, cleared area, which curves to the northwest, situated immediately north of the Mixed Waste Management Facility (MWMF), a former radioactive waste "burial ground" that received mixed waste and was closed under RCRA.

The ELLWF is composed of 200 acres for waste disposal and a surrounding buffer zone that extends out to the 100-m point of compliance. Radiological waste disposal operation at the ELLWF began in 1994. Disposal units within the footprint of the ELLWF include: Slit Trenches, Engineered Trenches, Components in Grout (CIG) Trenches, the Low-

PHYSICAL FACTORS	CHEMICAL FACTORS
Loss of Mass • Erosion - Water - Wind	**Loss of Mass** • Desiccation (Early water loss) – Cracking • Dissolution/Leaching – Increased Porosity - Water - Acids - Microbial degradation
Mechanical Cracking • Overload • Bio-intrusion • Freeze Thaw • Thermal Stress • Geological Stress - Earthquakes - Subsidence	**Addition of Mass (Expansion) – Cracking** • Sulfate (Ettringite) • Alkali (ASR hygroscopic gel) • Fe (rebar) + Oxygen, Carbonate, Chloride
	Addition of Mass – Fill/Seal Cracks and Pores • Carbonate (Calcium Carbonate Precipitation)

Figure 14. Physical and Chemical Factors Related to SRS FTF Cementitious Barriers Stability (SRS 2008)



LABEL	THICKNESS	MATERIAL
A Concrete Roof	48"	Class C Concrete
B Concrete Wall	30"	Class C Concrete
C Concrete Basemat	41"	Class C Concrete
D Primary Liner	0.5"	Carbon Steel (Tanks 25 - 28: ASTM A-516) (Tanks 44 - 47: ASTM A-537)
E Secondary Liner	3/8"	Carbon Steel (Tanks 25 - 28: ASTM A-516) (Tanks 44 - 47: ASTM A-537)
F Grouted Annulus	30"	Tank Fill Grout

Figure 15. SRS Type IIIA Tank Conceptual Model (SRS 2008)

Cementitious Material Lifetimes	Type I Tank (Years)	Type III Tank (Years)	Type IIIA Tank (Years)	Type IV Tank (Years)
FTF Spec Fill Grout Lifetime (Initial Properties)	0 - 2,600	0 – 5,000	0 – 4,800	0 - 800
Degrading FTF Spec Fill Grout Lifetime	2,600 – 13,000	5,000 – 18,900	4,800 – 18,700	800 – 63,800
Fully Degraded FTF Spec Fill Grout Lifetime	After 13,000	After 18,900	After 18,700	After 63,800
FTF Aged Concrete Lifetime (Initial Properties)	0 – 1,300	0 – 2,500	0 – 2,400	0 - 400
Degrading FTF Aged Concrete Lifetime	1,300 – 2,600	2,500 – 5,000	2,400 – 4,800	400 - 800
Fully Degraded FTF Aged Concrete Lifetime	After 2,600	After 5,000	After 4,800	After 800

Figure 16. SRS FTF Cementitious Barriers Hydraulic Degradation Sequence (SRS 2008)

activity Waste Vault (LAWV), the Intermediate Level Vault (ILV), and the Naval Reactor Component Disposal Area (NRCDA). The Slit and Engineered trenches and the NRCDA do not contain cementitious barriers, all the others do.

3.4.2.1 Role of Cementitious Barriers and Processes Considered

The cementitious barriers serve multiple purposes based on the disposal unit. For CIG trenches, the cementitious material serves as a barrier to advective and diffusive transport. A CIG trench, shown in Figure 17, is an unlined trench in which a layer of grout is poured, waste is emplaced, and then the waste is covered by another grout layer. The components are surround by at least 15 cm of grout.

The ILV and LAWV concrete vaults and cementitious fill materials are used primarily as infiltration barriers. Failure of these barriers is determined by post closure structural anayses. Only advective transport is assumed for these vaults.

3.4.2.2 Parameter Assumptions and Conceptual Models

A variety of parameter assumptions and conceptual models were used in the E-Area PA. The one assumption used in the conceptual models for the various cementitious barriers is that the K_ds were a function of the number of pore flushes through the cementitious material. The grout in the CIG trenches was assumed to remain intact for 40 years, after which its hydraulic properties become those of the closure cap.

Figure 17. Typical SRS CIG Trench

The hydraulic properties of intact vaults remained constant for both the ILV and LAWV. Structural analyses were performed which determined a cracking and collapse history for those vaults. Cracks were assumed to direct infiltration into the waste zones.

3.4.2.3 Relative Importance in Context of Assessment

The relative importance of the cementitious materials varied depending on the disposal unit. The grout in the CIG trenches is critically important as it is assumed to sufficiently contain tritium in the trench until it decays so that regulatory limits are maintained. The vaults of the ILV and LAWV exclude infiltration during the compliance period of DOE Order 435.1, but the structural analyses show that the vaults fail between 1,000 and 10,000 years.

3.4.3 Commercial Nuclear Facilities

Cementitious materials are used in a variety of applications for which the Nuclear Regulatory Commission (NRC) has regulatory oversight or independent technical review responsibilities. The types of applications include but are not limited to: cementitious barriers for near surface engineered waste disposal systems (e.g., waste forms and barriers, entombments, environmental restoration) and structural concrete components of nuclear facilities (spent fuel pools, dry spent fuel storage units, and recycling facilities e.g., fuel fabrication, separations processes). The approach to assessing the performance of cementitious materials in these applications varies from application to application. The following sections provide a summary of NRC experience in assessing the performance of cementitious materials for waste management applications (other applications are addressed in Section 2.3).

3.4.3.1 Overview

In the 1980's, the NRC anticipated that cementitious materials may be used in waste disposal applications, primarily for the commercial disposal of low-level waste (LLW). The NRC sponsored research to estimate the degradation, modeling approaches, and hydrologic performance of concrete barriers (Walton et al. 1990, Walton & Seitz 1991, Walton 1992, Snyder and Clifton 1995). In addition, the NRC developed a waste form technical position, to convey technical guidance to prospective licensees on evaluating the performance of waste forms used for LLW disposal, some of which were expected to be cement stabilized wastes. More recently, the NRC is sponsoring research on the performance of cementitious materials. A summary of the research and the waste form technical position is provided below.

The waste form technical position provides guidance on waste form test methods and results acceptable to the NRC staff for implementing the 10 CFR Part 61 waste form requirements for disposal of low-level waste. The waste form was recognized to potentially serve a number of different functions, including but not limited to providing structural stability, reducing leachability of radionuclides, and reducing the dispersion of waste from inadvertent intrusion. The waste form technical position was initially issued in 1983, and later revised in 1991. The revision provided an appendix on cement stabilized waste forms, in part because portland and pozzolonic cements were observed to exhibit unique chemical and physical interactive behavior when used with certain chemicals and materials encountered in low-level waste streams (Picuilo et al. 1985, Soo & Milian 1989). Main processes considered were: compression (structural strength), thermal cycling, irradiation, biodegradation, and leaching. Guidance was provided for product qualification testing, sample preparation, full-scale testing, and surveillance specimens.

In addition to the waste form technical position, the NRC sponsored research to estimate the performance of concrete engineered barriers (see Table 2). The emphasis of the research was to develop techniques to quantify the expected long-term performance of cementitious engineered barriers. In some of the earlier work, Walton et al examined empirical relationships to estimate degradation of concrete, with emphasis on sulfate and magnesium attack, carbonation, freeze-thaw, and alkali-aggregate interaction (Walton et al. 1990). The latter two degradation mechanisms were expected to be managed through design processes and were not assessed quantitatively. In addition, fracturing of concrete structures via a variety of mechanisms or processes was anticipated. Additional research was completed to evaluate the performance of cracked and partially degraded concrete structures with respect to hydrology and mass transport (Walton 1992, Seitz & Walton 1993).

More recently, the Center for Nuclear Waste Regulatory Analyses (CNWRA) completed a literature review and assessment of factors relevant to the performance of grouted systems for radioactive waste disposal (Pabalan et al. 2009). The report includes: reviews of portland cement-based materials and their properties, discussions of degradation mechanisms, assessments of modeling approaches for predicting chemical degradation, and evaluations of conceptual and mathematical models that can be used to assess the effects of fast pathways and bypassing pathways on radionuclide release. Radionuclide release mechanisms were evaluated, along with data on solubility limits and data on permeability and diffusion properties of cement-based materials.

In 2002 and 2005, the NRC performed independent technical reviews of Department of Energy (DOE) non-HLW determinations for tank closure at the Idaho National Laboratory (INL) and for disposal of salt waste resulting from tank waste retrieval at the Savannah River Site (SRS) (NRC 2006b, NRC 2005). Under Section 3116 of the Ronald W. Reagan National Defense Authorization Act for Fiscal Year 2005 (NDAA), the Secretary of Energy, in consultation with the U.S. Nuclear Regulatory Commission, may determine that certain radioactive waste resulting from reprocessing of spent nuclear fuel is not high-level waste. To facilitate risk-informing the independent technical reviews, the USNRC staff performed independent performance assessment modeling. A key aspect of that modeling has been the assessment of the performance of cementitious materials to stabilize radioactive waste. A brief summary of the modeling approaches is provided below.

3.4.3.2 INL Tank Closure Review

For tank closure at the Idaho National Laboratory, the NRC developed an abstracted source term model capable of representing a variety of system states for the cementitious material and residual stabilized waste (Esh et al. 2002). The residual radioactive waste was in solid and liquid forms in a thin layer on the tank walls and bottom. The stabilization strategy was to fill the tank with a cement formulation, comprised of cement, fly ash, and blast furnace slag, to limit water contact with the waste, provide structural stability, and modify the chemical environment to reduce the release rate of radionuclides. Because the site was semi-arid and water flow was expected to be limited by the natural environment, the USNRC analysis focused on the chemical aspects of the cementitious materials.

Radionuclides released from the waste were simulated to partition between the grout solid phase and cement-modified liquid phase. A key uncertainty was the long-term chemical conditions of the system. The model was developed to evaluate reducing or oxidizing conditions for sorption and application of solubility limits, as well as aging of the grout. Different states of the system could be evaluated, but kinetics for the rate of change from one state to another was not incorporated in the model. The level of refinement in the model was sufficient to develop risk insights for the INL site because of the limited waste inventory in the tanks after cleaning and the semi-arid site conditions.

3.4.3.3 Savannah River Site Saltstone Review

Salt waste is disposed of at the saltstone disposal facility at the Savannah River Site by blending the waste with a mixture of cement, fly ash, and blast furnace slag and pumping the grout mixture as a slurry to large above-ground vaults (USNRC 2005). The resultant waste forms are large cementitious monoliths. The waste form contains blast furnace slag to create reducing conditions in the waste form. Reducing conditions are beneficial primarily because reduced forms of technetium typically are much less mobile than oxidized forms of technetium. USNRC performed independent analysis of the saltstone disposal facility in order to develop risk insights to inform their technical review (Esh et al. 2006).

Preliminary analyses, similar to what was completed in 2002 for the INL tank farm, indicated that the assumption regarding whether the waste would maintain a reducing environment or become oxidizing would have a significant effect on the predicted dose to a member of the public. Because this assumption had a significant effect on dose, and because the assumption that the waste is either entirely reducing or entirely oxidizing is unrealistic, the model was refined to reflect the oxidation of waste as a function of time. In addition, the model was refined with a submodel that predicts physical degradation of the waste as a function of time.

Waste oxidation and waste form degradation were modeled as proceeding from waste surfaces, including the surfaces of cracks, inward in a shrinking core type of representation (Esh et al 2006). Waste form cracking may occur during curing, as a result of settlement, or as a result of other processes. The model did not attempt to predict the amount of cracking that will occur in the waste form. Instead, the potentially complex pattern of cracking in the waste form is represented in the model as a series of planar cracks through the waste, with the crack spacing being an important uncertain variable. At each fracture or exposed surface, an oxidation front and a degradation front were estimated to penetrate into the material. The oxidation and degradation fronts may propagate at different rates, resulting in different thicknesses of material that are oxidized and/or degraded. The degraded thickness as a function of time was estimated from an empirical model for sulfate and magnesium attack for lack of better information (Walton et al. 1990). The empirical models for waste form degradation and oxidation that were implemented in the performance assessment model did not necessarily represent the dominant mechanisms of degradation and oxidation of saltstone waste. Rather, the models served as a tool to evaluate time-dependent degradation or oxidation of the waste.

In the conceptual model, there were three regions in the waste form: intact, oxidized, and degraded. The predicted release of radionuclides from each region of waste was affected by the modeled physical and chemical properties of the waste in each region. The actual degraded waste form may have an extremely complicated collection of units of intact material with variable volumes and shapes. Consistent with the use of the PA model as a review tool, the potentially complicated geometry was simplified into three connected cells in the length dimension of the facility, one for each of the intact, oxidized, and degraded regions. The waste form was assumed to be broken into a series of blocks by fractures extending through the waste form. Therefore the results from the three cells were scaled up to represent the total number of blocks in the system based on the total length of the facility and the assigned fracture spacing. Infiltrating water was assumed to flow through the fractures, thereby resulting in a zero concentration boundary condition at the exposed side of the waste form. Diffusive transport between the three regions of the waste form and from the waste form to the surrounding soil was represented in the model.

3.4.3.4 United States Nuclear Regulatory Commission Summary

NRC has sponsored research over the past several decades on the properties and performance of cementitious materials for a variety of different applications. In addition, independent performance assessment modeling has been performed to develop risk insights in order to risk inform the review of USDOE waste determinations. Irrespective of the research and analysis, key uncertainties remain, particularly with regards to long-term properties and performance.

The key uncertainties for waste management applications include: 1) the initial physical and chemical characteristics of the system, 2) the extent of fractures and their influence on performance, 3) the importance of interactions between processes that may accelerate or limit impacts, and 4) lack of long-term monitoring data and characterization of in-situ, large-scale systems.

4.0 OTHER TYPES OF RISK ASSESSMENTS

In the previous section, examples of PAs for engineered systems were described for various USDOE facilities that incorporate cementitious barriers. In this section, the summary is extended to examples of other types of risk assessments for USDOE facilities including the Idaho, Hanford, and Savannah River Sites. These examples will demonstrate the similarities and differences between PAs and other types of risk assessments performed to support other regulatory processes (e.g., CERCLA and RCRA).

4.1 Idaho Sites

4.1.1 Non-Time Critical Removal Action for the Engineering Test Reactor under CERCLA

The Engineering Test Reactor (ETR) located at the Idaho National Laboratory (INL) is in the process of being decommissioned (i.e., decontaminated and dismantled) by CH2M-WG Idaho, LLC (CWI) (USDOE-ID 2007). The decommissioning strategy includes removal of the pressure vessel, grouting and disposal of the vessel and internals at the INEEL CERCLA Disposal Facility (ICDF)[30], and demolishing the reactor building to ground level (USDOE-ID 2007). This action is 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 the constituents that were not eliminated in the screening process (McCarthy 2006; Staley 2006)[31]. In September 2007, the pressure vessel was removed from the ETR, grouted, and finally disposed of at the ICDF[32].

4.1.1.1 Role of Cementitious Barriers and Processes Considered

After transport to the ICDF, the remaining voids in the ETR pressure vessel were filled with a

[30] The ETR vessel meets ICDF waste acceptance criteria for disposal as low-level radioactive waste (USDOE-ID 2007b).
[31] The action met the remedial action objectives "regarding long-term risk, minimizes short-term worker risk and radiation exposure, reduces the footprint of waste sites at the INL, is cost effective, and provides a safe and stable configuration that is environmentally sound" (USDOE-ID 2007b).
[32] A video of the relocation of the ETR is provided at Engineering Test Reactor (ETR) Vessel Relocated available at http://www.id.doe.gov/NEWS/PressReleases/PR071002.htm (accessed March 1, 2009).

cementitious grout as part of the disposal process. The performance assessment for the ICDF, where the ETR pressure vessel was grouted and disposed, uses the assumption that land-use controls prohibit future residential use, so there was no evaluation of risks to a future residential receptor (USDOE-ID 2007). Furthermore, by removing the pressure vessel and grouting the void volume in the vessel, the risks to a future resident were determined to be acceptable.

In the risk assessments for the ETR pressure vessel disposal, no hydraulic credit was taken for the cementitious materials employed. The ETR contaminant inventory was assumed to remain in place and the area was stabilized using native soil (McCarthy 2006; Staley 2006). The only credit taken for the cementitious materials was as a means to limit the potential for subsidence and resulting impacts on water flux by filling the voids in the pressure vessel.

Despite these assumptions, the risks were found to be acceptable if the ETR pressure vessel was removed. For groundwater impacts, the predicted groundwater concentrations satisfy the performance criteria for the site (McCarthy 2006)[33]. The maximum predicted cumulative risk to groundwater is 2×10^{-6}, which is dominated by C-14. The maximum cumulative fraction of nonradionuclide concentration to the relevant maximum contaminant level (MCL), which is dominated by chromium, is a factor of six times less than the performance criteria of 1 (McCarthy 2006). For all remaining pathways, the cumulative risk associated with removing the pressure vessel translates to a cancer risk 3×10^{-7}, which satisfies the NCP criterion of 10^{-4} (Staley 2006). These acceptable predicted risks were based on the use of native soil to stabilize the area after removal of the ETR pressure vessel. The use of grout would likely produce lower risks than those predicted.

4.1.1.2 Important Assumptions and Conceptual Models

The important assumptions made to predict groundwater risks for the ETR pressure vessel disposal include (McCarthy 2006):

- Native soils (and not grout) are used to fill both the reactor and ETR basement.
- Any hydraulic effects of grouting the ETR vessel are ignored.
- Contaminants are loose in the soil and immediately available for leaching to the subsurface—no consideration is taken for waste forms.
- Contaminants are assumed to move down through the vadose zone sediments without retardation or horizontal spreading resulting in a shorter travel time from the ETR to the aquifer than would be expected.
- The receptor was assumed to be at the edge of the ETR facility.

For the screening assessment for the other pathways (i.e., non-groundwater), the important assumptions include (Staley 2006):

- Contaminants down to 3 m (10 feet) below-grade remaining after decontaminating and dismantling are mixed uniformly in the top 3 m of soil and will be available to an intruder in the year 2095.
- A receptor will build a house at the site of the removal action, 3 m of contaminated material will be excavated, and the excavated material will be spread across the surface of the housing site.

[33] These criteria are designed to prevent contamination of the underlying sole-source aquifer to exceed a cumulative carcinogenic risk level of 10^4 or applicable State of Idaho groundwater quality standards (McCarthy 2006).

The receptor will live at the site for 30 years, including 6 years of childhood, and will be exposed to external radiation and to contamination through soil ingestion, fugitive dust inhalation, and ingestion of contaminated fruits and vegetables.

The primary conceptual model used in the more detailed phase of the groundwater risk screening for the ETR facility disposition is illustrated in Figure 18[34]. This detailed screening used an implementation of the groundwater-screening model GWSCREEN and was intended to be conservative (Rood 1994). The assumptions made with intent to maximize groundwater risks in the screening included (McCarthy 2006):

- All radionuclides present in the ETR are mixed homogeneously with soil and placed in a volume represented by the volume of the ETR below-ground structure, 35 m × 35 m.
- There was no containment, engineered barriers, waste form impacts, or solubility-limited releases.
- Transport is one-dimensional in an 18-m (60-ft) thick unsaturated zone composed of sedimentary interbeds because flow through the fracture basalt is much faster.
- The receptor well is placed on the downgradient edge of the facility.
- The infiltration rate is 10 cm/yr (3.9 in/yr).
- There is no dispersion in the unsaturated zone, which may or may not be "conservative."
- The aquifer is a homogeneous isotropic media of infinite lateral extent and finite thickness.
- Contaminants entering the aquifer mix with water in the aquifer over a depth defined by a typical well screen of 15 m (49.2 ft).

Therefore, no assumptions were made specifically pertaining to cementitious materials other than to assume that voids are sufficiently filled to preclude the potential for substantial amounts of subsidence that would significantly increase infiltration rates through the cover.

4.1.1.3 Relative Importance in Assessment

No credit was taken for grouting either the ETR pressure vessel (disposed of in the ICDF) or the area after its removal in estimating risks to a future residential receptor for all pathways (McCarthy 2006; Staley 2006). Despite these assumptions, the risks were found to be acceptable under CERCLA and other pertinent regulations for this disposal path. Therefore, the performance and properties of the grout actually used in disposal were inconsequential in the risk assessments and modeling performed to support disposal of the ETR pressure vessel and facility.

4.1.2 Radioactive Waste Management Complex under CERCLA

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 waste is the primary source of contamination (USDOE-ID 2008).

A Record of Decision (ROD) has been completed for the final closure of the RWMC (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). Whereas the previous example for disposition of the ETR pressure vessel took no credit for cementitious materials, contaminants in the RWMC were originally buried

[34] The initial contaminant screening phase relied on the screening techniques developed by the National Council on Radiation Protection and Measurements (NCRP) (NCRP 1996a; NCRP 1996b).

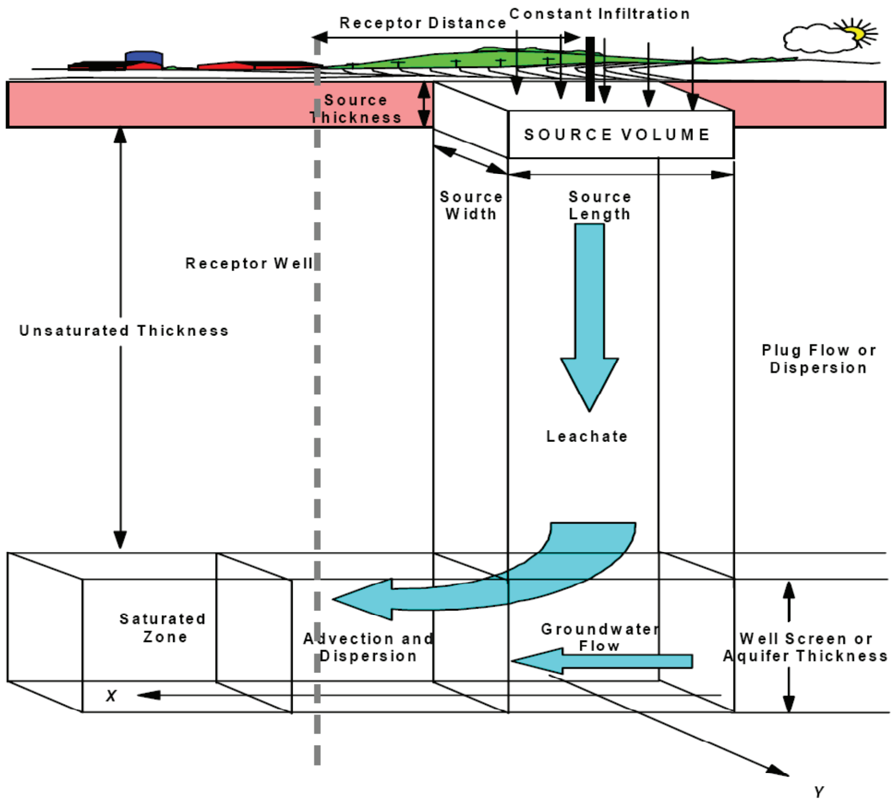


Figure 18. Conceptual Model for the GWSCREEN Groundwater Model (McCarthy 2006)

in containers, vaults, and cementitious waste forms, which have been accounted for in the risk assessments. Furthermore, beryllium reflector blocks from research reactors buried at the SDA were grouted using a paraffin-based grout under a CERCLA non-time-critical removal action[35].

4.1.2.1 Role of Cementitious Barriers and Processes Considered

Earlier baseline risk assessments for the RWMC included taking some credit for diffusional releases of radionuclides from specific concrete waste forms (Becker et al. 1998; Holdren et al. 2002). In the final baseline risk assessment developed to support the RWMC remedial investigation and ROD, no credit was taken for the diffusional release from concrete waste forms or the effect of containment of contaminants in concrete casks (Anderson and Becker 2006; Holdren et al. 2006)[36]. Therefore, no credit was taken for cementitious materials in the context of the final baseline risk assessment performed to support the CERCLA remedial investigation process and the final ROD for the RWMC.

[35] The Acid Pit (Operable Unit 7-02) was also partially grouted in 1997 as part of a treatability study (Loomis et al. 1998a; Loomis et al. 1998b), which also limited potential impacts from mercury.

[36] However, for the risk assessment performed to support the RWMC feasibility study, thick-walled concrete containers were assumed to not fail during the assessment period and any contaminants in such containers were isolated from contact with infiltrating water and thus release and transport (Anderson and Becker 2006). However, no credit was taken for cementitious waste forms in the assessments to support either the remedial investigation or feasibility study under CERCLA.

4.1.2.2 Important Assumptions and Conceptual Models

Numerous assumptions for source-term and flow and transport modeling were made to predict potential impacts of contaminants in the RWMC during the baseline risk assessments. However, because cementitious materials are only involved in containing or immobilizing buried contaminants, only those assumptions pertinent to source-term conceptual model are provided. The pertinent source-term-related assumptions for the baseline risk assessment performed for the CERCLA remedial investigation include (Anderson and Becker 2006):

- Wastes are either buried without containers or in containers and contaminants not in containers are available for immediate release.
- Once a container fails, the remaining contaminants are available for release.
- Wastes in wooden or cardboard boxes are assumed loose due to the relatively short life span of such containers and all other waste containers are assumed to be steel drums. There are no concrete containers.
- Once the mass is released from the waste form, it is available for transport.

Cementitious materials were treated like soil for the purposes of source-term release using a surface-wash-type model for materials with surface contamination readily leached by infiltrating water where release is controlled by partitioning between the waste form and water. Because specific waste-to-water distribution coefficients (or in this case, concrete-to-water coefficients) were not known, soil-to-water distribution coefficients were used to simulate the releases (Anderson and Becker 2006).

4.1.2.3 Relative Importance in Context of Assessment

In the baseline risk assessment (BRA) for the remedial investigation (RI), any contaminants associated with concrete waste forms were treated as if they were in soil. Furthermore, any wastes that were originally buried in concrete containers were assumed to be either loose or in drums in the RI BRA although some credit was taken in the BRA performed for the RWMC feasibility study. Thus the performance and properties of cementitious materials were inconsequential in the risk assessment modeling for the RWMC.

4.1.3 Waste Calcining Facility Landfill Closure under RCRA and NEPA Environmental Assessment

The Waste Calcining Facility (WCF) is located at the Idaho Nuclear Technology and Engineering Center (INTEC) on the USDOE Idaho Site (USDOE-ID 2006)[37]. The WCF was used from 1963 to 1981 to calcine and evaporate aqueous wastes generated from reprocessing spent nuclear fuel. In 1998, the WCF 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[38]. In 2003 the Idaho Department of Environmental Quality issued a final HWMA/RCRA post-closure permit.

This method of closing a RCRA facility (as a landfill) with mixed waste liabilities was innovative and well suited to closing highly radioactive process facilities which involve great expense and removal/remediation

[37]The INTEC Waste Calcining Facility (WCF) on the Idaho Site is often referred to as the “Old Waste Calcining Facility” in deference to a newer calcining facility located on the USDOE Idaho Site.
[38]Grouting was not required for structural support of the cap.

of large volumes of waste (Demmer et al. 1999). Regulations for the WCF waste piles require preparation of closure and post-closure plans, and the State of Idaho wanted the risk of release to be consistent with the Federal Facilities Agreement/Consent Order (FFA/CO) remedial goals for INTEC. The USDOE assessed the radionuclide risks in parallel with the RCRA closure for hazardous constituents (Demmer et al. 1999). The USDOE also assessed the WCF landfill closure using an Environmental Assessment (EA) to evaluate potential risks associated with hazardous and radioactive constituents[39].

4.1.3.1 Role of Cementitious Barriers and Processes Considered

The risk assessment developed to support the WCF closure was also innovative. Characterization efforts were often hampered because some WCF areas were highly radioactive and process equipment was located in areas with severely limited access (Demmer et al. 1999). Thus a model was developed based on conservative assumptions to represent process conditions and residual contaminants at closure[40]. The conservative model predictions were then used for source-term characterization.

The primary impact of cementitious materials on the WCF risk assessment was expressed in the modeling performed to estimate risks from ingestion of contaminated groundwater. Transport modeling for the groundwater pathway was performed in two phases:

- A screening phase which was based on conservative assumptions (i.e., no concrete cap or grouting) using the GWSCREEN model (Rood 1994).
- A detailed phase which took credit for both grouting within the WCF and the concrete cap using the PORFLOW transport model (ACRi 2002).

In the detailed modeling phase, the concrete is assumed to crack and water will enter the waste leaching contaminants, which are then transported into the surrounding soil[41].

4.1.3.2 Important Assumptions and Conceptual Models

The model used to predict residual levels of both radioactive and hazardous contaminants at the time of closure was based on conservative assumptions (Demmer et al. 1999). Major assumptions included:

- There were no organic constituents (or corresponding risks to human health or the environment) in the process residues found in vessels and piping or on the cell floors because of dry conditions and high process temperatures.
- The majority of the residual material was from the final “zirconium” campaign.
- All radionuclides were decayed from the time of the last WCF campaign to provide conservative values.
- Conservative estimates for hazardous chemicals that may have been used in the WCF were used for the source term.

In the risk assessment, two exposure scenarios were evaluated for human health risk evaluation: (1) current occupational and (2) 30-year future resident. For these receptors, the conceptual site model identified three possible pathways: groundwater ingestion,

[39] The risk assessment used to support the EA for the WCF followed the same methodology used for the CERCLA program under the FFA/CO (Demmer et al. 1999).
[40] The model estimated the quantity of residual material using operating records, interviews, design specifications, drawings, video inspections, and safety analysis reports (Demmer et al. 1999).
[41] This assumption was considered to be conservative because the RCRA closure requirement states that the integrity of the cap will be maintained (Demmer et al. 1999).

dermal contact with contaminated groundwater, and external exposure[42]. For the current occupational scenario, the external exposure pathway was evaluated. For the future resident, both the external exposure and groundwater ingestion pathways were evaluated. The potential impact of cementitious materials was considered in the groundwater transport modeling to estimate health risks to the future resident.

4.1.3.3 Relative Importance in Context of Assessment

As described above, the modeling performed for groundwater transport was performed in two phases: screening and a more detailed analysis (Demmer et al. 1999)[43]. Based on screening model results (assuming no grouting or cap), the maximum concentrations for all hazardous constituents and all but four radionuclides had risks below the NCP target risk of 10^{-6}. The other four radionuclides (i.e., Np-237, Pu-239, Pu-240, and Tc-99) had maximum concentrations within the NCP target risk range of 10^{-6} to 10^{-4}. These four radionuclides were then evaluated using the more refined transport model that took credit for the concrete cap and WCF grout including cracking.

The maximum predicted groundwater concentrations from the more detailed model translated into ingestion risks below the NCP 10^{-6} risk limit for Np-237, Pu-239, and Pu-240. The corresponding ingestion risk for Tc-99 was slightly above the 10^{-6} limit and approximately two orders of magnitude lower than the upper NCP risk limit of 10^{-4}. Furthermore, the maximum predicted groundwater concentration for Tc-99 was significantly less than the proposed maximum concentration for drinking water. Thus, the cementitious materials had a significant impact on the predicted groundwater concentrations and corresponding risks for the WCF landfill closure. However, in the typical CERCLA sense, the predicted risks when no cementitious materials are employed satisfy the upper bound risk level of 10^{-4} and the impact of cementitious materials can be considered to provide additional assurance that the WCF landfill closure would be protective of human health.

4.2 Savannah River Site

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

Since the 1950s, the 51 tanks in the F- and H-Area Tank Farms on the Savannah River Site (SRS) have received high-level radioactive waste generated by various SRS production, processing, and laboratory facilities. These tanks are permitted under a waste water operating permit and will be closed under this permit (Picha et al. 1999)[44]. In 1995 the USDOE began to prepare for closure of the high-level waste (HLW) tanks by preparing both a closure plan (SRS 1996) and an Environmental Assessment (under NEPA) to evaluate alternatives for the closure of SRS HLW tanks (USDOE-SR 1996a). The result of the NEPA EA evaluation process was a Finding of No Significant Impact (FONSI) (signed in 1996) in which it was concluded that closure of the HLW tanks in accordance with the closure plan would not result in significant environmental impacts (USDOE-SR 1996b).

[42] No toxicities were available for dermal contact with contaminated groundwater so this pathway was not evaluated in the assessment (Demmer et al. 1999).
[43] The same modeling results were used to support analysis of remedial alternatives in the environmental assessment for the WCF (USDOE-ID 1996).
[44] The primary regulatory driver for the removal of wastes from the HLW tanks at SRS is the FFA/CO (WSRC 1993) between USDOE-SR, the state of South Carolina, and the USEPA (Picha et al. 1999). The regulation governing closure is SC Regulation R.61-82, "Proper Closeout of Wastewater Treatment Facilities," which is intended for typical wastewater facilities and "provides virtually no guidance applicable to HLW tank closure" (USDOE 1999).

SRS HLW 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). These tanks were selected because they were known to have relatively low levels of residual radioactive waste (Picha et al. 1999). Bulk waste was removed to less than 113,550 L (30,000 gal)[45]. After heel removal, approximately 3,785 L (1,000 gal) gallons of residual waste was left in the tank (Elmore and Henderson 2002). Grouting of the tanks for closure was carried out in three stages. A reducing grout was initially added to mix with residual waste to stabilize it as well as possible. A large layer of a controlled low-strength grout material was then added and finally each tank was capped by the addition of a high-strength grout (Picha et al. 1999).

4.2.1.1 Role of Cementitious Barriers and Processes Considered

The risk evaluations that demonstrated that tank closures result in configurations that ensure overall protection of human health and the environment are provided in the closure modules (SRS 1997a; SRS 1997b). The risk evaluation for the SRS Tank 17-F closure is provided as an example and is similar to the one for Tank 20-F. The fate and transport modeling and corresponding risk analyses provide assurance that this tank closure will be protective of human and ecological receptors under reasonable land use controls (SRS 1997a).

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. Transport modeling for the groundwater pathway was performed using the Multimedia Environmental Pollutant Assessment System (MEPAS) computer code (Droppo et al. 1989). Credit was taken for the cementitious material (i.e., grout) after tank closure for two distinct periods (see Table 4): an initial 1,000-year period when the basemat, grout, and tank top were assumed undegraded and then a “failed” period after 1,000 years when each layer was assumed to have instantaneously and completely failed resulting in corresponding increases in hydraulic conductivities (SRS 1997a).

Table 4. Properties Impacted by Failure (at 1,000 years) for the Tank 17-F Model (SRS 1997a)

Simulation time (yrs)	Basemat Hydraulic conductivity (cm/yr)	Infiltration rate (cm/yr)
0 – 1,000	9.6E-09	4
1,000 – 10,000	6.63E-03	40

4.2.1.2 Important Assumptions and Conceptual Models

The fate and transport modeling performed to support tank closure modeling was based on the following assumptions (SRS 1997a):

- An institutional control for 100 years and then industrial land use.
- The assessment area where receptors may be exposed remains in commercial/industrial use for the entire 10,000-year assessment period.
- There is no credible scenario for the transport of contaminants to the atmosphere and so this transport pathway was not analyzed.
- Ponding above the contaminated waste zone does not occur.

[45] A waste determination indicated that tank closures would satisfy incidental waste criteria (Picha et al. 1999).

- The release of contaminants from the grout in the tank is controlled by a grout-water partition coefficient model. The tank and internal piping are assumed filled with a strongly reducing grout with constant partition coefficients over the entire simulation period.
- Site-specific exposure parameters were used when available although many default parameters in MEPAS were used.

Potential exposure impacts were predicted for an adult worker, a teenage intruder, a nearby adult resident, and a nearby child resident[46]. Receptors may be exposed via various surface pathways.

Upon closure, a tank is filled with grout and no engineered structures will be used to reduce infiltrating water. Example distribution coefficients and material properties assumed for cement-based materials are provide in Tables 5 and 6, respectively.

In the PA modeling, an infiltration rate of 4 cm/yr was used prior to failure, and a rate of 40 cm/yr used after the grout and basemat failure (SRS 1997a). Based on the SRS E-Area vaults performance assessment (Cook and Hunt 1994), a conservative assumption was made that the basemat, grout, and tank top fail at 1,000 years with resulting increases in hydraulic conductivities (SRS 1997a). An engineered cover over the tank after closure was not evaluated[47].

4.2.1.3 Relative Importance in Context of Assessment

In the risk analyses to support the Tank 17-F closure fate and transport modeling, cementitious materials (i.e., grout and concrete) are considered in two important areas over differing time periods. In the initial, undegraded period of 1,000 yrs, the concrete basemat has a relatively low hydraulic conductivity ($1x10^{-8}$ cm/s) and the infiltration rate is assumed to be 4 cm/yr[48]. At a simulation time of 1,000 years, the concrete tank top, grout fill, and concrete basemat are assumed to fail instantaneously resulting in changes to both the hydraulic conductivity of the basemat (to 1×10^{-2} cm/s) and infiltration rate through the grout (to 40 cm/s). The tank and internal piping are assumed filled with a strongly reducing grout during the entire simulation (SRS 1997a).

The results of the MEPAS simulation using the aforementioned assumptions indicated that none of the contaminants (i.e., radiological and hazardous) were predicted to violate any performance objectives[49]. For radiological concerns, the Tc-99 was the dominant contributor to the radiation dose to the receptors. The seepline concentrations remained low and the predicted gross alpha concentrations in both groundwater and surface water remained well below any of the performance objectives during the 10,000-yr simulation period (SRS 1997a).

Because no simulations were run without consideration of cementitious materials, it is difficult to characterize the specific impacts of these materials on the modeling results. The simulation results indicate that the maximum predicted lifetime cancer risk for various receptors exceeds the NCP 10^{-6} cancer risk limit when taking credit for cementitious materials. Thus it is likely that not taking credit for these materials will result in predictions that violate performance objectives. In addition, the properties and performance of the cementitious materials are important factors in the risk analysis. However, reducing the uncertainties

[46] An ecological risk assessment was also performed for tank closure.
[47] 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 can be assumed to be the same as for a grout-filled tank with no cover with an appropriate delay.
[48] The excess water produced under these conditions is assumed to run off (e.g., over the side) so that ponding above the contaminated waste zone does not occur (SRS 1997a).
[49] Similar results were found in the Tank 20-F risk analysis supporting closure (SRS 1997b).

Table 5. Selected Radionuclide and Chemical Partition Coefficients (K_d) Used in the Tank 17-F Model (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 and Sarott 1995)
i. All chromium modeled as Cr(VI)
j. Solubility limit used to estimate K_d (WSRC 1994)

Table 6. MEPAS Groundwater Parameters for Vadose and Saturated Zones for the Tank 17-F Model (SRS 1997a)

Parameter	Concrete Basemat		Vadose Zone	Water Table Aquifer	Tan Clay Layer	Barnwell-McBean Aquifer	Green Clay Layer
	Intact	Failed					
Thickness (cm)	1.5	1.5	13.7	101.6	7.6	152.4	12.7
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 (cm)	0.18	0.18	1.6	12.2	0.91	18.3	1.5
Vertical hydraulic conductivity (cm/s)	9.6E-09	6.63E-03	7.1E-03	7.1E-03	1.6E-06	5.6E-04	4.4E-09

and conservatism introduced into the modeling of cementitious materials in the risk analysis would result in more accurate predictions that might again demonstrate compliance with performance objectives for the tank closure. Increased accuracy in modeling cementitious barriers is one goal of the CBP.

4.2.2 P Reactor In-Situ Decommissioning under CERCLA

The P-Reactor facility is being decommissioned under the CERCLA process. A risk assessment was conducted as one input for the selection of the preferred closure option for closure in the feasibility study (Council 2008). The risk assessment included consideration of concrete and grout material properties that were assumed to degrade over time. This section includes a brief summary of the approach adopted to assess performance of the concrete and grout materials for the reactor vessel portion of the facility. Similar assessments were conducted for the other major features in the facility.

4.2.2.1 Role of Cementitious Barriers and Processes Considered

The concrete and grout are assumed to function as physical and chemical barriers. Hydraulic conductivity and distribution coefficients are assumed to change with time, in general degrading the performance of the barries. Cracks are not specifically modeled and are represented as changes in the bulk hydraulic conductivity of the porous media.

4.2.2.2 Important Assumptions and Conceptual Models

The concrete and grout fill are assumed to behave as porous media and are represented in a one dimensional manner as shown in Figure 19. Material properties

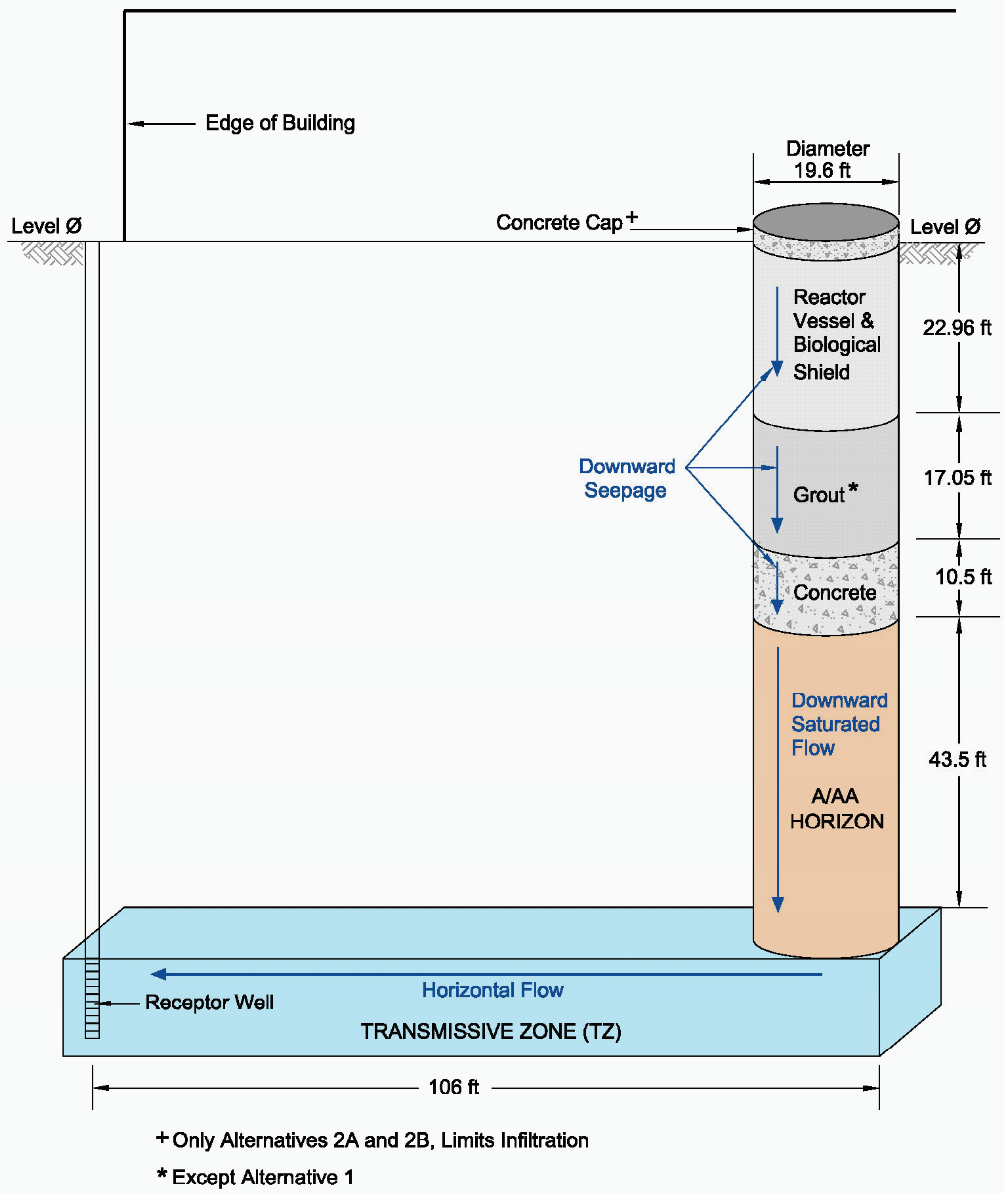


Figure 19. Conceptual Model for P Reactor Vessel (Council 2008)

for the concrete and grout were based on representative values from data used for the F-Tank Farm PA. A summary of the assumed initial values for porosity, particle density, and hydraulic conductivity is provided in Table 7.

The saturated hydraulic conductivity is assumed to change as a function of time according to an exponential function with an assumed half-life of 500 years. The hydraulic conductivity is assumed to increase half way to its maximum value on a log scale for each 500-year time period.

From a chemical barrier perspective, the performance of the concrete and grout fill is represented via distribution coefficients. Distribution coefficients are assumed to change as the concrete or grout ages. The changes in distribution coefficients are assumed to occur based on the number of pore volumes of fluid that pass through a given region of the domain. The number of pore volumes for transitions between the three different stages of degradation are based on assumptions in the F-Tank Farm PA model. For the probabilistic assessment, best-estimate and conservative values are used to define normally distributed inputs. The best-estimate is used as the mean and the standard deviation is calculated from one half of the difference between the mean and conservative value.

4.2.2.3 Relative Importance in Context of Assessment

The corrosion rate assumed for the stainless steel reactor vessel was the most sensitive parameter in the model, but the assumed distribution coefficient for Ni was also shown to be important resulting in increased doses at earlier times for lower values. Both of those parameters are linked to the projected dose from Ni-59, which was the primary dose contributor. Cementitious materials were not important contributors to the results other than serving to limit the potential for subsidence which could result in significant localized increases in infiltration.

Table 7. Example Input Parameter Values for P-Reactor Risk Assessment

	Mean (default)	Distribution	Std Deviation
Concrete			
Porosity	0.168	Normal	0.02
Particle Density	2.51 g/cm^3	Deterministic	
Initial Hydraulic Conductivity	3.5 x 10^{-8} cm/s	Log-normal	10
Grout			
Porosity	0.266	Normal	0.02
Particle Density	2.51 g/cm^3	Deterministic	
Initial Hydraulic Conductivity	3.6 x 10^{-8} cm/s	Log-normal	10

Note: Mean and standard deviation are geometric for the lognormal distribution.

4.3 Hanford Site

4.3.1 221-U Facility Remedial Actions under CERCLA and NEPA

The 221-U Facility is one of three canyon buildings originally constructed in the 200-Area in the mid-1940s to extract plutonium from fuel rods irradiated in Hanford Site reactors (USDOE-RL 2005). However, the facility was never used for plutonium extraction because existing facilities were available to meet production goals. The 221-U Facility was used to train plant operators until 1952 when it was converted to a tributyl phosphate (TBP) process to recover uranium from bismuth phosphate process wastes (USDOE-RL 2001a). The facility was placed in standby in 1958 and subsequently retired; all process hardware remained inside the building.

The selected remedy for the facility includes waste removal from vessels and equipment, removal and treatment of liquids, grouting of internal vessel spaces, demolition of various structures followed by stabilization to support an engineered barrier, construction of the barrier, institutional controls, barrier inspection and maintenance, and barrier performance and groundwater monitoring. These remedial actions will protect human health and the environment based on an industrial use scenario (USDOE-RL 2005).

Because entombment alternatives were considered for the 221-U Facility that would essentially create a low-level waste disposal unit, the requirements of USDOE Order 435.1 (Radioactive Waste Management) apply. These requirements were considered after the CERCLA ROD was issued (Bilson 2001). In the CERCLA process developed for the 221-U Facility, National Environmental Policy Act (NEPA) values were considered to evaluate the potential environmental consequences of the proposed remedial alternatives. These values included potential effects on transportation resources, air quality, cultural and historical resources, noise, visual, and aesthetic impacts, environmental justice, and socioeconomic impacts (USDOE-RL 2001a). Each of these values was evaluated as part of the final feasibility study for the 221-U Facility.

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 five canyon buildings in the 200 Area (USDOE-RL 2005). The Canyon Disposition Initiative (CDI) was designed to help identify end states and evaluate the potential for safe disposal of wastes from other Hanford cleanup actions in the 200 Area canyons (USDOE-RL 2005)[50]. The 221-U Facility will serve as the pilot for the other Hanford canyon buildings, which will be addressed under CERCLA remedial or non-time critical removal actions in accordance with appropriate CERCLA, RCRA, and NEPA review processes (USDOE-RL 2005).

4.3.1.1 Role of Cementitious Barriers and Processes Considered

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 2001a). Instead risk analyses were performed to define baseline and closure conditions. Preliminary remediation goals (PRGs) were also provided in the final feasibility study report for the 221-U Facility (USDOE-RL 2001a). These calculations were performed using the RESidual RADioactivity (RESRAD) code (Yu et al. 2001) for

[50] The CDI resulted from the Tri-Party Agreement among USDOE, USEPA, and the State of Washington Department of Ecology.

radionuclide doses. The non-carcinogenic human health risks from hazardous chemicals were predicted using the equations in the Hanford Site Risk Assessment Methodology (HSRAM) (USDOE 1995). The final feasibility study indicated that the risks due to the contaminants in the 221-U Facility were unacceptable based on CERCLA requirements and ARARs. However, the selected remedial alternative (i.e., Close in Place – Partially Demolished Structure) would be protective of human health and the environment for an industrial land use scenario as long as the surface barrier remains intact (USDOE-RL 2005).

The results of the assessment indicated that the selected remedial alternative for the 221-U Facility was based primarily on the long-term effectiveness of the engineered cap that will be constructed over the facility after the structure is demolished and vessels are grouted in-place (USDOE-RL 2001a). No credit was taken for cementitious materials in the modeling performed to support the ROD for the 221-U Facility.

4.3.1.2 Important Assumptions and Conceptual Models

The maximum baseline risks for the 221-U Facility were predicted for the industrial use scenario. To evaluate human health risks for the industrial use scenario, the following assumptions were made (USDOE-RL 2005):

- Adult workers are exposed.
- The site is under industrial-exclusive use for the first 50 years after closure and industrial use for at least 100 years after that.
- Direct exposure to onsite workers is from residual contamination to a depth of 4.6 m (15 ft).
- The exposure pathways for calculating radionuclide risks include: 1) direct exposure, 2) soil ingestion, and 3) inhalation.
- Standard exposure assumptions from the relevant USEPA guidance are applicable.

Therefore, no assumptions used to predicted the baseline risks pertain to the use of cementitious materials or their properties even though the vessels in the 221-U Facility will be grouted prior to emplacement of an engineered surface barrier.

4.3.1.3 Relative Importance in Context of Assessment

In the fate and transport modeling and risk analysis performed to support the ROD for the Hanford 221-U Facility, no credit was taken for cementitious barriers (i.e., grout) (USDOE-RL 2001a; USDOE-RL 2005). The selected alternative is predicted to be protective of human health and the environment as long as the engineered barrier remains effective. The only credit taken for cementitious materials (grouting process vessels prior to cap placement) is as a “defense-in-depth” measure in case the engineered barrier fails during the 1,000-year simulation period (USDOE-RL 2005). Therefore, the properties and performance of the cementitious materials used in the remedial actions selected for 221-U facility are not relatively important; they only provide defense-in-depth.

4.3.2 Tank Waste Remediation System Final EIS under NEPA

The National Environmental Policy Act (NEPA) requires Federal agencies to evaluate potential environmental impacts of proposed actions to promote public awareness, provide for public involvement, and aid in decision-making. Examples in this section illustrates that NEPA Environmental Assessments have been developed in conjunction with, or to support, CERCLA and State assessments. However, none included an Environmental Impact Statement (EIS). The NEPA EIS assessment process is important enough to be described in this example.

The USDOE is responsible for waste management and environmental restoration at the Hanford Site near Richland, Washington. The proposed action analyzed below is for the management and ultimate disposal of wastes in the Tank Waste Remediation System (TWRS) (USDOE-RL 1996). From 1943 to 1989, the principal mission was the production of weapons-grade plutonium. The associated chemical separations processes resulted in large volumes of radioactive wastes. These wastes are 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.

Hanford is a large USDOE site residing in a semi-arid region near Richland, Washington. Almost half a million people live within an 80-kilometer (50-mile) radius of the site (USDOE-RL 1996). Agricultural land borders the site except to the southeast where the city of Richland is located. The Columbia River, which is used for irrigation and drinking water, flows through the northern area of the Site and forms part of its eastern boundary (USDOE-RL 1996). Groundwater flows beneath the 200 Areas at depths ranging from 70 to 90 + meters (230 to 300 + feet). Past practices have resulted in extensive contamination in the soils beneath the 200 Areas especially near waste management facilities and locations of unplanned releases. These contaminants have migrated to the groundwater and toward the Columbia River (USDOE-RL 1996).

4.3.2.1 Role of Cementitious Barriers and Processes Considered

The National Environmental Policy Act (Pub. L. 91-190) was the first of the new major environmental laws enacted in the U.S. in response to growing concerns about environmental pollution and quality. NEPA provides the foundation for inserting environmental considerations into federal decision-making and established the U.S. national environmental policies (CEQ 2007).

An EIS must be prepared if a proposed major federal action may significantly affect the quality of the human environment. A draft EIS is prepared for public comment. The focal point of the EIS is the alternatives analysis. Substantive comments are addressed in the final EIS.

The NEPA process at Hanford resulted in the development of an EIS to consider safe storage and disposal alternatives for the tank wastes. The focus of the EIS was an alternatives analysis. Alternatives were selected to represent the wide range of possibilities for Hanford tank wastes and can be grouped into the following categories based on the extent of waste retrieval as illustrated in Figure 20 (USDOE-RL 1996):

- **Continued Management**: Two alternatives were analyzed: one without replacing double-shell tanks and the other with replacing these tanks and upgrading tank farm systems to provide long-term management. No retrieval would be performed for these alternatives.
- **Minimal Retrieval**: Only liquid wastes would be retrieved from the double-shell tanks and concentrated with concentrated wastes returned to the tanks. Solid waste would be disposed of in-place. Two alternatives were analyzed: one without treatment and one with in-tank treatment.
- **Partial Retrieval**: The tank wastes with the fewest potential environmental impacts would be disposed of *in situ* and the liquid and solid wastes with the greatest potential long-term groundwater impacts would be retrieved from the tanks for immobilization and disposal. Retrieved wastes would be separated into low-activity and high-level wastes. Low-activity wastes would be immobilized and disposed of onsite in near-surface concrete vaults covered with a cap. High-level wastes would also be immobilized and stored onsite for eventual disposal in a geologic repository. Two partial retrieval alternatives were analyzed: one reducing long-term human health risk by approximately 90 percent and the other by 85 percent.
- **Extensive Retrieval**: Practically all solid and liquid wastes would be retrieved and separated into low- and high-level fractions. The waste treatment and disposal methods are the same as the partial

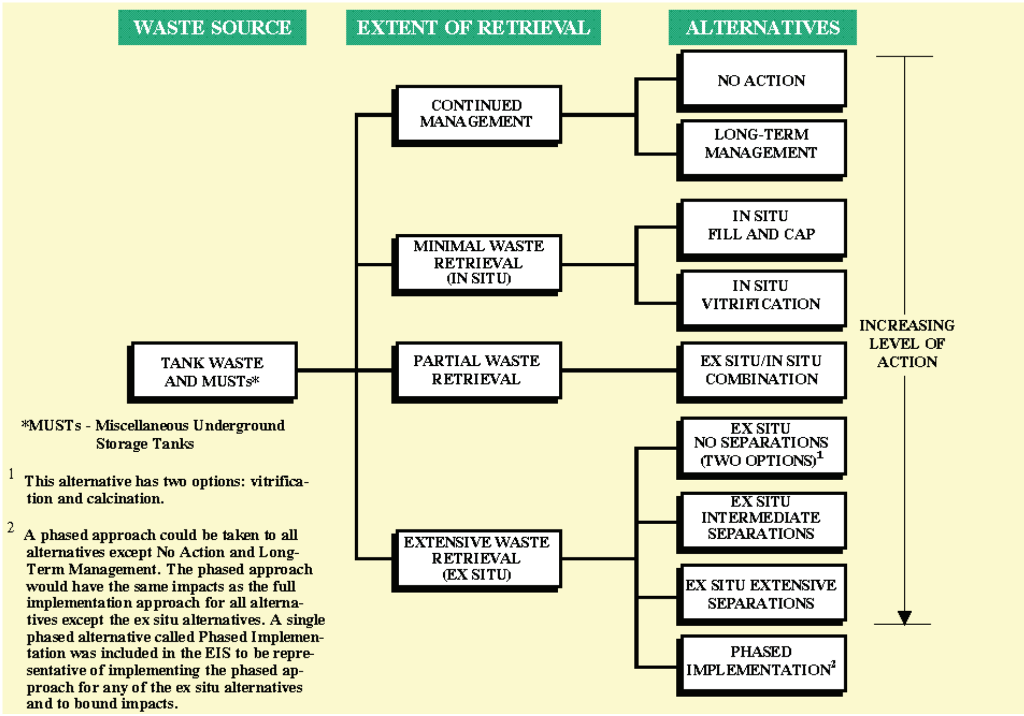


Figure 20. Tank Waste Remedial Alternatives (Reproduced from USDOE-RL 1996)

retrieval alternatives. Three alternatives were analyzed with different levels of separations. A fourth alternative evaluating the implementation of the extensive retrieval alternative in phases was also analyzed.

The TWRS EIS was prepared to support the decisions that must be made concerning safe storage and disposal of Hanford tank wastes (USDOE-RL 1996). One potential option for treating low-activity tank wastes upon retrieval is grouting; another is vitrification. Vitrification is also proposed for high-level wastes retrieved from the tanks. The EIS proposed that empty waste tanks be grouted instead of being removed entirely[51].

4.3.2.2 Important Assumptions and Conceptual Models

The EIS is typically posed at a higher level of analysis than the risk assessments performed under regulatory processes like CERCLA and RCRA although often the assessment processes are integrated (Shedrow et al. 1993). To allow for meaningful comparisons of the Hanford TWRS alternatives, a single and consistent method of closure was assumed. This method was closure as a landfill, which includes placing an earthen cap over the tanks after remedial actions have been completed (USDOE-RL 1996). The actual closure method selected will impact releases to the groundwater from residual waste and potential

[51]The USDOE plans to address tank farm closure issues in a second EIS to address alternatives for closing the tank farms that would look at these issues in much greater detail (USDOE-RL 1996).

health effects and the disturbances at potential earthen borrow sites (USDOE-RL 1996)[52].

The TWRS EIS also incorporated the assumption that immobilized high-level waste produced from retrieval and treatment actions would be disposed of at the candidate geologic repository at Yucca Mountain, Nevada (USDOE-RL 1996), which until recently was the only site under consideration. However, funding for Yucca Mountain has been cut and it may no longer be an option. Environmental impacts occurring at the repository are not addressed in this EIS but will be addressed when the new alternatives for spent nuclear fuel and high-level wastes are defined.

One assumption made for the use of cementitious materials is that grouting (phyiscal and chemical stabilization) would produce acceptable waste forms for the *ex situ* treatment of retrieved Hanford tank wastes and is thus a viable alternative for consideration. The EIS identified grouting as the preferred alternative for certain tank wastes. The second primary assumption is that grouting (physical stabilization of the tank and the chemical stabilization of the residual waste) would also be acceptable for tank closure after waste removal has been completed.

4.3.2.3 Relative Importance in Context of Assessment

The relative importance of cementitious materials to the alternatives evaluated in the TWRS EIS varies depending on whether these materials are used to close tanks or treat retrieved wastes. For example, grouting has already been applied to the operational closure of two former high-level waste tanks at the Savannah River Site (as described in the examples) and is the preferred method for closing high-level waste tanks at both Savannah River and Hanford. A more significant impact of cementitious materials is on the potential *ex situ* treatment of retrieved wastes from Hanford tanks. For example, solidification using a cementitious grout has been used at the Savannah River Site to treat low-activity wastes for disposal in the onsite Saltstone facility. Grout is a common treatment technology that has been employed in the management of hazardous and radioactive waste[52].

On the other hand, the Hanford Tri-Party Agreement specified vitrification as the preferred treatment method for low-activity wastes at Hanford based partially on the relative risks of the two treatment methods and not necessarily the acceptability of the risks provided by the methods[53]. 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 on safe and more economic treatment of retrieved wastes possibly including low-activity waste from Hanford.

4.4 Commercial Nuclear Facilities

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

The Big Rock Point Nuclear Power Plant near Charlevoix, Michigan was initially a research and development center to study life extension and efficiencies of different nuclear fuel combinations and to prove that large power reactors could be reliably used to generate electricity (ITRC 2008). In 1962, Big Rock Point became the nation's fifth commercial nuclear plant and the world's first "boiling water, direct-cycle, forced-circulation, high-power-density"

[52]Grouting tank wastes has been studied extensively at the Hanford and Savannah River sites for LAW disposal. Grouting was originally selected as the preferred treatment method for LAW in the Hanford Defense Waste EIS although the Tri-Party Agreement changed the method to vitrification. Hanford LAW included liquid tank waste (after separation) and secondary waste from the HLW vitrification facility.

[53]The risks of vitrified low-activity waste may be lower than those for the same wastes in an appropriate grouted form; however, this does not necessarily mean that the risks from the grouted waste form are unacceptable.

nuclear reactor facility to produce power (ITRC 2008). In 1997, it was determined that the small size of the plant made continuing operations uneconomic and that operations would be ceased even though there were still three years remaining on its license. When shut down in August 1997, Big Rock Point was the "oldest and longest-running nuclear plant in the United States" (ITRC 2008). The process for decommissioning the Big Rock Point facility was begun shortly after the plant was shut down.

Because of the small footprint of the Big Rock Point nuclear facility and the high value of the land, a "Greenfield" approach was selected for decommissioning (EPRI 2004)[54]. Before the plant was dismantled, the contaminated areas and components were decontaminated (Tompkins 2006). The spent fuel was removed to the spent fuel pool (and later to an independent spent fuel storage installation[55]) allowing dismantlement to begin. 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. The steam drum was removed and shipped by rail to the Envirocare Facility in Utah. The concrete reactor cavity inside the containment sphere was cut into pieces and the ventilation stack was dismantled. By April 2006, the containment sphere and turbine building were also demolished. More than 53 million pounds of low-level waste were shipped to disposal facilities in South Carolina, Tennessee, and Utah, and more than 59 million pounds of nonradioactive building materials were shipped to an industrial landfill in Michigan[56].

The company holding a reactor license must seek USNRC permission to decommission a facility. A Post-Shutdown Decommissioning Activities Report (PSDAR) must be submitted that describes how environmental impacts from decommissioning activities will be assessed. An application for termination of the license must be submitted to the USNRC for approval and to be accompanied by the License Termination Plan (LTP). The licensee must also demonstrate that the requirements of the License Termination Rule (LTR) (10 CFR §20.1401 et seq.) will be satisfied.

Because the intent was to release the Big Rock Point site for unrestricted use after decommissioning, the USNRC prepared an environmental assessment (EA) to evaluate potential environmental impacts (both radiological and non-radiological) (USNRC 2005).

4.4.1.1 Role of Cementitious Barriers and Processes Considered

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 0.25 mSv/yr (25 mrem/yr) LTR dose limit for unrestricted use (10 CFR §20.1402). 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)[57]. 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.

[54] In a "Greenfield" approach, all structures including those below grade (e.g., foundations and basements) are demolished and disposed of off-site.
[55] The independent spent fuel storage installation is the only structure remaining on-site after decommissioning.
[56] One unique aspect of the decommissioning approach was gaining approval under the alternate disposal regulations (10 CFR 20.2002) to ship slightly contaminated debris (including concrete) to a State of Michigan licensed landfill resulting in significant cost savings (EPRI 2004).
[57] 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.

The only area where cementitious materials impacted the analyses to support decommissioning of the Big Rock Point facility was for the dose assessment for transportation of the reactor pressure vessel to the Chem-Nuclear Systems, L.L.C, Barnwell, SC Low-Level 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[58] to demonstrate the design of the new transportation cask complied with all of the 10 CFR 71 criteria for a Type B package (BNFL 2001). In these dose calculations, credit was taken for the concrete used to fill voids in the pressure vessel as well as the annular space between the vessel and package. The computations demonstrated that the package satisfied the 10 CFR 71 criteria for a Type B package (BNFL 2001), which was confirmed by the USNRC (USNRC 2002).

4.4.1.2 Important Assumptions and Conceptual Models

The dose assessments performed to support the license termination plan for the Big Rock Point (including defining DCGLs for the final survey) were predicated on a modified resident farmer scenario using the RESRAD code (Yu et al. 2001). In this scenario, the receptor is exposed to residual radioactive material in the surficial and subsurface soils as well as in three groundwater zones at maximum concentration levels. The farmer moves onto the site after closure, grows his or her diet in a garden, and uses water from the aquifer under the site. The additional assumptions made to evaluate doses to the resident farmer include (CEC 2004):

- Residual radioactive contamination is found in the surface soil, subsurface soil, and three groundwater zones.
- Residency can occur immediately after release of the property.
- The property will not be used for livestock or dairy animal production.
- Radioactive doses can result from exposure via external exposure, inhalation, and ingestion pathways.

Assumptions were made to intentionally overestimate the doses to the resident farmer. The results of the dose assessment allowed the concrete and debris from decommissioning activities to be sent to off-site disposal units and the final survey (when compared to the DCGLs) allowed the USNRC to release the Big Rock point site for unrestricted use. However, none of the assumptions made to perform the dose assessments to support the license termination plan involved cementitious materials or their properties.

A separate dose assessment was performed to demonstrate that the design of the transportation cask used to transport the Big Rock Point reactor pressure vessel to the Barnwell disposal facility complied with all of the 10 CFR 71 criteria for a Type B package. In these dose calculations using ISOSHLD-PC, important assumptions included (BNFL 2001):

- The relative radionuclide abundances from the Trojan Nuclear reactor vessel activation activities provide a reliable and complete radionuclide profile for the Big Rock Point reactor vessel assembly and internals (RVAI). Use of the Trojan radionuclide abundances result in conservative estimates for shield thickness because the Trojan

[58]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).

Co-60 inventory is higher and this is the dominant radiation source for shielding calculations.

- The annular region between the RVAI and the transport cask steel shielding is filled with low density cellular concrete with a minimum density of 800 kg/m^3 (50 lb/ft^3). The concrete in the RVAI 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.
- All reactor vessel components are made of 304 stainless steel, the vessel wall is made of carbon steel, and the shield material for the transport package is made of pure iron.
- Radionuclide activity is homogeneously distributed within the component or component portion under study.
- The radiation source term includes 1 curie from surface contamination with the same radionuclide distribution as that used for activation source terms.

In summary, shielding credit was taken for the low density cellular concrete used to fill voids in the pressure vessel and the annular space between the reactor vessel and package. The computations demonstrated that the package satisfied the 10 CFR 71 criteria for a Type B package (BNFL 2001), which was independently confirmed by the USNRC (USNRC 2002).

4.4.1.3 Relative Importance in Context of Assessment

In the model and dose assessments performed to support the Big Rock Point license termination plan and demonstrate that the requirements of the LTR were met, cementitious materials (i.e., low density cellular concrete) were considered. The detailed shielding calculations that were used as the basis for demonstrating that the transport package used for the reactor pressure vessel satisfied 10 CFR 71 criteria took credit for the cement both in the reactor vessel and in the annular space between the vessel and the package (BNFL 2001). In August 2003, the reactor vessel was removed, placed in the transport package, the voids and annular space were filled with concrete, the package was welded shut, and the package containing the reactor vessel was shipped to Barnwell, SC where it was disposed of as low-level waste in October 2003[59]. On the other hand, because a "Greenfield" approach was taken to decommissioning the Big Rock Point facility and all concrete and other debris was to be disposed of off-site, no impacts from these cementitious materials were included in the dose assessments used to demonstrate compliance with the LTR requirements for unrestricted release. In 2007, the USNRC released most of the Big Rock Point Nuclear Plant site for unrestricted public use.

To demonstrate compliance with 10 CFR 71 criteria, dose rates for both shielded and unshielded conditions are determined at 1 and 3 meters from the package. Both sets of calculations took credit for concrete in the reactor vessel; however, no credit was taken for concrete in the annular space for unshielded dose rates. Because no shielding calculations were run without consideration of cementitious materials, it is difficult to characterize the specific impacts of these materials on the dose rate results. However, the amount of concrete in the reactor vessel voids probably had a small shielding impact relative to the assumptions made for the source term (specifically Co-60).

[59]The residual radioactive source term for Big Rock Point reactor vessel was well below the Chem-Nuclear Systems, L.L.C, Barnwell, SC disposal limit of 40,000 Curies (BNFL 2001) so the concrete inside the reactor pressure vessel and transport package was not accounted for in terms of the disposal itself.

4.4.2 Spent Fuel Pools

Spent nuclear fuel pools are constructed to meet USNRC requirements, and typically are 9 to 18 m (30 to 60 feet) long, 6 to 12 m (20 to 40 feet) wide, and 12 m (40 feet) deep with 1.2- to 1.8-m (4- to 6-foot) thick steel-lined concrete walls and floors (GAO 2005). Commercial nuclear power reactors in the United States are of two basic types: boiling water or pressurized water reactors[60]. 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 as illustrated in Figure 21. Pools tend to be located in external structures on or partially embedded in the ground for pressurized water reactor as illustrated in Figure 22. Regardless of reactor type or location, the storage pools must be constructed to protect the public against radiation exposure.

The decommissioning of the Big Rock Point nuclear facility provides an example of how a spent nuclear fuel pool may be decommissioned as part of the overall strategy for the facility. In that case, the storage racks and pool liner were completely removed as part of the overall plan and the site was released by the USNRC for unrestricted use under a “Greenfield” approach to decommissioning. Credit for cementitious materials was accounted for only in the certification of the transport package used to ship the reactor pressure vessel to the Barnwell low-level waste disposal facility in South Carolina. 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). At the Idaho Site, four spent fuel pools have either been decommissioned or are in the process using this method. Dresden Generating Station Unit 1, which began operations in 1960, was the first full-scale, privately-financed nuclear plant in the US[61]. Dresden Station Unit 1 was retired in 1978 and has been declared a Nuclear Historic Landmark[62]. Unit 1 is a boiling water reactor with a spent fuel pool that is assumed to be in the configuration indicated in Figure 21 making a “Greenfield” approach to decommissioning the fuel pool impossible.

In 2004, Exelon decided to reduce the risk of further fuel pool leakage by cleaning, draining, and coating the spent fuel pool (Demmer et al. 2006). The original Exelon approach was to use long-handled tools and coat the pool as the water level was decreased. This approach posed significant health and safety concerns from potentially high levels of airborne contamination over the long period of time it would require to drain and coat the pool. The INL approach that had been successfully used onsite consists of applying an epoxy-based coating to the pools and floors while underwater. Thus the INL method, while also requiring extensive environmental, health, safety, and engineering efforts including an underwater team with nuclear experience greatly reduced airborne contamination and corresponding health concerns. The INL option

[60]A boiling water reactor uses steam generated in the reactor to drive a turbine and generate electricity; the steam condenses to water that is returned to the reactor repeating the cycle. On the other hand, a pressurized water reactor sends pressurized water to a steam generator creating nonradioactive steam in a separate loop. The slightly radioactive water returns to the reactor where the cycle is repeated (USGAO 2005).

[61]For details, see http://www.exeloncorp.com/ourcompanies/powergen/nuclear/dresden_generating_station.htm (accessed March 20, 2009).

[62]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).

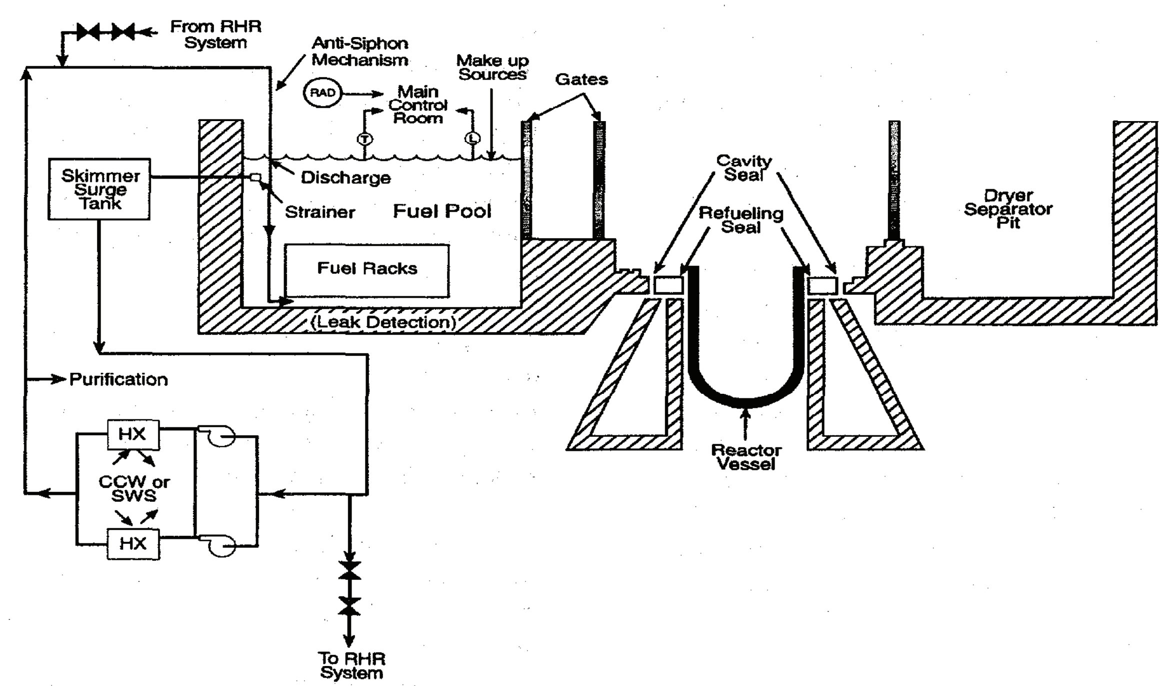


Figure 21. Boiling Water Reactor (BWR) Spent Fuel Cooling Systems (Reproduced from Ibarra et al. 1997)

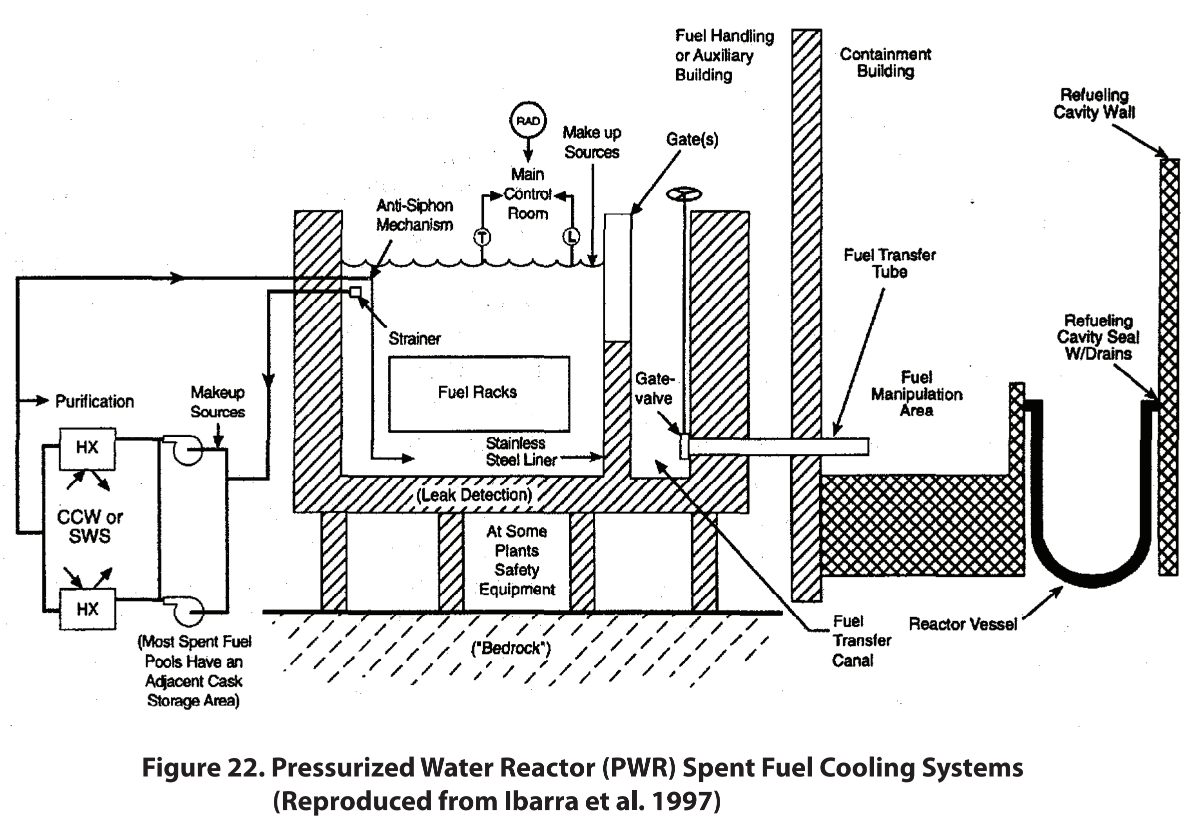


Figure 22. Pressurized Water Reactor (PWR) Spent Fuel Cooling Systems (Reproduced from Ibarra et al. 1997)

did not appear at first to be safer from industrial and radiological perspectives, but INL has demonstrated statistically that the method is safe (Demmer et al. 2006). 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 spent nuclear fuel pools, it appears that including the cementitious components into the models would not significantly impact the decisions made. However, when alternatives are considered that may leave contaminated cementitious materials onsite analogous to the entombment activities at the Idaho and Hanford site, the explicit representation and accuracy of the properties and performance of cementitious materials may become critical factors in the decision-making process.

4.4.2.1 Containment Performance for Spent Nuclear Fuel Pools

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 structural and thermal properties may improve the assessment of consequences (e.g., doses to the public) from these accident events. However, the likelihood of these events is typically very low and thus the ability to better assess the magnitude of contaminant releases associated with the occurrence of an accident appears limited in affecting decisions concerning spent fuel pools.

Spent nuclear fuel pool leakage has resulted in the release of radioactive water to the environment at several NPP's. For example, in July 2005, seepage from the spent nuclear fuel pool was observed at the Palo Verde NPP site. Blocked lines in the spent nuclear fuel pool tell-tale drain caused water to back up and leak through two adjacent concrete walls.

Again in 2005, leakage of radioactive water was identified from the Unit 2 spent nuclear fuel pool at the Indian Point NPP. A hairline crack with moisture was discovered along the south concrete wall of the spent nuclear fuel pool. Although initial samples did not detect any radioactivity, a month later contamination was first detected in a sample from the crack. A second crack was discovered two weeks later and a temporary collection device was installed to capture leaking liquid. Analysis of the moisture indicated that the material had the same radiological and chemical properties as pool water. Leak from the crack increased to a maximum of 1-2 liters per day and remained stable declining to a minimal amount three months later.

To assess the resulting contamination, the Indian Point licensee contracted geotechnical and groundwater consultants to assist in mapping the contaminant plumes. Based on the studies, the licensee concluded that the releases did not pose a risk to public health, and at the most may have resulted in a radiation dose to the public of well below 1 mrem for tritium. For strontium-90 releases, the dose may be higher but still below the NRC's 10 CFR Part 50, Appendix 1 ALARA values.

In 2003, tritium was detected in shallow ground water on-site near the Salem Unit 1 NPP. Contaminated water leaked through a concrete wall into the Unit 1 Auxiliary Building. The contamination was due to Unit 1 pool water that had leaked into a narrow "seismic gap" and entered the Auxiliary Building. The source of the abnormal release was identified as clogged drains in the Salem Unit 1 spent nuclear fuel pool. Later, the clogged drains were repaired which stopped the leak. The licensee has reported that there is no evidence of tritium concentrations exceeding limits.

5.0 SUMMARY OF MODELING APPROACHES

A broad perspective of different approaches for considering performance of cementitious barriers in PAs and PA-like analyses serves as a good illustration of the need for improved communication and sharing of knowledge. The examples provided perspective regarding the frequency with which conservative, simplifying assumptions are made in lieu of trying to defend the assumptions necessary to take credit for specific degradation processes. This appeared to be the case more often in PA-like assessments rather than traditional PAs, which reflects the fact that PAs have been dealing with cementitious barriers as part of disposal facilities for many years. People from other regulatory environments traditionally have focused on clean-up situations, where cementitious barriers are not as important. To provide some additional focus on the PA-like regulatory environment, a brief summary of the regulations is provided below. This is followed by a summary of the examples and a comparison of the approaches used.

5.1 Overview of Regulations for PA-Like Analyses

The cornerstones of the USDOE 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 USDOE but instead by the USEPA and the states (NAS 2006). Whereas PAs are required under DOE 435.1, 10 CFR 61, Section 3116 and the AEA, the other laws require different types of assessments. Because the License Termination Rule (LTR; 10 CFR Part 20 Subpart E), which is administered by the USNRC, also does not require a performance assessment, it was also examined in this section.

The laws related to PA-like analyses that do not require a formal performance assessment are listed along with the assessment methods in Table 8.

Because multiple laws (including CERCLA, RCRA, and NEPA) may be applicable to the same contaminated site, numerous policies have been adopted in the DOE Complex for integrating these laws and their assessments (Cook 2002; Shedrow et al. 1993; USDOE 1994a). The performance of NEPA environmental assessments and impact statements are part of the decommissioning process and demonstration of compliance with the LTR.

For the three laws administered by the USEPA, there are no specific legal requirements regarding the approaches that must be used for assessments when cementitious barriers are present. Although NEPA does require that all "reasonable" alternatives be considered during the Environmental Impact Statement

Table 8. Summary of Regulations Relevant for PA-Like Analyses

Regulation	Description	Assessment-Related Requirements	Requirements for Cementitious Barriers
Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA)	Agency: USEPA Purpose: identify and remediate sites where hazardous substances were, or could be, released into the environment Applies: all Federal agencies	Preliminary site assessment Remedial investigation (RI) and Baseline risk assessment Feasibility study (FS) Record of Decision (ROD)	No specific requirements. Credit may be taken per guidance documents
Resource Conservation and Recovery Act (RCRA) (Subtitle C)	Agency: USEPA Purpose: protect human health and environment via comprehensive approach to hazardous and solid waste management at operating facilities Applies: hazardous waste treatment, storage, or disposal facilities and transporters of hazardous wastes	RCRA Facility Assessment (RFA) RCRA Facility Investigation (RFI) Corrective Measures Study (CMS) and corrective measure selection Corrective Measures Implementation	No specific requirements. Credit may be taken per guidance documents
National Environmental Policy Act (NEPA)	Agency: USEPA Purpose: insert environmental considerations into federal decision-making and increase public involvement Applies: all Federal agencies in Executive branch	CATegorical EXclusion (CATEX) Environmental Assessment (EA) and Finding of No Significant Impact (FONSI) Environmental Impact Statements (EIS) including Draft EIS for public comment, Final EIS, and ROD—focus is on the alternatives analysis	No specific requirements. Requires all "reasonable" alternatives be considered for EIS
License Termination Rule (10 CFR Part 20 Subpart E)	Agency: USNRC Purpose: provide radiological criteria for license termination Applies: decommissioning of facilities (or parts of facilities) licensed by USNRC	Dose assessment for restricted release or unrestricted release of facility	No specific requirements for dose assessment, but detailed guidance is provided in NUREG-1757. Release of contaminated solid materials regulated on a case-by-case basis

(EIS) process. The EIS is where alternatives including barriers or grouting are often considered for action and evaluation. Demonstration of compliance with the USNRC 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)). There are no specific requirements for cementitious materials when performing the LTR dose assessment to determine site release characteristics[63].

One type of assessment that is common to the CERCLA, RCRA, NEPA, and 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. Because the conversions from exposure or intake dose to response (e.g., cancer risk or total effective dose equivalent) are determined by regulatory fiat, the primary factors determining dose or risk are the exposures. Thus the key assumptions and parameters in the risk and dose assessments often pertain to the necessary exposure modeling including the source term and release characteristics, fate and transport, and exposure scenario (e.g., resident or intruder), which are essentially the same as what is considered for a PA.

5.2 Summary of Approaches Used for Cementitious Barriers

Examples are provided demonstrating how risk and dose assessments have been performed to support the management of LLW disposal facilities, D&D of large facilities, remediation of 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 or risk limits are mandated. As illustrated in Table 9, the credit taken for cementitious materials in the modeling performed to support the assessments typically impacts the allowable source term, release, and near field transport conditions. Of the various approaches represented in this section, tiered and iterative approaches consistent with CERCLA guidance (USEPA 1989a) and with PA recommendations from the DOE, NRC, and IAEA are considered excellent practice (Brown 2008).

Given that cementitious materials are engineered features, the key assumptions tend to be related to the source release and near field transport. In the examples provided in this section, the credit taken for cementitious materials ranged from no credit to considerable credit for physical and chemical properties, including timing of degradation. A summary of information from the examples is provided in Table 9.

One consistent theme running through the various dose and risk assessments performed in the example cases presented in this section was that gross simplifying assumptions were often made even when cementitious materials were considered in the assessment process. Conservative assumptions were often made because of a lack of site and facility-specific information for the cementitious materials or for expediency to avoid having to defend the assumptions associated with more detailed consideration. Since the results were acceptable as is, it was not deemed necessary to delve into more detail. This approach works well for many cases, but such assumptions add conservatism that could limit potential future activities.

The examples show that cementitious materials provide two different functions: (i.e., physical barriers and chemical barriers). In general, the role as a physical barrier is shorter-term than the role as a chemical barrier. This is consistent with the findings of Seitz and Walton (1993) that recommended that

[63]However, there are specific requirements imposed by the USNRC on the release of contaminated solid materials including building concrete from licensed facilities. Such material can be removed if the facility license is terminated based on meeting the 0.25 mSv/yr (25 mrem/yr) LTR dose limit for unrestricted use (10 CFR §20.1402).

Table 9. Summary of Examples of Assessments

Example	Description	Role of Cementitious Barriers and Processes	Important Assumptions and Conceptual Models	Relative Importance of Cementitious Materials
INTEC Tank Farm (Idaho Site)	Tank Closure under Section 3116. Tanks cleaned to maximum extent practicable and filled with grout.	Voids in tanks filled with grout, many tanks surrounded by concrete walls. Cementitious materials assumed to serve as physical and chemical barriers.	Multiple degradation mechanisms quantitatively assessed. Physical failure of concrete represented as step change in hydraulic conductivity. Timing based on conservative degradation scenario. Chemistry assumed unchanged.	Reducing conditions in cementitious materials were significant factor. Hydraulic properties important early, but degradation expected to occur later than assumed.
Radioactive Waste Management Complex (Idaho Site)	LLW disposal facility managed in accordance with DOE Order 435.1.	Cementitious materials used in vaults and containers. No credit taken for cementitious materials, except releases are diffusion controlled for one type of concrete cask container.	Diffusion assumed to occur without considering tortuosity or chemical effects.	Diffusion controlled release with conservative diffusion rate was sufficient to demonstrate compliance for cask containers. No credit needed for other cementitious barriers.
Integrated Disposal Facility (Hanford Site)	Combination LLW and RCRA waste disposal facility managed respectively under DOE Order 435.1 and RCRA.	"Treated" LLW form assumed to be grouted. Diffusion controlled release assumed for grouted waste.	Most probable and conservative diffusion coefficients were developed for each key species. The diffusion coefficients account for tortuosity and chemical reactions in the cementitious material.	Diffusion controlled release sufficient to contain radionuclides. Overall grouted waste not a major contributor.
Solid Waste Storage Area 6 (Oak Ridge)	LLW disposal facility managed in accordance with DOE Order 435.1.	Cement silos and tumulus pads with concrete containers used for disposal. Cementitious materials are assumed to function as physical and chemical barriers.	Detailed coupled structural and degradation modeling conducted to predict onset of cracking, which is assumed to compromise role as a physical barrier in a step change. Diffusion and chemical reactions in cementitious materials also considered with K_ds and solubilities.	Results were shown to be sensitive to several parameters associated with cementitious materials. Performance was deemed sufficient, even with assumption of total failure as a physical barrier at the onset of cracking.

Table 9. Summary of Examples of Assessments (contd)

Example	Description	Role of Cementitious Barriers and Processes	Important Assumptions and Conceptual Models	Relative Importance of Cementitious Materials
F Tank Farm (Savannah River Site)	Tank Closure being conducted under Section 3116	Multiple tank designs, in general with steel liners inside concrete walls and tanks filled with grout after cleaning. Cementitious materials assumed to serve as physical barrier to water flow and to chemically limit releases of radionuclides and also to delay onset of corrosion of steel tank.	Multiple degradation mechanisms were considered, including physical changes and chemical changes in the cementitious materials. Distributions of degradation times were developed for changes in hydraulic conductivity and for changes from reducing to oxidizing conditions.	Results were dependent on performance of the cementitious materials in delaying the onset of corrosion of the steel tank. The chemical properties of the cementitious materials were important after failure of the tank.
E-Area (Savannah River Site)	LLW disposal facility managed in accordance with DOE Order 435.1	Multiple disposal concepts using different types of cementitious barriers. Cementitious materials serve as physical and chemical barriers. Cracking is assumed to compromise performance as a physical barrier.	Structural and degradation models were used to determine timing of cracking and failure of cementitious materials. Transitions from reducing to oxidizing conditions were also calculated.	The grout used for components in grout trenches was important in terms of limiting releases of tritium. The vault walls are assumed to maintain a physical barrier until after the time of compliance, which precludes significant releases.
Engineering Test Reactor (Idaho Site)	Decommissioning under a non-time-critical CERCLA removal action. ETR reactor vessel removed and disposed on-site	Voids in pressure vessel were filled with grout for on-site disposal. Credit taken as a means to limit subsidence and resulting impact on water movement through cap.	None made specific to cementitious materials other than voids are filled to preclude subsidence that would increase infiltration rate through the cover.	Performance and properties of the grout actually used in disposal were inconsequential in the risk assessments and modeling performed.
Radioactive Waste Management Complex (Idaho Site)	Closure under the CERCLA remedial investigation/ feasibility study (RI/FS) process	No credit taken for diffusional release from concrete or the effect of containment in concrete casks in final baseline risk assessment. Some credit taken in previous assessments.	Cement forms treated as soil for modeling release for materials with surface contamination leached by infiltrating water and controlled by partitioning between the waste form and water.	Performance and properties of cementitious materials were inconsequential in the risk assessment modeling.

Table 9. Summary of Examples of Assessments (contd)

Example	Description	Role of Cementitious Barriers and Processes	Important Assumptions and Conceptual Models	Relative Importance of Cementitious Materials
Waste Calcining Facility (Idaho Site)	Landfill closure under RCRA supported by NEPA Environmental Assessment (EA)	Credit taken in detailed modeling phase (using PORFLOW) for grouting and concrete cap including cracking. No credit taken in initial screening phase (using GWSCREEN).	In the detailed modeling phase, cementitious materials impact source release and transport when estimating risks to the future resident.	Significant impact on predicted groundwater concentrations and risks and provided assurance that landfill closure would be protective of human health.
Tanks 17-F and 20-F (Savannah River Site)	Operational closure under SCDHEC industrial wastewater permits supported by NEPA Environmental Impact Statement (EIS)	Credit taken for grout and concrete in modeling fate and transport (using MEPAS) of residual contaminants from grout to the aquifers and receptors.	Basemat, grout, and tank top remain intact for 1,000 years and then fail instantaneously resulting in significant increases in hydraulic conductivities and infiltration rate.	Not taking credit would likely result in predictions that violate performance objectives—properties and performance of these materials likely important to the risk analysis.
P Reactor (Savannah River Site)	In-Situ Decommissioning under CERCLA	Concrete and grout are physical and chemical barriers controlled by the assumed hydraulic conductivity and distribution coefficients.	Concrete and grout behave as porous media. Hydraulic conductivity changes as a function of time. Distribution change as concrete or grout ages.	Grout-water distribution coefficient for Ni was also shown to be important to risk.
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	Credit taken for grouting as a “defense-in-depth” measure if the engineered barrier fails during the 1,000-yr simulation period	No assumptions pertain to the use of cementitious materials or their properties even though vessels will be grouted prior to cap emplacement.	Properties and performance of these materials are not relatively important; they only 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	A potential option for treating retrieved low-activity tank wastes is grouting and the EIS proposes that empty waste tanks be grouted instead of being removed entirely.	Grouting would produce acceptable waste forms for ex situ treatment of wastes and would be acceptable for tank closure after waste removal operations are complete.	Use of these materials for disposal could have a large impact in the future, safe and economic treatment of retrieved wastes possibly including Hanford LAW.

Table 9. Summary of Examples of Assessments (contd)

Example	Description	Role of Cementitious Barriers and Processes	Important Assumptions and Conceptual Models	Relative Importance of Cementitious Materials
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.	Considered for the dose assessment supporting certification of the cask used to transport the reactor pressure vessel to the Barnwell low-level disposal facility.	Shielding credit taken for the low density cellular concrete used to fill voids in the pressure vessel and the annular space between the reactor vessel and package.	Concrete in reactor vessel voids likely to have small shielding impact relative to the assumptions made for the source term in the analysis that allowed certification of cask.

cementitious barriers be designed to function physically over the short term to effectively contain short-lived radionuclides and be designed to function over longer times to limit the release rate for longer-lived radionuclides. In practice, the modeling has focused on both aspects.

Physical failure tends to be represented as a change in bulk hydraulic conductivity resulting from cracking. Prior to cracking, it is generally assumed that releases are controlled by diffusion, although different assumptions have been made for diffusion rates. Because cementitious waste forms can be considered a diffusion barrier to contaminant release, cracking is often critical as it alters the flow of water and vapor through the waste form (increasing the potential for leaching) and the diffusional properties[64].

Because of the difficulties in quantifying the extent and impact of cracking, in cases where physical properties of cementitious barriers were considered, gross simplifying assumptions were often made, e.g., the cementitious barriers fail completely at the onset of through-wall cracking. A variety of different approaches were used to identify the onset of cracking. From the examples, it appears that there is still a lack of confidence regarding being able to take credit for more gradual changes as cracking progresses, but that lack of confidence does not appear to have a negative impact on the conclusions of the assessments.

From a chemical barrier perspective, the most common consideration has been the use of K_ds that account for the waste stabilization properties of cementitious materials. The examples provided many cases where the presence of reducing conditions in a grouted waste was an important consideration for the results of the assessment. More recently, solubilities are also being developed for specific radionuclides stabilized in cementitious matrices. There have been substantial successes in the use of these types of assumptions. This illustrates the apparent improved confidence related to taking credit for long term performance from a chemical perspective as opposed to the remaining concerns regarding taking credit for evolution of cracking over time.

Parameter uncertainties and temporal degradation and the resulting effects on properties for the cementitious materials are often not taken into account or over simplified although they can have significant impacts on predictions used to characterize doses and risks for decision-making purposes. Improvements in both the characterization and modeling of these phenomenological properties for cementitious materials used in disposal will provide more accurate predictions and support their continued use in future disposal and other nuclear-related activities in the USDOE. For example, one major 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 have been adequate for Hanford LAW[65]; however, the extensive work performed on vitrified waste forms for high-level waste (HLW) provided the assurance needed for stakeholders to rely on these waste forms for both Hanford HLW and LAW. 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.

[64] 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" (Walton 1992). When cracked, concrete cannot be relied upon for contaminant isolation.

[65] SRS LAW is currently treated via grouting.

6.0 CONCLUSIONS AND MODELING/DATA NEEDS

Cementitious materials have been used in disposal applications regulated under various federal regulations including USDOE, IAEA and USNRC 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 the USEPA and the LTR do not require performance assessments, but can include calculations similar to a performance assessment for more complex situations. Although, there can be different goals and frameworks for these different applications, there are many similarities and experiences that can be shared. There is a critical need to create a means to share information regarding the lessons learned and good practices associated with modeling of cementitious barriers for all of 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. There are more guidance documents beginning to be developed, primarily by the USNRC. A significant area of need is to update existing guidance to account for the latest developments and to make that guidance useful across the spectrum of different types of assessments that are being conducted, recognizing the different goals and philosophies applied for those assessments.

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.

From a technical perspective, significant advances have been made in the consideration of the role of cementitious barriers as chemical barriers, although consideration of cracking in the context of physical properties remains a significant challenge. There remains a tendency to make gross simplifications in the context of performance of cementitious barriers as a physical barrier to flow and in many cases as a chemical barrier as well. Improving both the characterization of the properties of these materials and the accuracy of the models used to predict their performance, especially over long assessment periods, would increase the applicability of cementitious materials for nuclear applications.

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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.

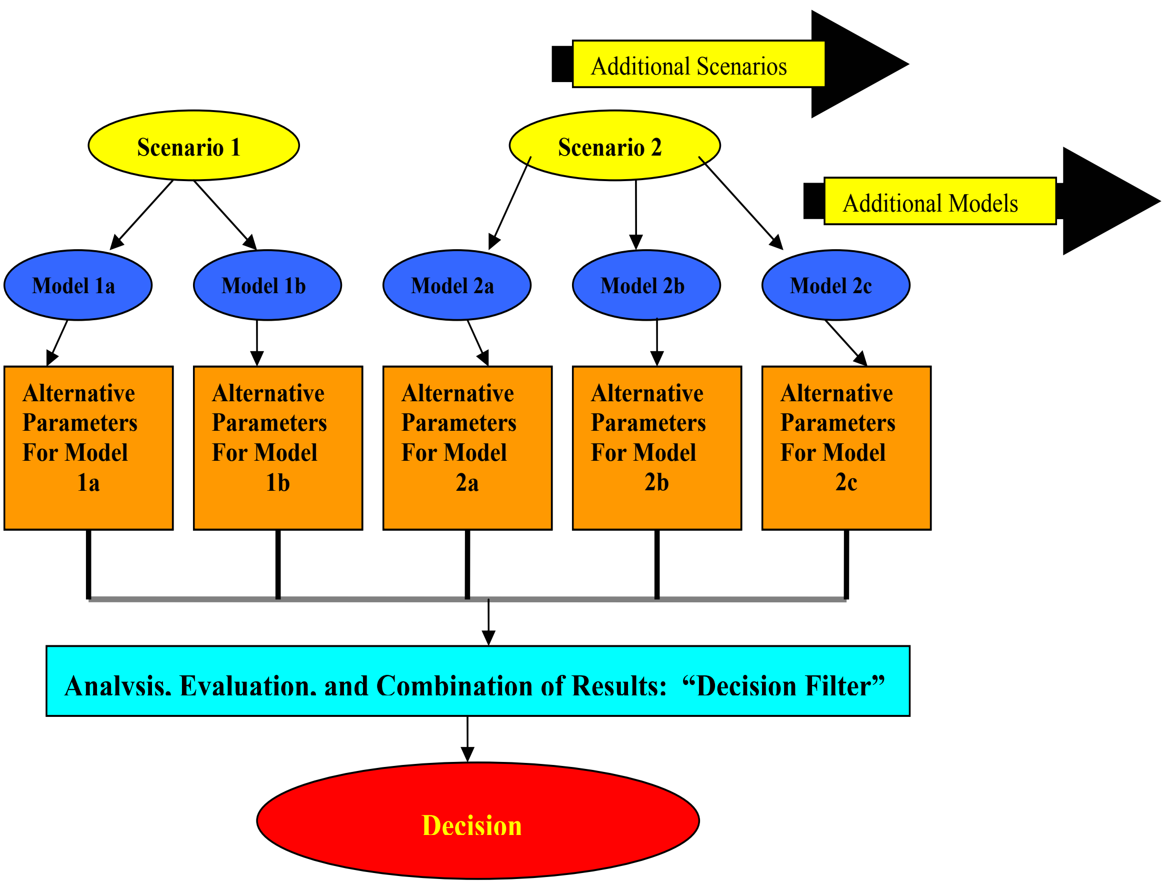

Figure 1. Representation of Approach to Address Scenario and Conceptual Model Uncertainty (Kozak et al. 1993)

[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.

[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.

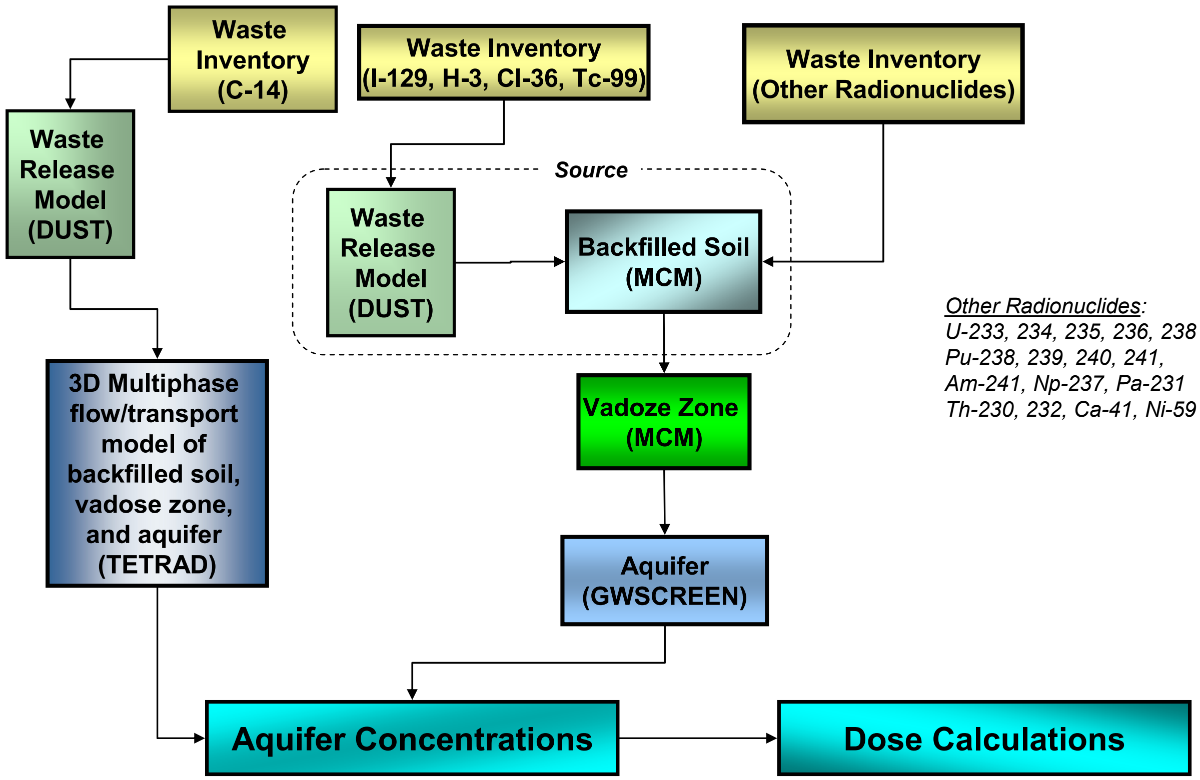

Figure 2. Idaho RWMC Modeling Approach (DOE NE-ID 2007)

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

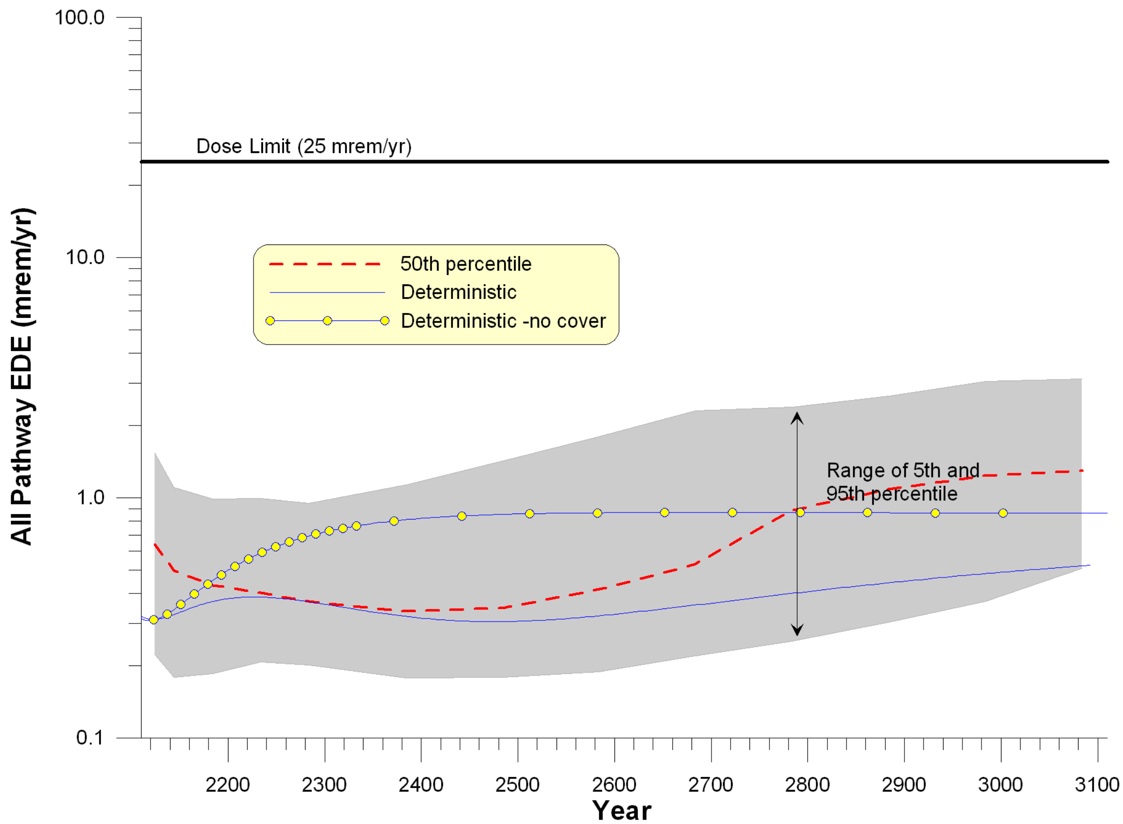

Figure 3. Uncertainty Analysis Results from the RWMC PA (DOE-NE/ID 2007)

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.

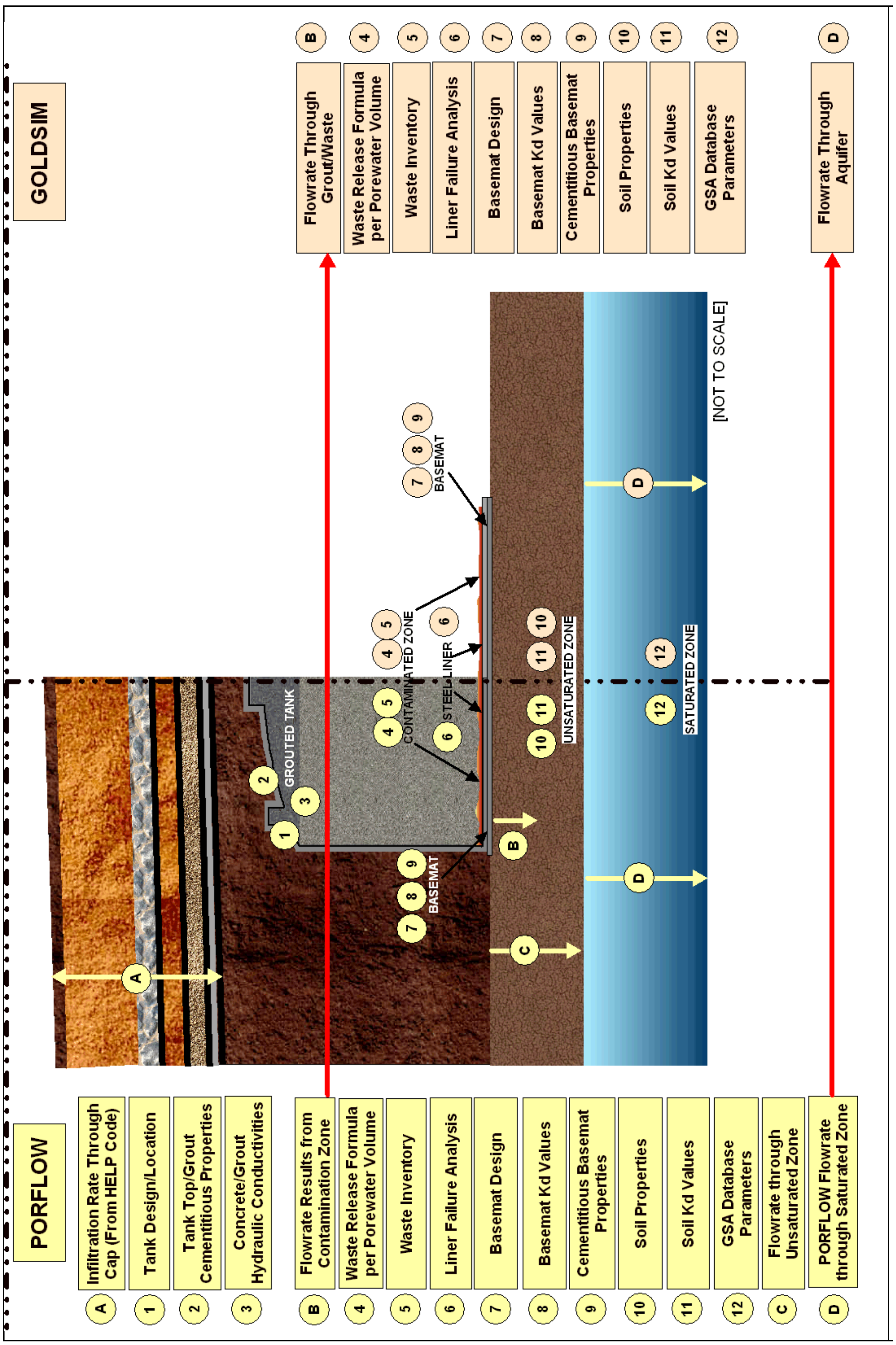

Figure 4. Groundwater Pathway Models for the SRS F-Tank Farm (SRS 2008)

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

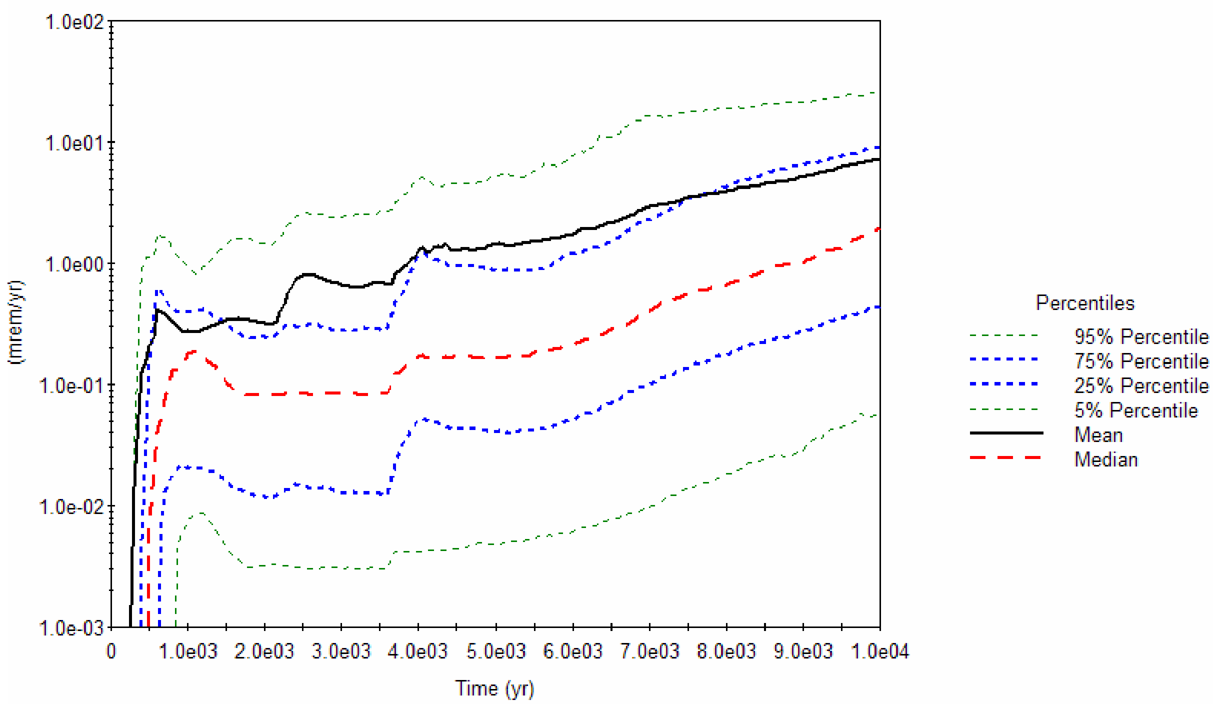

Figure 5. SRS F-Tank Farm PA Maximum Exposure Results for the 10,000 Year Compliance Period (SRS 2008)

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

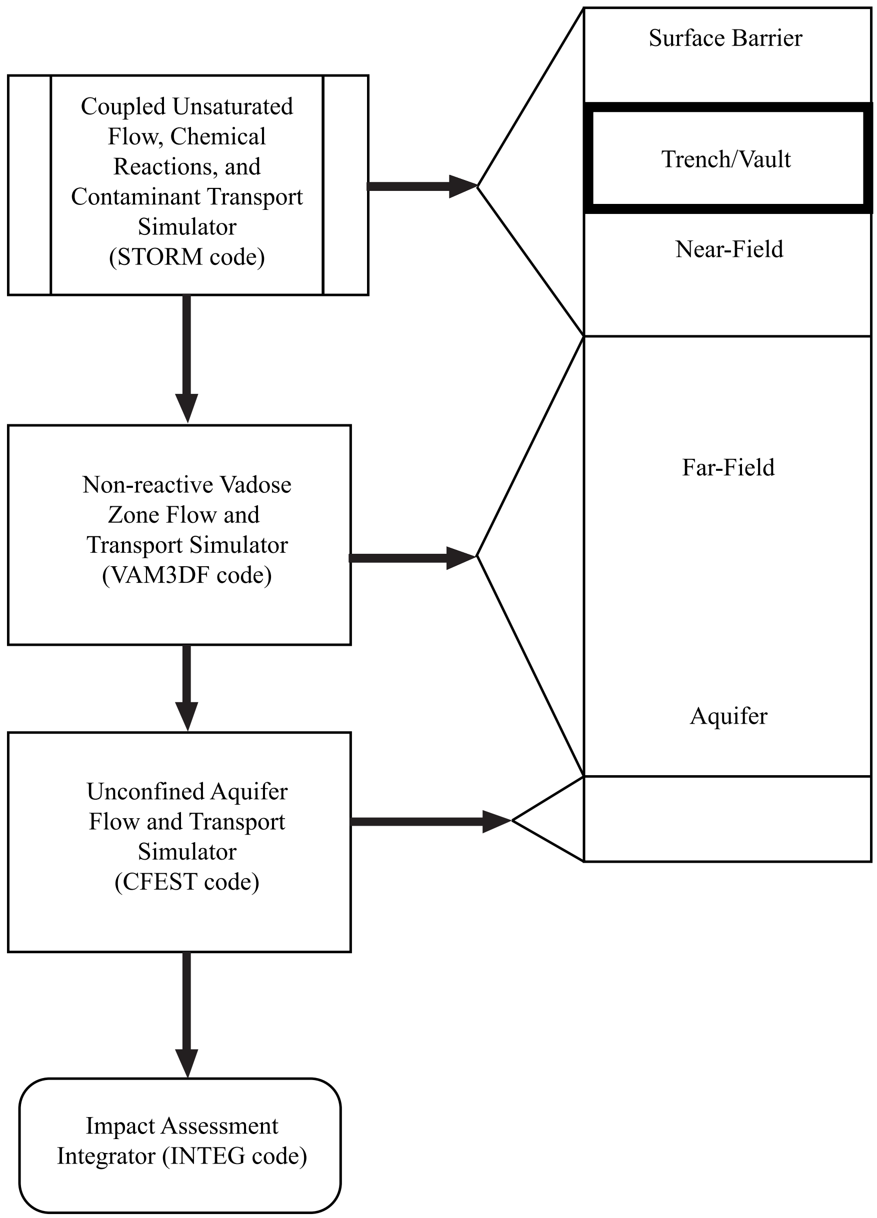

Figure 6. Hanford IDF PA Modeling Approach (Mann et al. 2005)

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

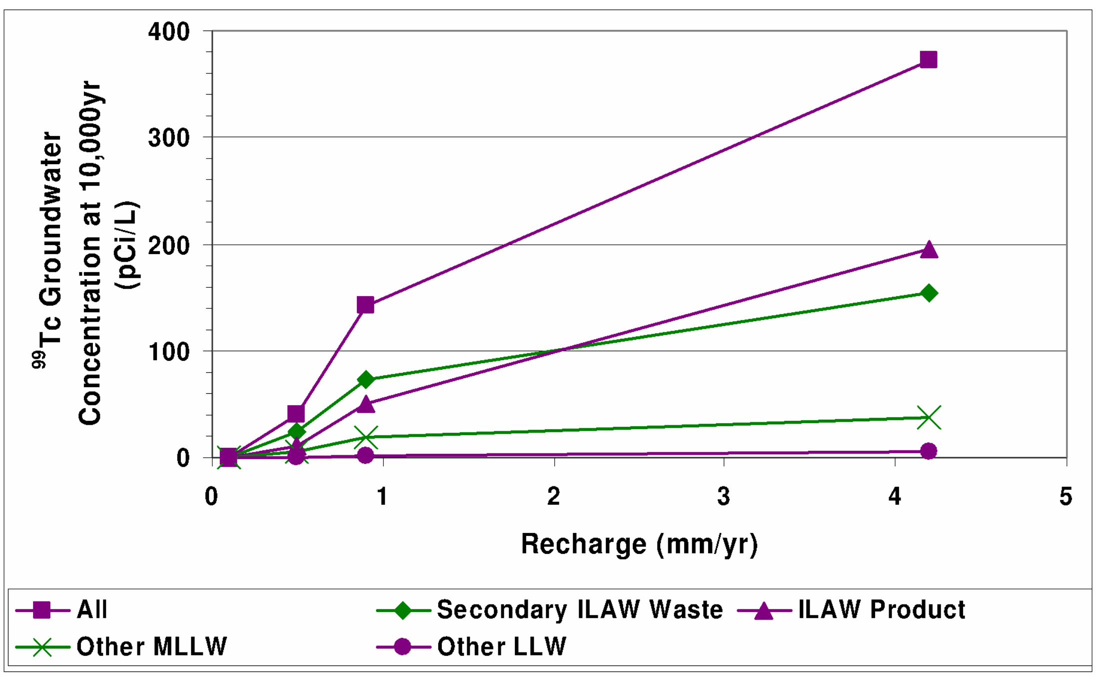

Figure 7. Sensitivity of Tc-99 Concentration in Groundwater at 10,000 Years to Changes in Recharge Rate (Mann et al. 2005)

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.

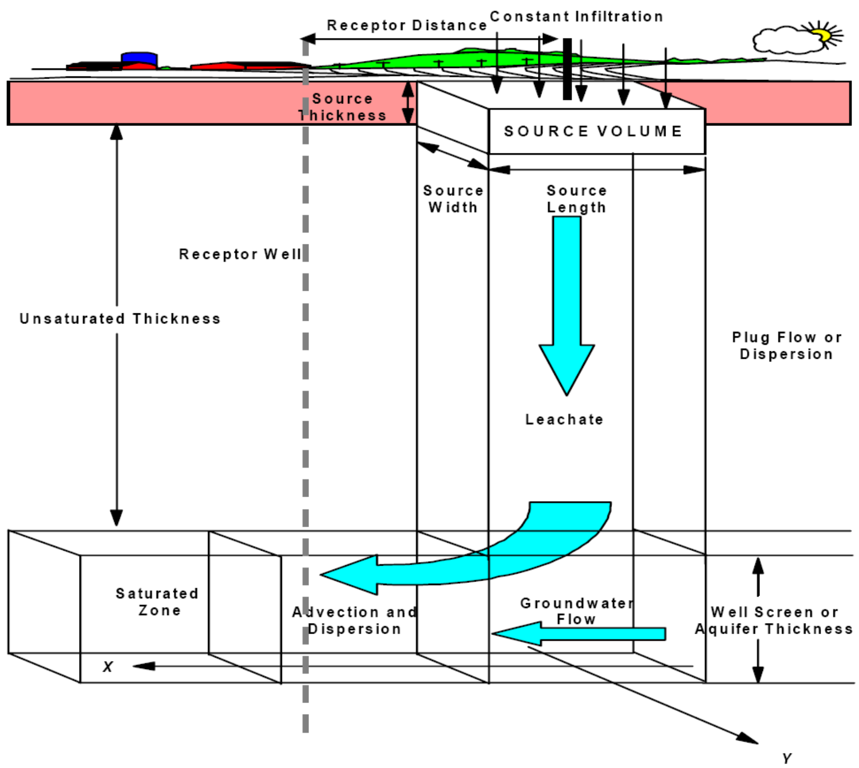

Figure 8. GWSCREEN Conceptual Groundwater Model (reproduced from McCarthy 2006)

- 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

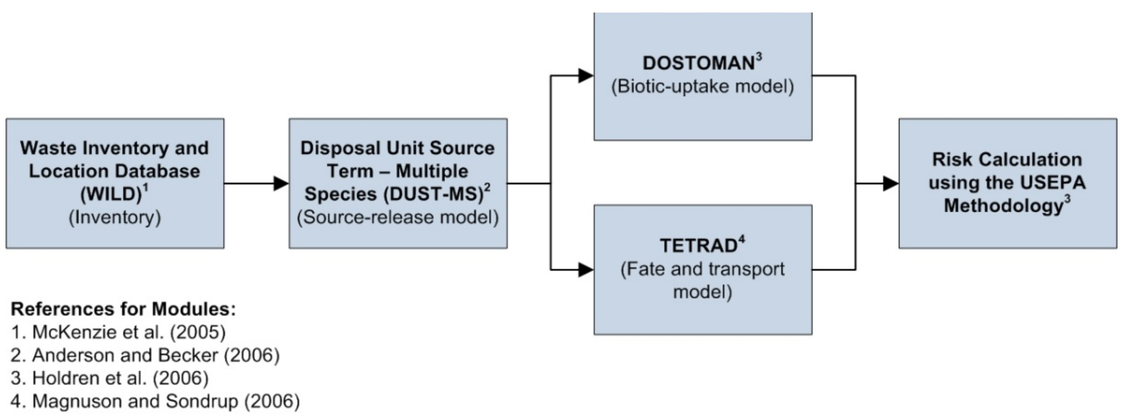

Figure 9. Risk Analysis Modules for the Idaho Subsurface Disposal Area (adapted from Holdren et al. 2006)

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

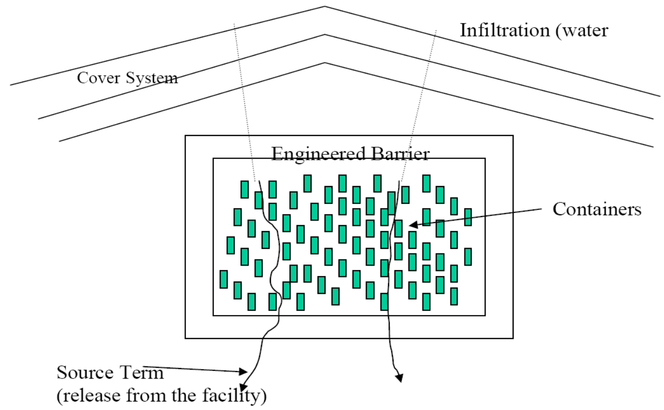

Figure 10. Low-Level Waste Disposal Facility Layout (reproduced from Sullivan 2006)

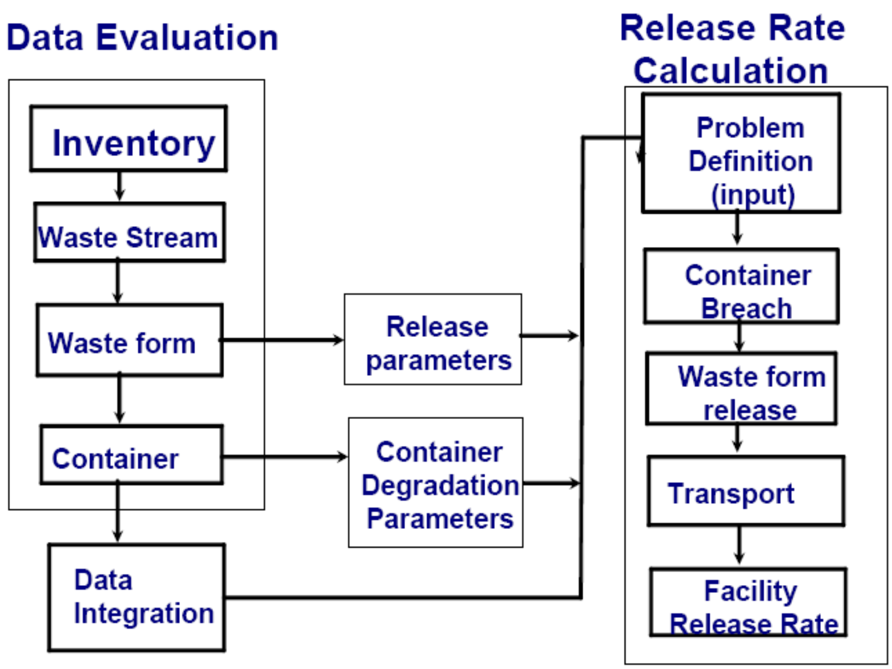

Figure 11. Procedure for Estimating Release Rates for a LLW Disposal Facility Using DUST-MS® (reproduced from Sullivan 2006)

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.

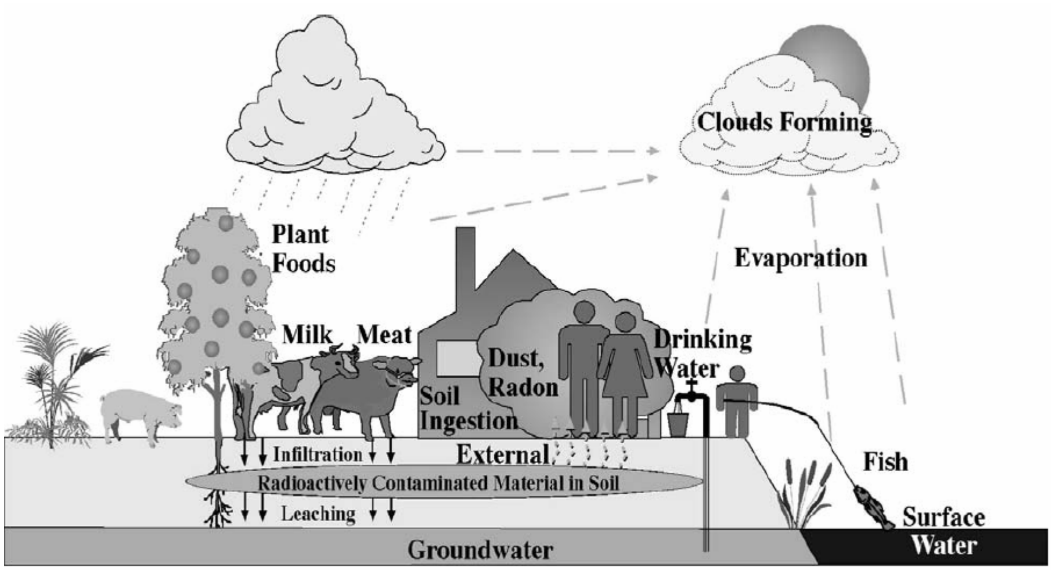

Figure 12. Exposure Pathways and Processes Considered in RESRAD® (reproduced from Yu et al. 2001)

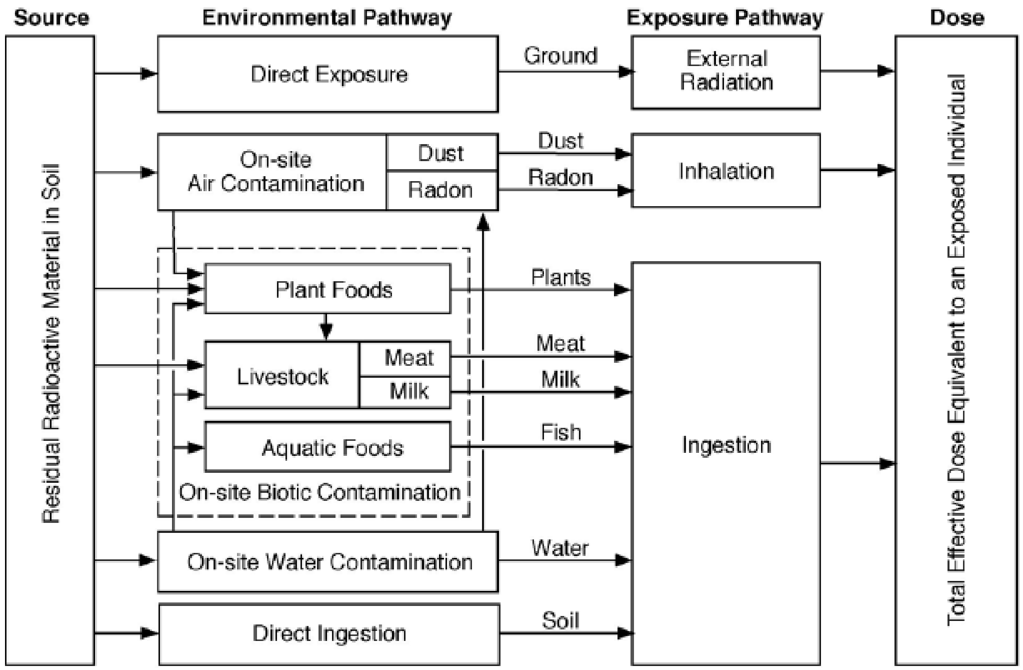

Figure 13. Schematic Representation of the RESRAD® Exposure Pathways (reproduced from Yu et l. 2001)

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.

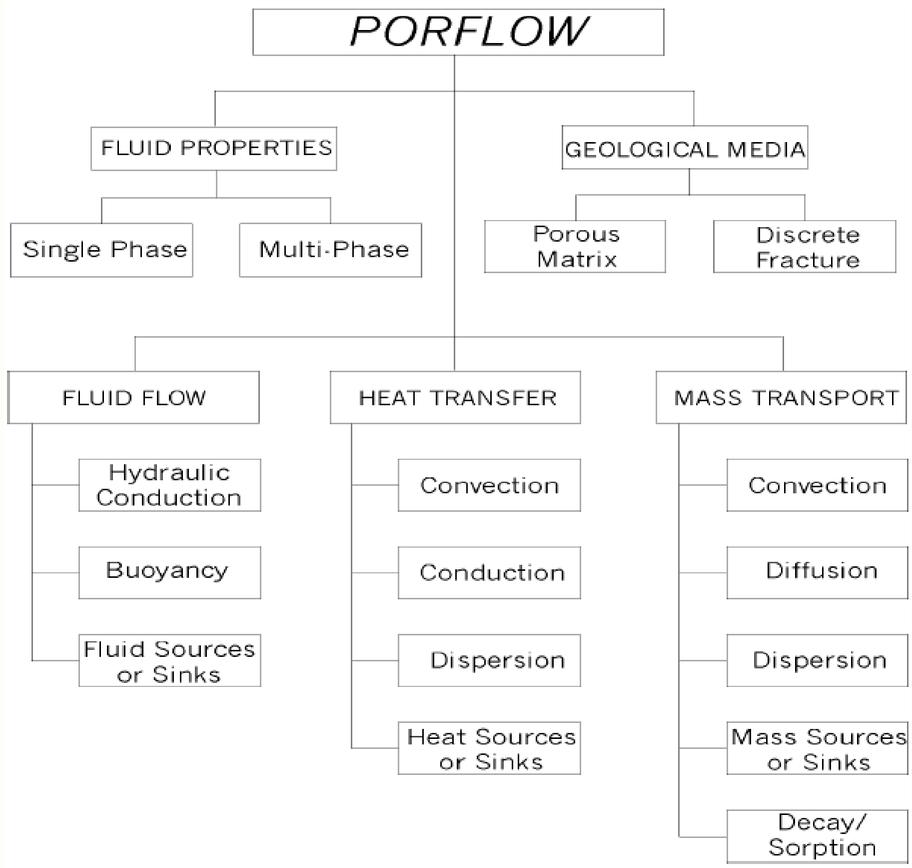

Figure 14. Properties Considered in PORFLOW® (SRS 1997a)

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).

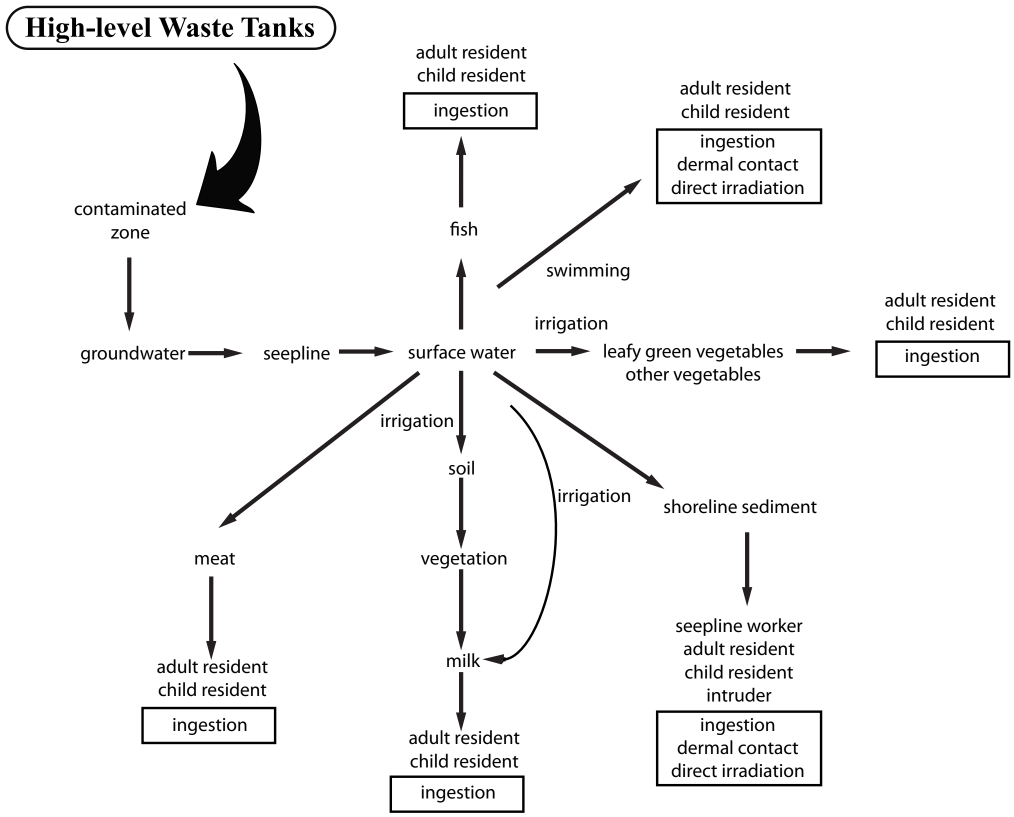

Figure 15. Potential Exposure Pathways for Human Receptors for SRS Tanks 17-F and 20-F (SRS 1997a)

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.

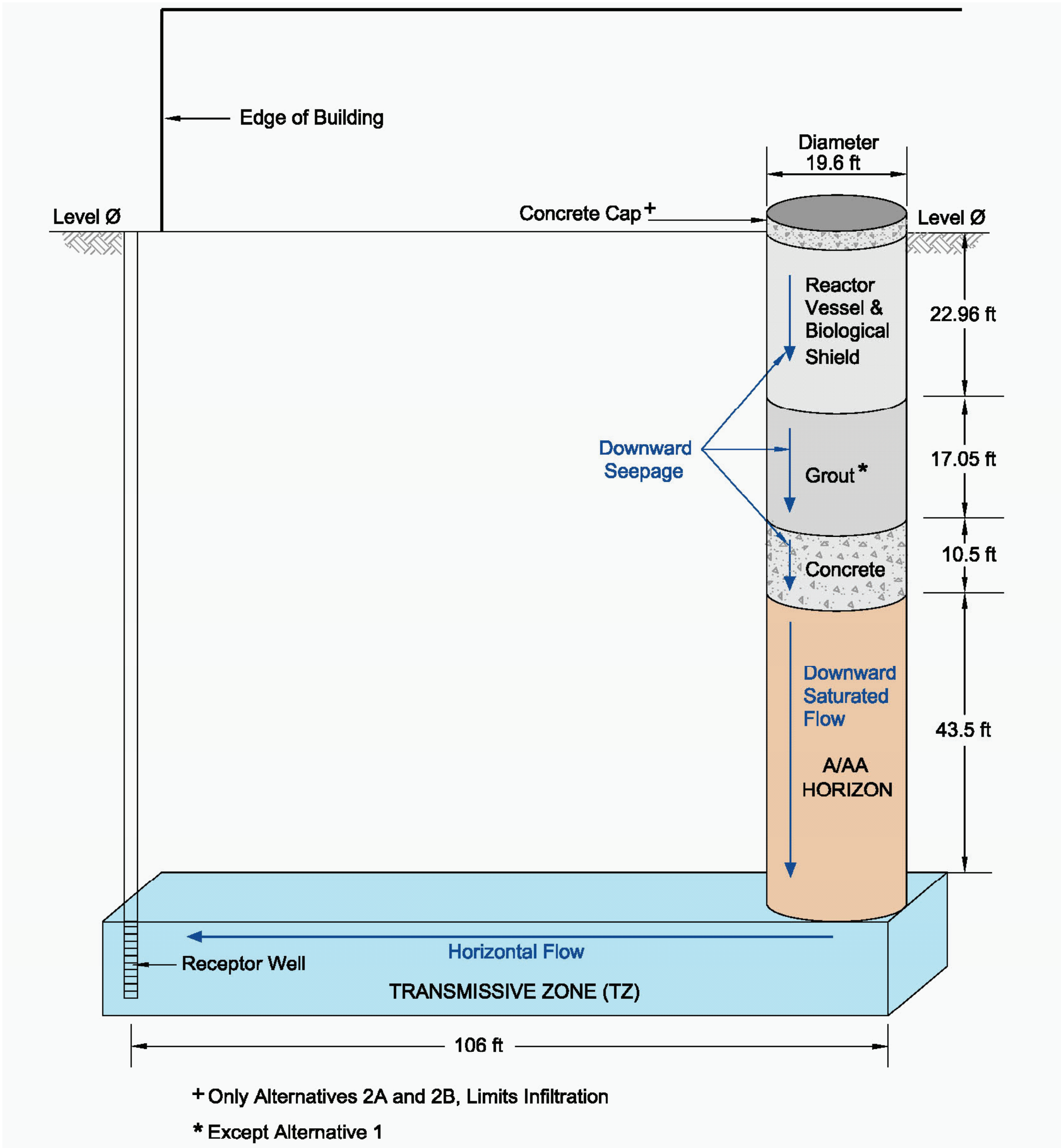

Figure 16. Conceptual Model for P-Reactor Vessel (Council 2008)

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	$3.5x10^{-8}$ cm/s	Log-normal	10
Grout			
Porosity	0.266	Normal	0.02
Initial Hydraulic Conductivity	$3.6x10^{-8}$ 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).

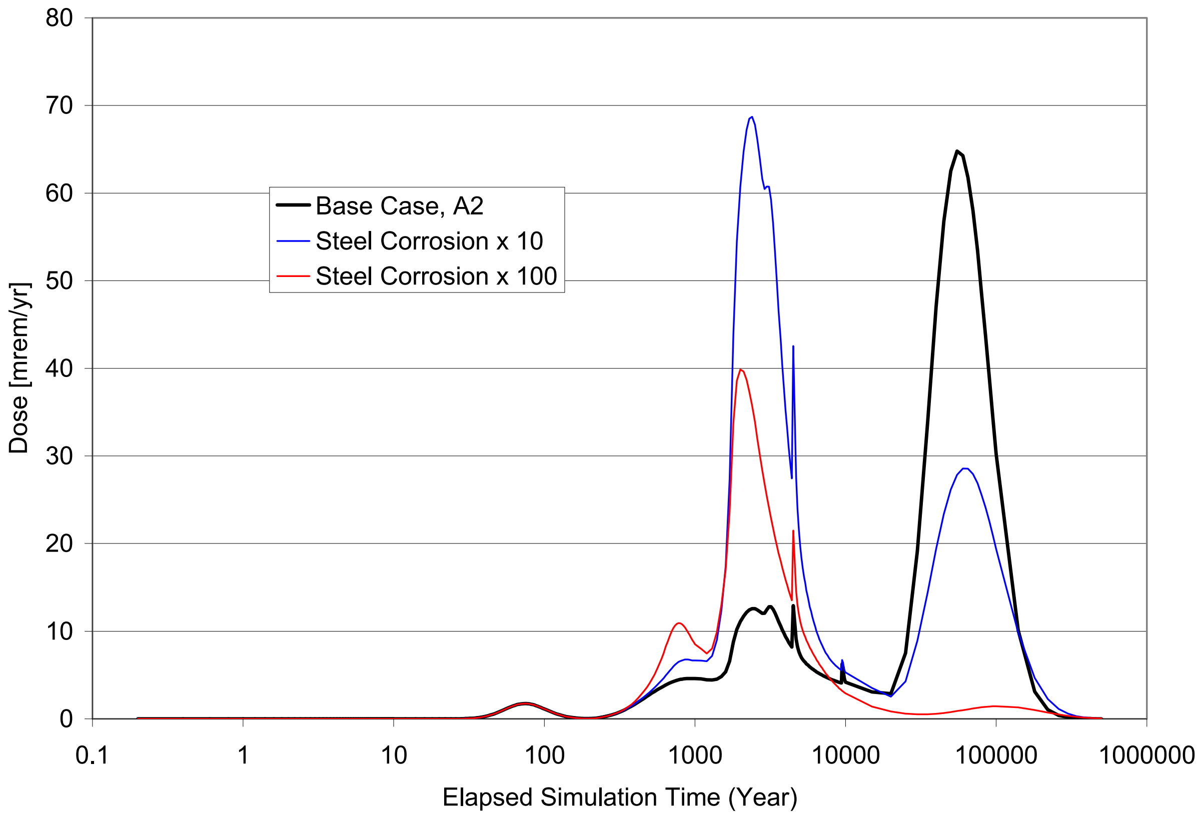

Figure 17. Sensitivity Results for Different Corrosion Rates in P Reactor Assessment (Council 2008)

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.

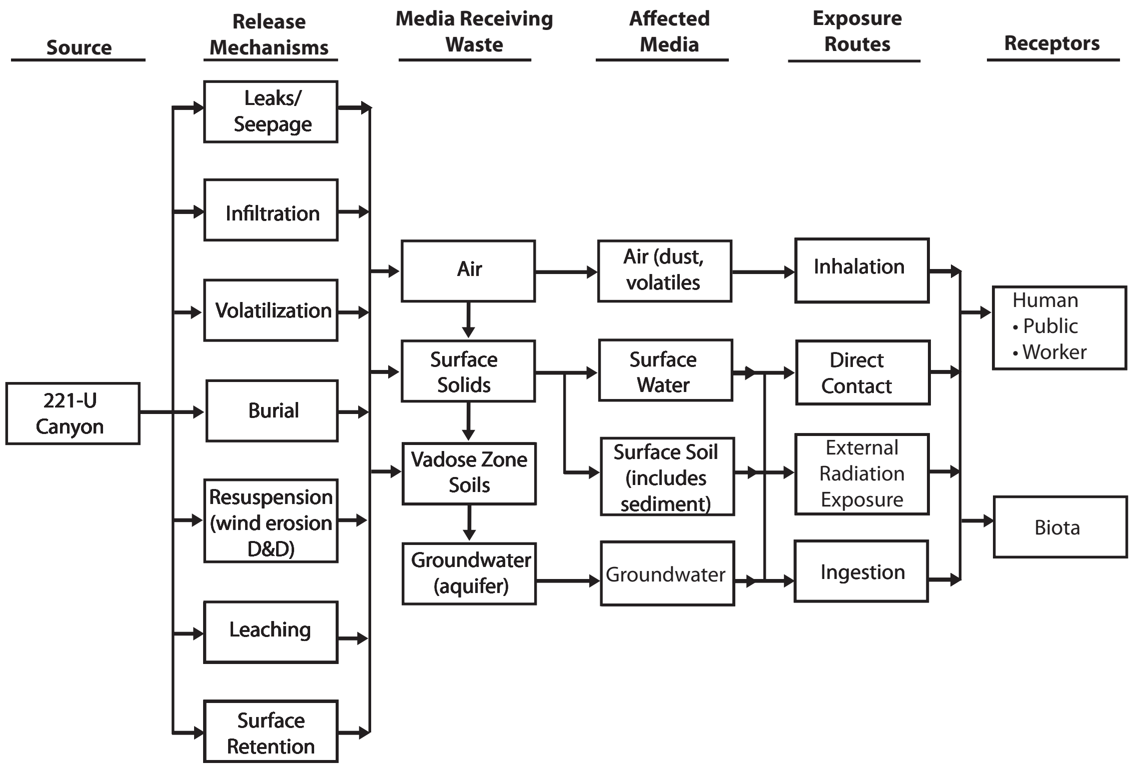

Figure 18. Conceptual Site Model for the Hanford 221-U Facility (reproduced from USDOE-RL 200b)

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.

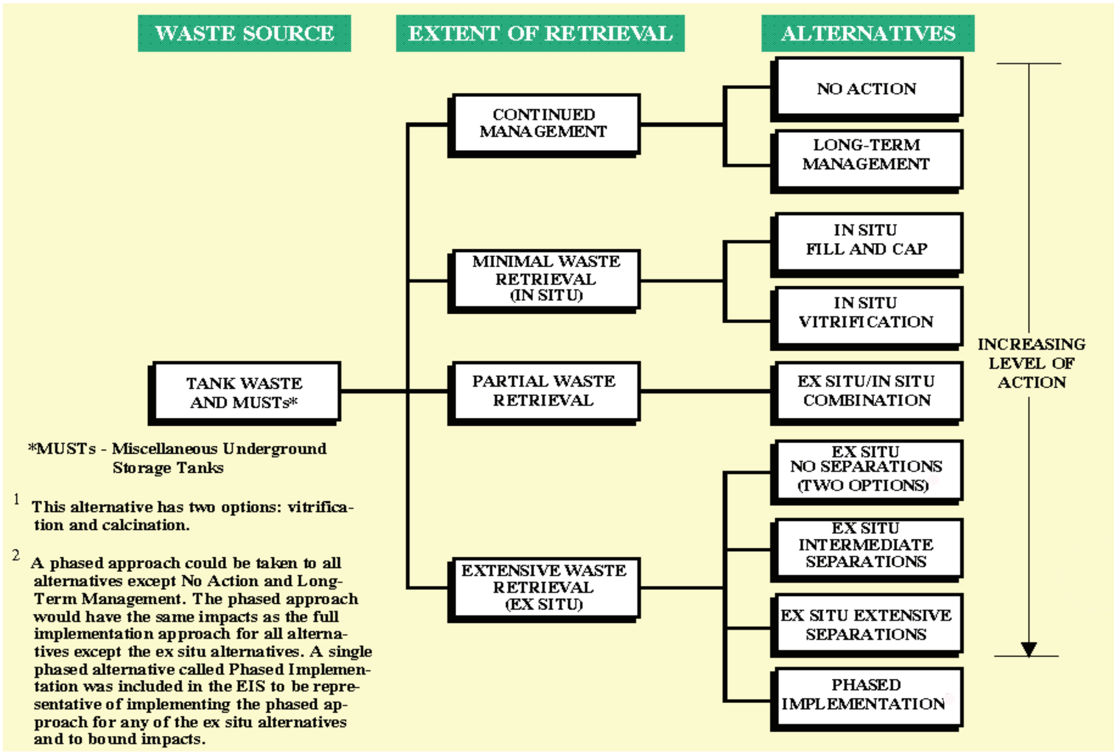

Figure 19. Tank Waste Remedial Alternatives (reproduced from USDOE-RL 1996)

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).

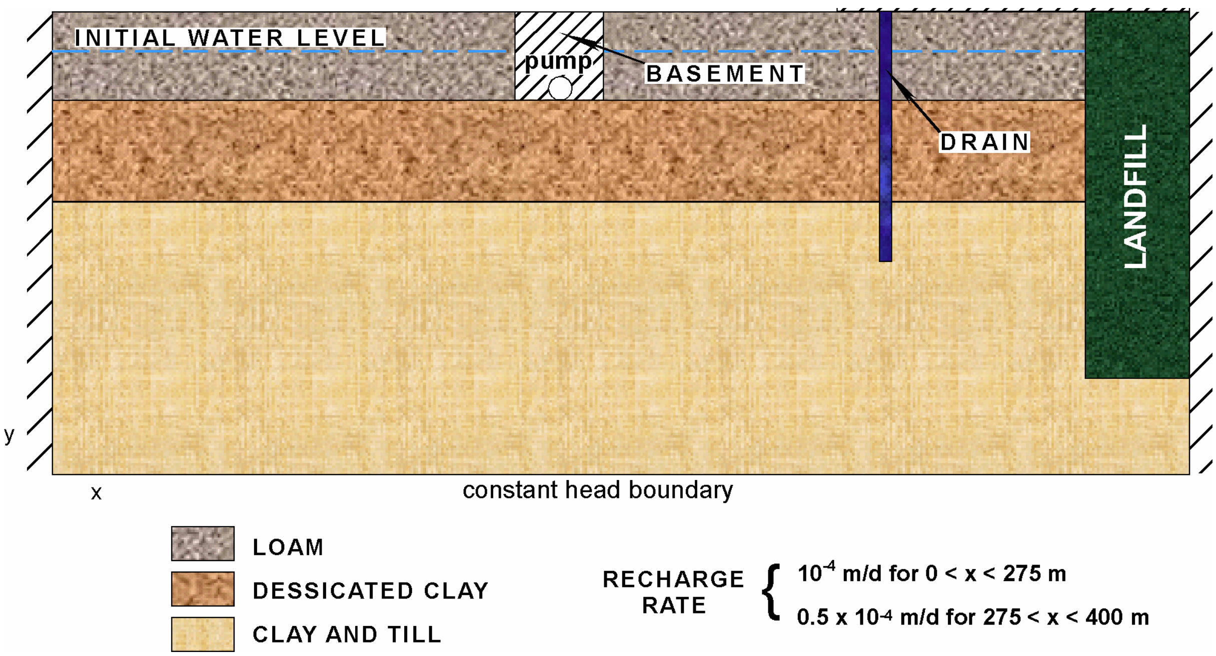

Figure 20. Conceptual Model for a Transient Flow Problem In An Unconfined Groundwater System Adjacent to A Landfill (adapted from Huyakorn et al. 1989)

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

[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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