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Sandia National Laboratories Performance Assessment Methodology for Long-Term Environmental Programs: The History of Nuclear Waste Management

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ABSTRACT

Sandia National Laboratories (SNL) is the world leader in the development of the detailed science underpinning the application of a probabilistic risk assessment methodology, referred to in this report as performance assessment (PA), for (1) understanding and forecasting the long-term behavior of a radioactive waste disposal system, (2) estimating the ability of the disposal system and its various components to isolate the waste, (3) developing regulations, (4) implementing programs to estimate the safety that the system can afford to individuals and to the environment, and (5) demonstrating compliance with the attendant regulatory requirements.

This report documents the evolution of the SNL PA methodology from inception in the mid-1970s, summarizing major SNL PA applications including: the Subseabed Disposal Project PAs for high-level radioactive waste; the Waste Isolation Pilot Plant PAs for disposal of defense transuranic waste; the Yucca Mountain Project total system PAs for deep geologic disposal of spent nuclear fuel and high-level radioactive waste; PAs for the Greater Confinement Borehole Disposal boreholes at the Nevada National Security Site; and PA evaluations for disposal of high-level wastes and Department of Energy spent nuclear fuels stored at Idaho National

Laboratory. In addition, the report summarizes smaller PA programs for long-term cover systems implemented for the Monticello, Utah, mill-tailings repository; a PA for the SNL Mixed Waste Landfill in support of environmental restoration; PA support for radioactive waste management efforts in Egypt, Iraq, and Taiwan; and, most recently, PAs for analysis of alternative high-level radioactive waste disposal strategies including repositories deep borehole disposal and geologic repositories in shale and granite. Finally, this report summarizes the extension of the PA methodology for radioactive waste disposal toward development of an enhanced PA system for carbon sequestration and storage systems.

These efforts have produced a generic PA methodology for the evaluation of waste management systems that has gained wide acceptance within the international community. This report documents how this methodology has been used as an effective management tool to evaluate different disposal designs and sites; inform development of regulatory requirements; identify, prioritize, and guide research aimed at reducing uncertainties for objective estimations of risk; and support safety assessments.

ACKNOWLEDGEMENTS

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Sandia National Laboratories Development of the Performance Assessment Methodology for Long-Term Environmental Programs: The History of Nuclear Waste Management

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ACRONYMS

AEC	Atomic Energy Commission
AMW	Advanced Mixed Waste
AMWTP	Advanced Mixed Waste Treatment Project
CCA	Compliance Certification Application
CCDF	complementary cumulative distribution function
CRA	Compliance Recertification Application
CRWMS M&O	Civilian Radioactive Waste Management System Management & Operating contractor
DOE	U.S. Department of Energy
EIS	environmental impact statement
EPA	U.S. Environmental Protection Agency
EPRI	Electric Power Research Institute
FEPs	features, events, and processes
GCD	Greater Confinement Disposal
GTCC	greater-than-Class-C [low-level radioactive waste]
HLW	high-level radioactive waste
INL	Idaho National Laboratory
IPSC	Integrated Performance and Safety Codes
LHS	Latin hypercube sampling
LLW	low-level radioactive waste
NRC	U.S. Nuclear Regulatory Commission
PA	performance assessment
PDF	probability density function
SDP	Subseabed Disposal Project
SNF	spent nuclear fuel
SNL	Sandia National Laboratories
SPM	Systems Prioritization Method
TSPA	total system performance assessment
TRU	transuranic [radioactive waste]
WIPP	Waste Isolation Pilot Plant
YMP	Yucca Mountain Project

1. INTRODUCTION

Sandia National Laboratories (SNL) is the world leader in the development of the detailed science underpinning the application of a probabilistic risk assessment methodology, referred to in this report as performance assessment (PA), for (1) understanding and forecasting the long-term behavior of a radioactive waste disposal system,[1] (2) estimating the ability of the disposal system and its various components to isolate the waste, (3) developing regulations, (4) implementing programs to estimate the safety that the system can afford to individuals and to the environment, and (5) demonstrating compliance with the attendant regulatory requirements. The SNL PA methodology and its associated tools have been applied in the management and completion of several major programs, which led to the opening of the Waste Isolation Pilot Plant (WIPP), initiation of licensing proceedings for the Yucca Mountain repository, and the disposal of certain specialized nuclear wastes at the Nevada National Security Site.[2] SNL also applied and extended their PA methodology in assisting the U.S. Nuclear Regulatory Commission (NRC) and U.S. Environmental Protection Agency (EPA) in the design and promulgation of the federal regulations that establish environmental standards for (1) safe management, storage, and disposal of spent nuclear fuel (SNF), high-level radioactive waste (HLW), and transuranic (TRU) radioactive wastes in EPA's 40 CFR Part 191 and 40 CFR Part 197, (2) licensing SNF and HLW deep geologic repositories in the NRC's 10 CFR Part 60 and 10 CFR Part 63, and (3) low-level radioactive waste (LLW) management in the NRC's 10 CFR Part 61.

This report summarizes the development and application of the SNL PA methodology from both the regulatory studies and the completion of numerous waste management programs.

1.1 Overview

The foundations leading to modern probabilistic risk assessment methods go back hundreds of years (e.g., see Rechard (1999a)), developing from probability theory and, eventually, reliability analysis of systems including aerospace and weapons systems, and estimation of nuclear reactor hazards. The *Reactor Safety Study* (also known as "WASH-1400" or the Rasmussen report), initiated in 1972 and published in 1975, was the first detailed, comprehensive, quantitative, and probabilistic look at the health risks from a large, complex facility. The report, prepared by a 60-member team that included participants from Sandia National Laboratories, defined hazards, estimated associated probabilities, and evaluated consequences on the Surrey and Peach Bottom plants. Participation in this study was one of the important events at SNL between 1973 and 1975 that became the foundation for the application of probabilistic risk assessment to radioactive waste management projects. With a history now spanning nearly 40 years, SNL has developed and applied the PA methodology, informing key decisions concerning radioactive waste management both in the United States and internationally.

[1] A waste disposal system as referred to in this report is the combination of natural barriers (i.e., geologic formations) and engineered barriers (i.e., man-made barriers, such as waste containers) working individually and jointly to prevent or delay waste from reaching the environment accessible to humans.
[2] Formerly known as the Nevada Test Site.

Though a PA methodology can be and has been applied to analyze other kinds of risks (e.g., risk associated with nuclear reactor operations or weapons systems), this report focuses on the methodology SNL developed for probabilistic risk analysis of radioactive waste (or mixed waste) disposal methods, facilities, and systems. As mentioned above and as will be shown in the other sections of this report, the SNL PA methodology has been used to gain an understanding of the key processes and phenomena influencing the long-term performance of a disposal system, as a management tool to identify and prioritize research needs, and finally to demonstrate that a disposal system meets or exceeds the performance objectives established by the attendant regulations for the long-term protection of human health and the environment. As described in Section 8.7, this methodology for radioactive waste management has recently been extended to analysis of long-term disposal systems for another important waste material that society is beginning to address: carbon dioxide.

1.1.1 Beginnings of Waste Disposal Performance Assessment at Sandia National Laboratories, 1973–1975

In April of 1973, an ad hoc committee at Sandia Laboratories (Sandia)[3] explored ways in which Sandia could contribute solutions to the problems associated with management of radioactive wastes (Winter, et al. 1973). The committee had two stated goals:

> 1. *To identify segments of the waste management sequence where Sandia's general research and engineering skills could be useful to the AEC Division of Waste Management and Transportation.*
>
> 2. *To seek out long term major problems, not currently being worked by other laboratories, where experience unique to Sandia could be profitably brought to bear.*

At the time, the Atomic Energy Commission (AEC)—the predecessor agency of the U.S. Department of Energy (DOE) and the NRC—was focused on salt beds as potential repository sites, but their selected pilot site in an abandoned salt mine in Lyons, Kansas, had recently been abandoned as unsuitable. The committee's report (Winter, et al. 1973), identified igneous rock "as a promising alternative toward which Sandia could contribute," and, among other disposal ideas, proposed very deep boreholes and mined repositories for Sandia study. It was recognized that a comprehensive analytical and experimental program would be needed to reduce the uncertainties associated with deep geologic disposal. The committee recommended further investigation of ongoing or proposed activities at other laboratories, continued investigation of disposal in other geologic media not currently under investigation, and laboratory evaluation of material properties and preliminary testing of simple models. In addition to its strengths in the fields of geophysics, thermodynamics, health physics and engineering, and its computational and laboratory facilities, the committee identified Sandia's capabilities in systems analysis for optimization of solutions, quality assurance, testing, and management of field operations as key to organizing its technical skills into a large focused project, which would be required to develop an ultimate disposal solution for radioactive waste.

[3] Until 1979, SNL was known as "Sandia Laboratories."

That same year, contacts between Sandia and Woods Hole Oceanographic Institution led to discussions of the feasibility of disposal of radioactive wastes in the thick sediments at the floor of deep ocean basins (previously, disposal in relatively near-shore sediment deposits of submarine deltas, and fans had been a focus of study elsewhere). As a result of preliminary discussions, a workshop, made up of Sandia staff and oceanographers from Woods Hole, Scripps, and Worcester Polytechnic Institute was held in Albuquerque in June 1973 to consider the deep ocean floors and radioactive materials. It was concluded that the deep ocean basins were the best ocean regions to consider for nuclear waste repositories (Bishop and Hollister 1973), and within a few years the Subseabed Disposal Project (SDP) at Sandia rapidly developed into an international program, and new approaches to performance assessment for disposal of radioactive wastes were being developed to optimize repository systems and forecast repository safety.

In 1974, Sandia participated in a team led by Professor Norman Rasmussen from the Massachusetts Institute of Technology that evaluated the potential health risks associated with accidents from a commercial nuclear power plant. This work led to the publication of the *Reactor Safety Study* (WASH-1400) in 1975 (NRC 1975), which set the stage for the probabilistic risk methodology used to evaluate nuclear power plants. By the late 1970s and early 1980s, SNL was advocating a probabilistic approach for the modeling of geologic waste repositories, which was influenced by the earlier involvement on WASH-1400, as well as ongoing consequence analyses that SNL developed for the NRC, specifically CRAC-II (Aldrich, et al. 1982), and NUREG-1150 (NRC 1990). Concurrent with work on NUREG-1150, SNL was also applying the probabilistic risk assessment methodology to the evaluation of the reliability of nuclear weapons systems (Carlson, et al. 1991). Although the early NRC work has now been replaced by state-of-the-art reactor consequence analyses, that early work played an important role in the first stages of development of the probabilistic risk assessment methodology SNL developed for deep geologic repositories.

Sandia was named by the AEC in January 1975 as the lead laboratory for further site characterization of a proposed repository site in bedded salt in southeastern New Mexico and for development of a conceptual repository design and an environmental impact statement (EIS). By the end of 1975, based on potentially disqualifying evidence found by exploratory drilling on the first proposed site, Sandia recommended relocating the potential repository site 11 km southeast of the first location to a new location, 42 km east of Carlsbad NM and nearer the center of the Delaware Basin, where more predictable geology was to be found (Rechard 1999b). This is the location of WIPP, which opened in 1997 as the world’s first deep geologic repository designed and constructed for disposal of TRU waste.

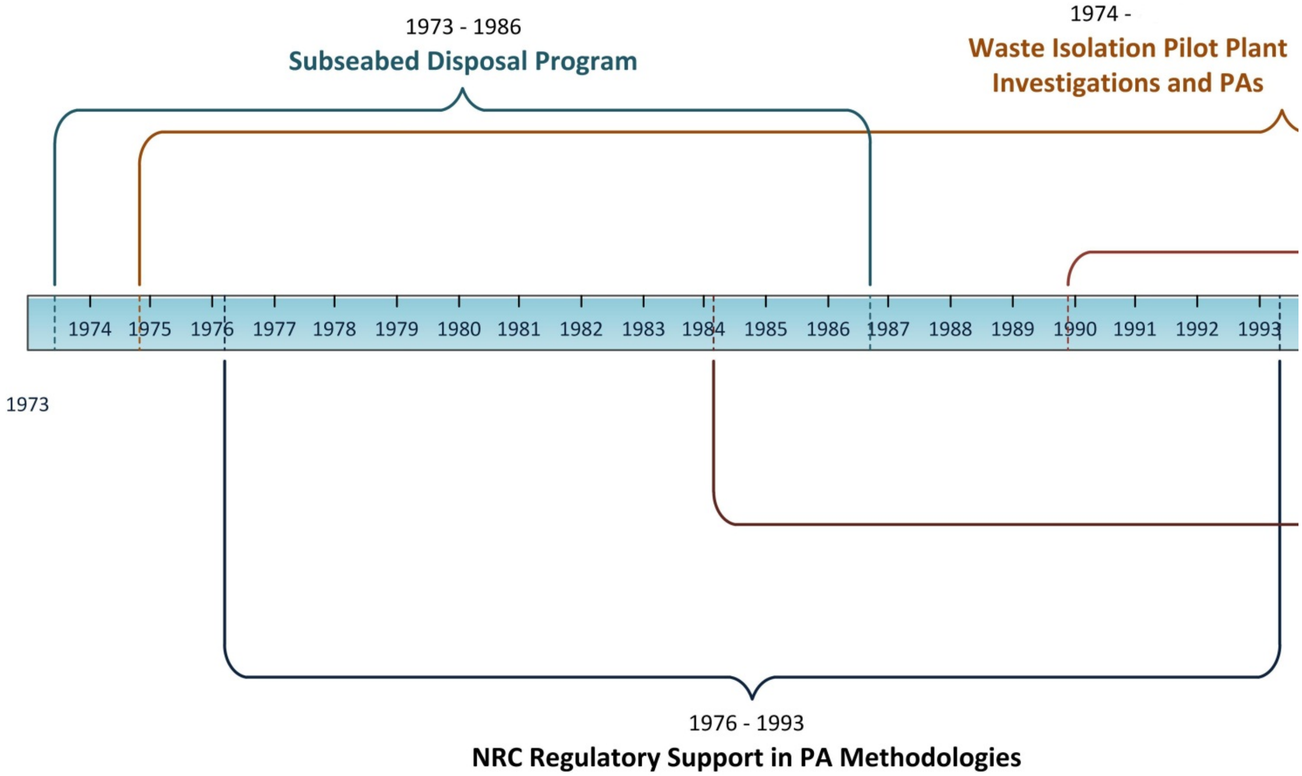
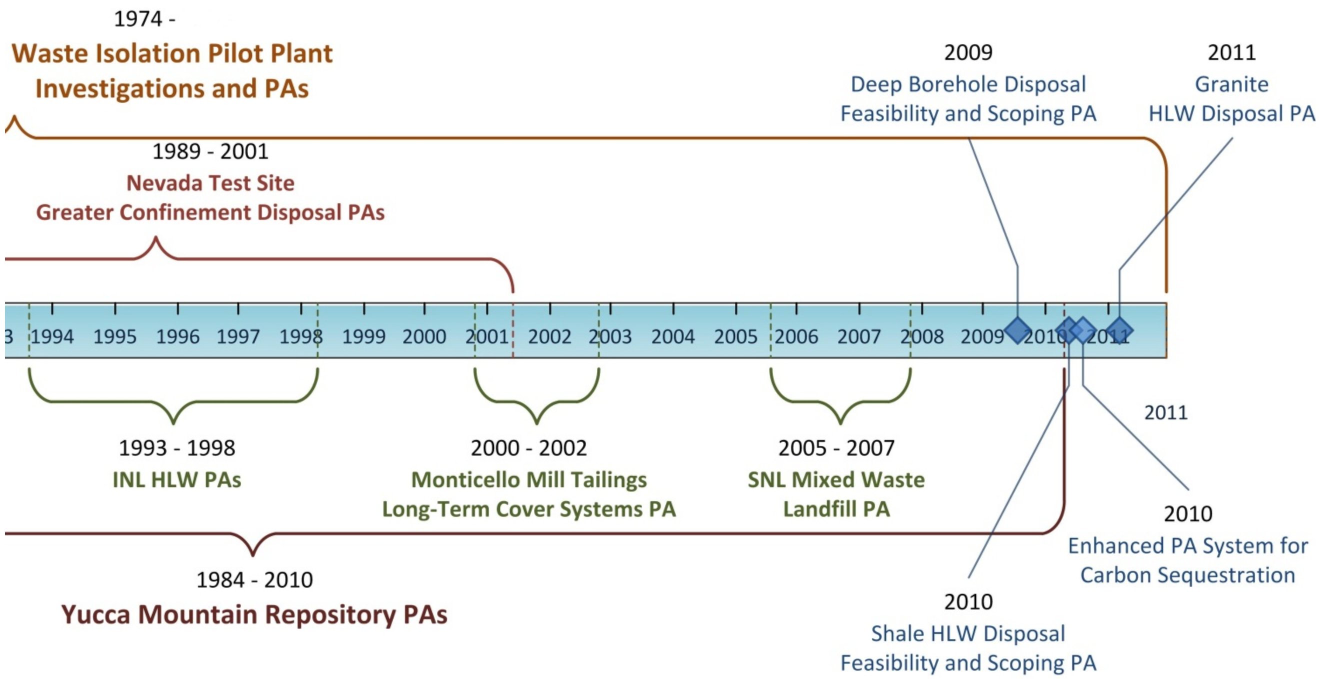


Figure 1. Timeline of Sandia National Laboratories performance assessment

1.1.2 Historical Outline of SNL Performance Assessment Analyses for Radioactive Waste Management

The SNL history of developing and applying PA methodologies to radioactive waste management problems is illustrated in the four-decade-long timeline in Figure 1. This report documents the evolution of the SNL PA methodology over that timeframe, touching on the following SNL PA applications that are shown in Figure 1:

- Subseabed Disposal Project (SDP) (Section 2)
- NRC PA methodology development and demonstration for deep geologic disposal of SNF and HLW (Section 3)
- NRC PA methodology development and demonstration for near-surface LLW disposal (Section 3)
- Technical support to the NRC and EPA in the development of radioactive waste disposal health standards and regulations (Section 3)
- Waste Isolation Pilot Plant (WIPP) PAs for disposal of defense TRU waste (Section 4)
- Yucca Mountain Project (YMP) total system PAs (TSPAs) for disposal of HLW and SNF (Section 5)
- Implementation of the PA methodology to Greater Confinement Borehole Disposal (GCD) boreholes at the Nevada National Security Site (Section 6)
- Evaluation of two generic geologic repositories for disposal of HLW and SNF stored at Idaho National Laboratory (INL) (Section 7)



- Monticello Mill Tailings PA methodology (Section 8.1)
- Application of the PA methodology to the SNL Mixed Waste Landfill in support of DOE's Environmental Restoration Program (Section 8.2)
- Deep Borehole Disposal feasibility and scoping PA (Section 8.3)
- Shale HLW repository feasibility and scoping PA (Section 8.4)
- Granite HLW repository feasibility and scoping PA (Section 8.5)
- Support for international radioactive waste management efforts in Egypt, Iraq, and Taiwan (Section 8.6)
- Development of an enhanced PA system for carbon sequestration and storage systems (Section 8.7).

These efforts have produced a generic PA methodology for evaluation of waste management systems that has gained wide acceptance within the international community. More importantly, this methodology has been used as an effective management tool to evaluate different disposal designs and sites; inform development of regulatory requirements; identify, prioritize, and guide research aimed at reducing uncertainties for objective estimations of risk; and support safety assessments.

Though differences appear in the actual detailed applications and program approaches (e.g., in various performance measures selected independently or defined by applicable regulations, in the extent to which model abstraction is applied, in details of the computational implementation and software tools applied), at a fundamental level the PA methodology is identical in each of the above applications. The PA methodology consists of characterizing the overall system (i.e., waste, facility, and site), scenario selection and screening, building the system model (i.e., conceptual, mathematical, and computational or numerical models), consequence modeling (e.g., source term estimation, groundwater flow, radionuclide transport, biosphere transport, and health effects), and evaluating system and subsystem performance, including uncertainty and sensitivity analysis. Fundamental to the SNL's PA methodology is that it is iterative, allowing new information and data to be incorporated as they become available and enabling efforts to be focused on what is most important to system performance measures of interest.

1.2 PA Methodology

1.2.1 Overview

Kaplan and Garrick (1981) famously described risk as a set of triplets, where risk analysis consists of developing answers to the following three questions:

1. What can happen? What can go wrong?
2. How likely is such an outcome to happen?
3. If it does happen, what are the consequences?

The first question, "What can happen?" is answered in the form of scenarios, combinations of events or processes that could occur. The second question, "How likely is it to happen?" is answered from available evidence on the frequency of such events, where data exists, or, when there is little or no data available, from predictive analyses of probability and uncertainty (in contrast to probabilistic risk assessment, deterministic risk assessments that select only specific events for consequence analysis provide implied, unquantified answers to this second question). The third question, "What are the consequences?" is answered for each scenario to assess the range of possible outcomes.

This concept of risk analysis has been used on the SNL PAs since their first introduction, and was implicit even in SNL's earliest PAs, predating Kaplan and Garrick's formal statement of the concept. It was the starting point for the description of the PA mathematical framework that would impact the treatment of uncertainty, calculation design, and information displays for diagnosis, understanding, and communication of results.

Because of the large temporal and spatial scales required to analyze radioactive waste disposal systems (i.e., tens of kilometers and thousands to hundreds of thousands of years), uncertainty permeates all aspects of PA applications. For that reason, the SNL PAs explicitly consider a fourth question: "What is the uncertainty in the answers to the first three questions?" or "What is the level of confidence in the answers to the first three questions?" Uncertainties are propagated through the analysis. To a large extent, the credibility of the analysis and its results hinge on the manner in which uncertainties are identified and quantified. Approaches and considerations for quantifying uncertainty are described in more detail in Section 1.2.2.

The PA methodology provides the framework for assembling, organizing, and assessing the large quantity of data and information needed to evaluate the performance of complex systems, such as radioactive waste disposal systems. The PA methodology incorporates data and information from multiple sources and organizes them in a logical manner to support decision-making, explicitly taking into consideration the different sources of uncertainty that will influence the analysis. It also provides a framework that enhances the traceability, transparency, reproducibility, and retrievability of the technical work. Finally, it allows for the analysis of how the different components (i.e., subsystems) of the disposal system behave in isolation and in conjunction with each other.

The PA methodology is applied to analyze the behavior of specific subsystems or of the total system. This type of analysis allows estimation of the ability of specific subsystems, in isolation or in conjunction with other subsystems to prevent or delay the release of radioactive contaminants to the environment accessible to humans. The waste isolation capability of subsystems and of the total system is determined by estimating the numerical value of specific performance measures.

Radioactive waste disposal projects typically evolve over several decades (OECD/NEA 2007) moving through several phases during that timeframe, from early site selection, site characterization, preliminary analyses and, eventually, the safety assessments employed to inform a final licensing decision for a disposal facility. The SNL PA methodology has served as an effective management tool for such long-term, evolutionary projects because the same methodology is used from initial the research and development to the final safety assessment. The results of early applications of the methodology are used to systematically validate the system design, focus the associated research and testing program in the most important areas, identify opportunities to reduce costs, and ensure the design incorporates best practices.

SNL has demonstrated in numerous projects that applying the PA methodology in an iterative manner ensures that research and development activities are closely linked to the behavior and performance of subsystems and of the total system. The results of the analyses typically improve with successive iterative applications of the methodology as more data and information become available and understanding of the system improves. New information can, and has been, used to refine performance measures, alternatives, and models, thus reducing important sources of uncertainty following each iteration. As a project progresses and the subsystems and the total system are better understood, the application of the methodology informs additional data needs.

In the very early phase of a radioactive waste disposal project, applications of the PA methodology tend to be exploratory in nature and rely on relatively simple models focused on the identification of opportunities for improving understanding of the system under consideration. As the project evolves, more detailed models are incorporated into the methodology. In the intermediate phases of the project, applications of the methodology provide opportunities to review alternative models, conduct uncertainty and sensitivity analyses, identify shortcomings in the analysis or model implementation, and communicate with stakeholders. Eventually, once the understanding of the disposal system is sufficiently mature to proceed to the licensing phase, the application of the PA methodology provides the foundation for the safety analysis that informs the licensing decision.

Over time, the SNL PA methodology has evolved based on the knowledge gained from numerous projects to allow:

- Evaluation of subsystem and total system performance with respect to specific measures (up to and including compliance with applicable regulatory requirements for the system of interest),
- Quantification of performance margin by comparing the numerical results from the analysis to the established limits for the attendant performance measures,
- Evaluation of design options/alternatives,
- Development and streamlining of the underlying models used to simulate the different phenomena and process affecting the performance of the disposal system,
- Determination of significant sources of uncertainty,
- Prioritization of research and testing needs, and
- Prioritization of risks with respect to the decision of interest.

Because of these capabilities, both the EPA and the NRC require that PA is used to estimate the behavior and performance of radioactive waste disposal system when demonstrating compliance with their respective standards and regulations; e.g., 40 CFR Part 191 and 40 CFR Part 197 for the EPA and 10 CFR Part 60, 10 CFR Part 61, and 10 CFR Part 63 for the NRC.

1.2.2 Steps in the PA Methodology

While the applicable performance measures of interest vary among individual projects and/or disposal systems, the overall approach in the SNL PA methodology is fundamentally the same for all projects and generally comprises the following steps (see Figure 2), which are summarized in this section:

1. Define performance goals;
2. Characterize system (waste, facility and site);
3. Identify scenarios for analysis;
    a. Identify and screen relevant FEPs
    b. Construct and screen scenarios
    c. Estimate scenario probabilities
4. Build models and abstractions of relevant FEPs;
    a. Conceptual models
    b. Mathematical models
    c. Computational models
5. Quantify uncertainty;
6. Construct integrated PA model and perform calculations
7. Uncertainty and sensitivity analyses;
8. Evaluate performance; and
9. As needed, direct science and testing program.

Figure 2 shows the iterative nature of the application of the SNL PA methodology. First, the evaluation of performance can be used to determine whether the system complies with the desired performance goals. Results from early applications of the methodology can be used to evaluate the practicality of preliminary performance goals. The early work performed by SNL for both the EPA and the NRC is an example of the use of the PA methodology in this context. Second, a directed science and testing program can be used to prioritize research needs. In both instances, the application of the PA methodology is repeated. This iterative approach continues until it is determined that sufficient confidence exists in the results of the analysis to inform a decision regarding the project; e.g., move the project to next phase up to and including licensing and operating decisions.

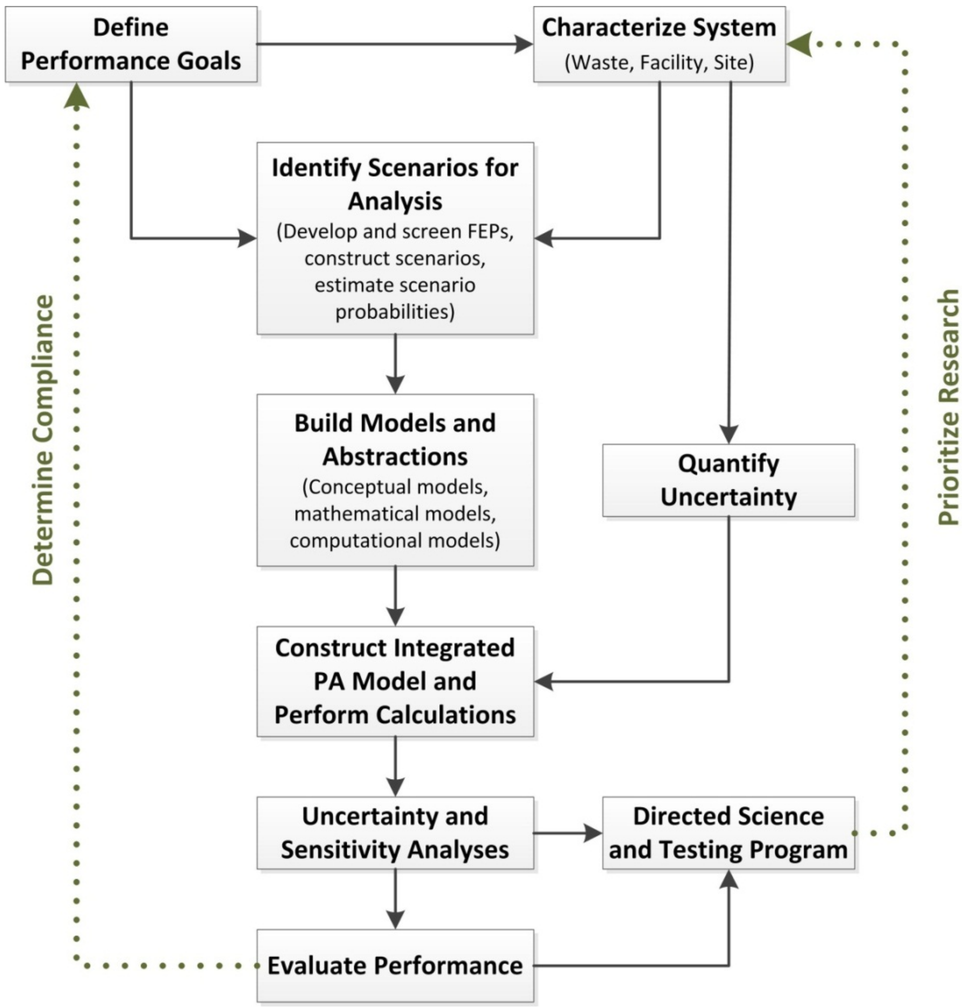


Figure 2. SNL performance assessment methodology

Define performance goals. Performance goals are typically defined up front because they determine the design of the PA analysis and have considerable influence on scenario construction, model development, and research programs. In the case of the WIPP and Yucca Mountain projects, the performance goals were standards and requirements specified by EPA regulations and also, for Yucca Mountain, NRC regulations. Even though both WIPP and YMP were deep geologic disposal facilities, their respective postclosure performance was governed by different measures and different timeframes. It is also instructive to note that, in both cases, the preliminary PA iterations were conducted based on assumptions regarding regulatory performance measures, because of changes in the law or legal challenges forcing remand of the rules. Similarly, no radiological protection criteria existed for subseabed HLW disposal, so the SDP conducted its PAs based on hypothetical standards and focused considerable effort into developing recommended interim radiological standards (Kaplan, Klett, et al. 1985, Klett 1997a).

The primary measure for postclosure performance at WIPP was cumulative release of radionuclides to the environment accessible to humans over 10,000 years. Therefore, exposure pathways were not analyzed. The application of the PA methodology at WIPP put large emphasis on human intrusion, requiring models of drilling intrusion. For the YMP, the primary postclosure performance measure was annual dose to maximally exposed individual over both 10,000 years and 1 million years. The application of the PA methodology at YMP emphasized evaluation of the ability of engineered and natural barriers to contain radionuclides as well as dose calculations. The performance measures adopted by the SDP, lacking any regulatory standards, assumed limits for annual individual dose rates and world population dose rates, while also assessing other potential impacts, including biota doses.

These projects serve to illustrate the importance that the performance goals play in determining the details of how the analysis is conducted, even though the fundamental steps in the PA methodology are the same.

Characterize system. Once the performance goals and other requirements have been identified, a complete description of the system to be evaluated is needed. A system description includes the characteristics of the waste (e.g., radionuclide inventory, decay chains, half-lives, etc.), the facility (e.g., size, thermal loading from emplaced waste, design, properties of engineered barriers, etc.), and the site (e.g., geology, hydrology, geochemistry, etc.). In the early stages of a nuclear waste disposal program, the characterization program is broad-based and focused on gaining an adequate understanding of the system, identifying the greatest sources of uncertainty, and identifying the FEPs most likely to affect the system's long-term performance. As knowledge and understanding of the system improve, resources are allocated to characterization activities that most likely will result in reasonable reductions to important sources of uncertainty.

Identify relevant features, events, and processes and scenarios for analysis. The processes and events, or sequences of processes and events, that may be relevant over the time frame of interest need to be identified and included in the PA. Relevant (i.e., retained or included) FEPs are used in the construction of the scenarios evaluated in the PA modeling. Steps in the identification of relevant FEPs include (1) identifying the universe of FEPs, (2) classifying the FEPs, (3) screening the FEPs, and (4) thoroughly documenting the results.

The goal of identifying the universe of FEPs potentially relevant to the long-term performance of the system of interest is to be comprehensive (i.e., nothing is too insignificant or improbable to be considered as potentially relevant). This is of considerable importance because a key source of uncertainty, often referred to as "completeness," is addressed by compiling a comprehensive list of potentially important FEPs. To ensure completeness, the FEP list should include both general FEPs from other radioactive waste disposal programs and site-specific FEPs identified during site characterization. A good source of general FEPs is the international Nuclear Energy Agency FEP list (NEA 2006). The Nuclear Energy Agency FEP database contains approximately 1,650 FEPs from 10 radioactive waste disposal programs worldwide and represents the most complete list of FEPs potentially relevant to radioactive waste disposal.

To facilitate screening, FEPs are classified and grouped to the coarsest level at which technically sound screening decisions can be made, while still preserving adequate detail for the PA analysis. FEPs are systematically classified according to relevant subject areas and redundant FEPs are combined, reducing the number of screening arguments to a manageable level.

FEP screening involves examining the universe of potentially relevant FEPs to identify those FEPs that should be included or retained in the PA analysis. FEP screening is typically based on exclusion criteria, most commonly (1) low probability of occurrence, (2) low consequence, or (3) inconsistency with applicable regulatory guidance. Explicit guidance, such as directed in regulations, regarding certain assumptions can result in the exclusion of FEPs that are inconsistent with those assumptions (e.g., the regulatory treatment of the human intrusion scenario).

Probability-based screening arguments are used to exclude those FEPs with very low likelihood of occurrence because they do not represent major contributors to total risk. Threshold probability values, below which FEPs need not be considered, can be defined by the applicable regulations (e.g., events that have at least one chance in 10,000 of occurring during a specified period can be specified by regulation for inclusions in the PA). FEP probabilities may be based on frequency of past events, expert judgment, analogue information, or regulatory guidance.

FEPs can be excluded on the basis of low consequence because they are also expected to be small contributors to total risk. Consequence-based screening arguments may evaluate impact on intermediate performance measures, use deterministic and bounding analyses, use models and codes external to the PA, or rely on varying levels of analysis such as reasoned arguments based on the literature, hand calculations, extensive site characterization or modeling outside of PA, or sensitivity analysis.

Once screening of individual FEPs is completed, scenarios—combinations of FEPs each representing a possible realization of the future state of the system—are developed. All retained FEPs must be accounted for in the PA in at least one scenario. The process for scenario construction is similar to that for FEPs development: (1) formulate scenarios using retained FEPs, (2) screen scenarios, and (3) thoroughly document results.

A typical approach is to include retained FEPs that are expected to occur in a nominal or “expected” scenario and to form one or more “disturbed” scenarios from the FEPs that were retained in screening but not included in the nominal scenarios (the FEPs in these disturbed scenarios are typically called “disruptive FEPs”). Another approach for constructing scenarios is to build them around release pathways. The scenario probabilities are the product of the probabilities of the FEPs included in the scenario. Scenario screening criteria are identical to that used to screen FEPs: probability of occurrence, consequence, or regulatory guidance.

Build models and abstractions of relevant FEPs. The FEPs and scenarios retained after the screening process are represented in the performance assessment through conceptual models, mathematical models, and computational (numerical) models. Conceptual models describe system behavior for the scenarios of interest (e.g., Darcy flow in porous media). Conceptual models consist of the FEPs that are active in each scenario (e.g., groundwater flow), the different laws of physics and chemistry that govern those FEPs (e.g., Darcy’s Law), the dimensionality of the model (one-, two-, or three-dimensional model), and other assumptions (e.g., time-dependent or steady state). Mathematical models quantify conceptual models in terms of mathematical expressions ranging from simple representations (e.g., response surfaces, independent linear relationships) to complex representations (e.g., coupled nonlinear partial differential equations). Computational models provide numerical (or analytical) solutions to the mathematical models. The models that are used vary in complexity, and a hierarchy of models can exist. An overarching conceptual model of each scenario is developed to guide the development of more detailed mechanistic models of individual FEPs that comprise the scenario.

The models consist of sets of hypotheses, assumptions, simplifications, and idealizations that, together, describe the essential aspects of a system or subsystem of the repository relative to performance. An example of such a process model is one that describes the movement of water and dissolved radionuclides by diffusive flow in rock pores or by advective flow in fracture openings in the unsaturated bedrock surrounding the repository. Because the PA methodology deals with future outcomes and includes uncertainty in both descriptions of processes and parameter values, an essential element of the PA methodology is to capture uncertainty in probabilistic analyses that represent likely outcomes, based on the best available values of process model parameters and the processes involved.

Abstractions are progressive simplifications of the detailed models of physical and chemical processes to more compact, efficient numerical models. Abstractions consist of statistical or mathematical abstractions, including lookup tables, equations representing response surfaces, probability distributions, linear transfer functions, or reductions of model dimensionality. Abstractions, where incorporated, should be compact but still capture the salient features of the detailed models, along with their associated uncertainties.

The computational models for each component will be linked into an overall system model. Data transfer between codes can be automated to ensure consistency and traceability throughout a large number of repetitive computer simulations. The integration of the more detailed models may include the models themselves, detailed models with the unimportant subroutines turned off and unimportant data distributions set at their mean or median value, or a simplified abstraction of the model results. For computational efficiency, some level of simplification or abstraction of the more detailed process models is needed for the PA models for Monte Carlo simulation. However, both the detailed process models and the efficient, “streamlined” PA models are important to the PA process.

Quantify uncertainty. The Kaplan and Garrick (1981) risk triplet was introduced in Section 1.2.1 in an informal fashion. Presented more formally, risk is represented by Kaplan and Garrick as a set, R, defined as:

$$R = (S_i, pS_i, \mathbf{cS}_i),\ i = 1, 2, \ldots nS$$

where S_i is a set of similar occurrences; pS_i is the probability that an occurrence in set S_i will take place; $\mathbf{cS}_i$ is a vector of consequences associated with S_i; nS is the number of sets selected for consideration; and the sets S_i have no occurrences in common (i.e., the S_i are disjoint sets). This representation formally decomposes risk into the three questions described previously:

1. S_i, “What could happen?”
2. pS_i, “How likely is it to happen?”
3. $\mathbf{cS}_i$, “What are the consequences if it does happen?”

The S_i are "scenarios", the pS_i are scenario probabilities, and the $\mathbf{cS}_i$ are consequences of scenarios. This equation serves as the mathematical basis for PA calculations. As described in Section 1.2.1, there is also a fourth question that must be addressed in the PA calculations:

4. "What is the uncertainty in the answers to the first three questions?" or "What is the level of confidence in the answers to the first three questions?"

Three major sources of uncertainty must be considered in PA (Gallegos and Bonano 1993):

- Uncertainty in the future state of the disposal system,
- Model uncertainty, and
- Data and parameter uncertainty.

Uncertainty in the future state of the system is represented by Questions 1 and 2 of the risk triplet above. The uncertainty characterized by the scenarios (S_i) and the scenario probabilities (pS_i) results from a perceived randomness in future occurrences that could take place at the facility under consideration. This uncertainty arises because it is difficult to forecast exactly how a disposal system will evolve over tens or hundreds of thousands to millions of years. The system's evolution can be affected by natural events (e.g., earthquakes, volcanoes, etc.) as well as external and/or human-induced events (e.g., drilling). This form of uncertainty is referred to as aleatory uncertainty and is also known as stochastic, type A, or irreducible uncertainty. Aleatory uncertainty is typically addressed in a PA model through scenario construction and screening, where each retained scenario represents a possible future state of the disposal system and is weighted according to its probability of occurrence.

Once the set of scenarios to be analyzed has been developed, conceptual, mathematical, and computational models are developed to represent each scenario to allow estimation of the consequence(s) if the scenario was to occur. Independent of their complexity, models are simplifications of reality; therefore, model development necessarily entails making assumptions. Model uncertainty is introduced due to (1) the accuracy and appropriateness of the assumptions, (2) the completeness or exhaustiveness of the assumptions, and (3) the complexity of interactions between the assumptions. One approach to reduce model uncertainty is to introduce alternative conceptual models.

Data and parameter uncertainty arises from incomplete knowledge of the present system and the inherent complexity of natural systems. Often, the parameter values are not measured directly but rather are inferred from measured data that require interpretation. Uncertainty in measured data can be introduced from limitations in measuring instruments, inability to fully characterize spatial and other variabilities, or human error. Uncertainty in parameter values comes from the uncertainty in the measured data as well as the uncertainty that may be associated with data interpretation. Many parameters used in the PA models of complex systems like a geologic repository, such as parameters derived from site data on common rock properties (e.g., porosity and permeability), have natural variability. Data and parameter uncertainty also arises when, for example, the future state of a changeable property—which cannot, obviously, be measured—must be estimated. This form of uncertainty is referred to as epistemic uncertainty and is also known as subjective, state-of-knowledge, type B, and reducible uncertainty (Hansen, Helton and Salaberry 2010).

Epistemic uncertainty is typically accounted for by developing distributions of values for important, imprecisely known parameters rather than using single deterministic values. Each distribution describes a range of values within which the true value is believed to fall, with an expected value that corresponds to the best estimate of the true value. Of course, not all parameters in the PA calculations require uncertainty distributions; properties that are well known or uncertain parameters that have been shown to have little or no effect on overall performance are represented by single fixed values. In cases where realistic uncertainty distributions or parameter values cannot be adequately justified based on available information, parameter distributions or values may be chosen that are deliberately conservative, in the sense that they result in a calculation that shows worse performance than would result from more realistic input values. The use of conservative or bounding values for input parameters has a potential to mask effects of processes that, if treated more realistically, might reveal important insights about system performance; additionally, a parameter considered conservative at a subsystem scale might actually have nonconservative effects at the scale of the total system or as a result of interactions with other conservatively modeled parameters. For these reasons, conservatism should be used cautiously in PA.

One key attribute of the SNL PA methodology is the identification and quantification of uncertainties using all of the available information (including expert opinions) without biasing the approach optimistically or pessimistically.

Construct integrated PA model and perform calculations. Once parameter values and uncertainty distributions are assigned and the computational models are linked together, the consequence analysis, a calculation of overall system performance, can be performed. A "consequence model" is, in this way, not a single model, but a suite of many submodels that interface through a system controller such as MARINRAD, a software controller developed for the SDP (see Section 2), or CAMCON, applied in the PAs for WIPP (see Section 4) and for INL HLW disposal (Section 7).

In performing the overall system calculations, uncertainty associated with the selection of scenarios is included in the PA by conducting separate analyses for each scenario class. The parameters are sampled and propagated through the sequence of models associated with each scenario to generate a distribution of potential outcomes. Parameter uncertainty is propagated into the PA by conducting multiple calculations for each scenario using values sampled from the distributions of possible values (e.g., Monte Carlo simulation). Each individual calculation uses a different set of sampled input values. In a statistical sense, the result of each individual PA calculation represents a different possible realization of the future overall performance of the system, consistent with the uncertainty in the input parameters. When using Latin hypercube sampling (LHS), the sampling method applied in nearly all SNL PAs, the variable space is sampled with few samples, so the number of model runs can be comparatively small. Iman and Helton (1985) have shown that, when using the LHS technique, running a number of simulations equal to 4/3 the number of uncertain parameters usually gives statistically satisfactory results and represents a good balance of accuracy and computational cost for PA models with a large number of uncertain parameters.

Uncertainty and sensitivity analysis. The results of the Monte Carlo consequence analyses are analyzed further using statistical techniques, such as step-wise linear regression, to identify and rank the importance of uncertain parameters. Caution must be used in interpreting the results of the sensitivity analysis because the rankings are conditional on the modeling assumptions and parameter distributions, because the processes not included in the models cannot be evaluated, and because fixed-value parameters cannot be evaluated.

The sensitivity analysis identifies parameters for which reductions in uncertainty will reduce the uncertainty in the estimate of overall system performance. Sensitivity analyses are included in the PA to enable the process to be iterative, providing feedback during research and development activities to efficiently and effectively reduce important sources of uncertainty. The sensitivity analyses also allow the detailed PA models to be simplified for more rapid iterative calculations by turning off unimportant functions. Thus, the results of the sensitivity analysis are used to guide programmatic decision-making, which is especially important as the project evolves.

Evaluate performance. Results of preliminary performance assessments can be analyzed at the system and subsystem levels to identify the models and parameters that have the greatest effect on the behavior of the system. Identification of the uncertainties that are most important in preliminary PAs can help guide testing for site characterization, model development and streamlining, and repository design through a directed science and testing program. When the PA models are sufficiently well developed and documented to support regulatory decisions, results can be used to evaluate compliance with applicable long-term requirements. The steps in the PA process are repeated, as needed, until a final decision is reached.

Directed science and testing program. Information from the overall performance evaluation and uncertainty and sensitivity analyses serves to identify important parameters and systems for further investigation. This may include identifying systems whose performance can be improved by modifications to the design, or parameters with uncertainties that, if reduced through further site or laboratory investigations, would significantly increase confidence in the overall PA results. The PA process thereby helps inform programmatic decision-making toward the testing and scientific investigations that will most effectively and efficiently improve the accuracy and confidence in PA results and toward design decisions most likely to improve real system performance.

Eventually, a PA helps outline performance confirmation testing and monitoring for the constructed and operating systems that serve to confirm that the forecasts from the PAs correspond to actual performance. The enhanced PA system SNL has proposed for application to carbon sequestration and storage (Section 8.7), implementing real-time system performance optimization and data fusion can be viewed as a dynamic extension of a directed science and testing program into ongoing operation of systems for carbon dioxide injection and reservoir sequestration.

1.2.3 Evolution and Adaptation in the SNL PA Methodology

Since the SNL PA methodology was first developed in the 1970s, it has evolved to take advantage of improved technology and to reflect lessons learned from practice, and it has been adapted to different radioactive waste disposal problems as well as to different regulatory schemes. In addition, the methodology has been framed and characterized differently over time, even in cases where there were no significant differences in general methodology being applied.

An outline of the evolution and adaptation of the PA methodology over time provides a broad and detailed picture of the SNL PA Methodology.

Table 1 is a matrix of steps in the PA methodology, as described in a selection of methodology summaries taken from SNL PAs from 1982 to the present. The matrix is arranged to show the fundamental consistency of the methodology, and indicates the steps that are frequently left unstated as implicit steps in the process. Even though they are not explicitly included in the PA documentation's description of their methodology, these steps are typically discussed extensively and they are usually exercised clearly in the PA. Implicit steps are often those that are of less immediate focus of a given PA. For example, the WIPP, GCD, and YMP PAs had relatively well-established performance goals by the time the PAs selected as examples were conducted, and they omit defining the performance goals in their summary-level outlines of the PA methodology. In contrast, in a description by Klett (1997b) of the SDP PA, for which no legal or regulatory performance criteria existed, this step is very explicit because of the importance that definition of performance goals had in that PA.

The earliest graphical representations of steps in the PA methodology tended to focus on the relationship of models. The 1973 SDP "review strategy," shown in Figure 3 (top) is perhaps the earliest example. The 1973 "review strategy" and the 1978 "structure of the [PA] methodology" presented in *Risk Methodology for Geologic Disposal of Radioactive Waste: Interim Report*, which was prepared by Campbell et al. (1978) for the NRC and shown here in Figure 4, are primarily outlines of information flow or modeling strategy. The 1979 SDP "research approach" illustration shown at the bottom of Figure 3 (Anderson 1979, Anderson, Boyer, et al. 1980, SNL Seabed Programs Division 1980) may be the earliest process-oriented depiction of the steps in a PA methodology. This illustration represented the process of developing, exercising, evaluating, and iterating component models, rather than the overall methodology.

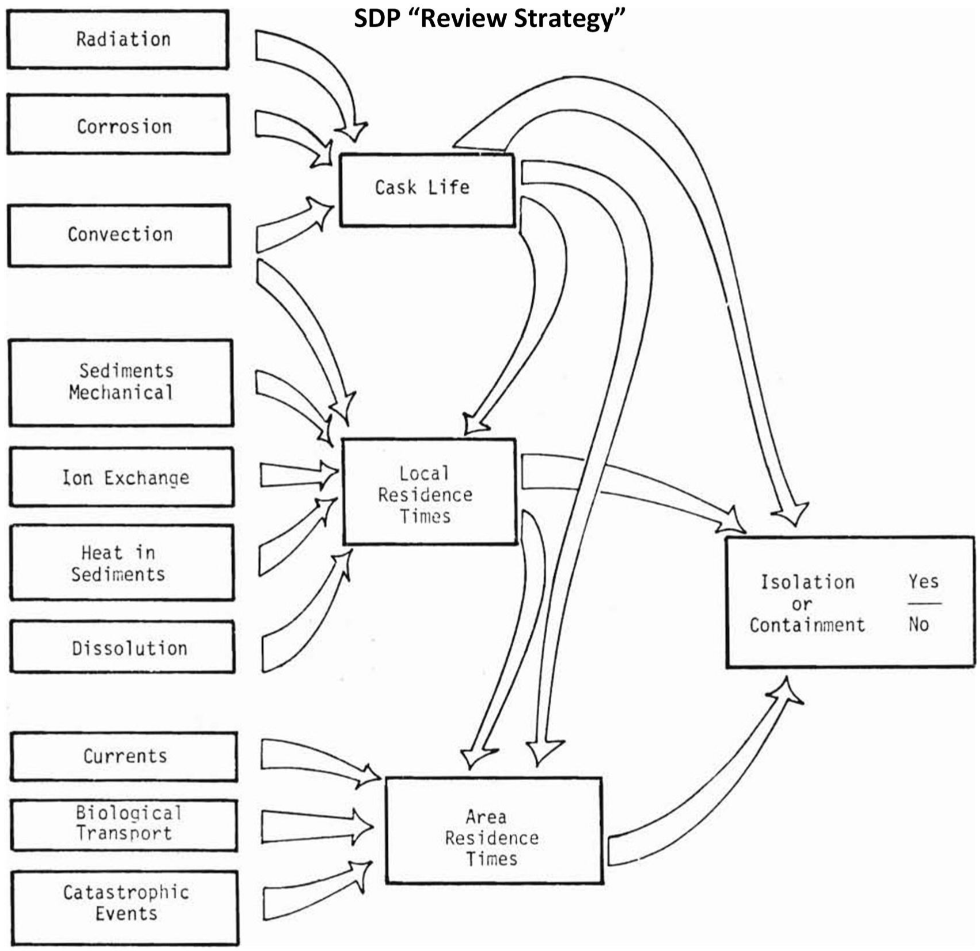
The first mature formulation of the SNL PA methodology was probably that of Cranwell, Campbell and Helton, et al. (1987), who delivered the *Risk Methodology for Geologic Disposal of Radioactive Waste: Final Report* to the NRC in 1982, though it was not published until 1987. The methodology then explicitly included a scenario development process as well as sensitivity and uncertainty analysis. The seven-step outline of PA methodology from the Final Report, provided in Table 1, provided the first comprehensive general outline of the PA methodology. The updated outline of the PA methodology included in 1982 in the Risk Methodology Final Report (Cranwell, Campbell and Helton, et al. 1987) shows both increased detail in the modeling plan and indicates the general process steps for PA. Their more detailed illustration of the process is shown in Figure 5.

Table 1. Matrix providing a historical comparison of descriptions of the SNL PA methodology

THIS REPORT	Cranwell, Campbell et al. 1982/1987 “Risk Methodology for Geologic Disposal of Radioactive waste: Final Report” SAND81-2573 NUREG/CR-2452	SNL Generic PA Methodologies developed for NRC (early 1990s)	Barnard et al. 1991 YMP TSPA-1991	Rechard 1993a Intro to Mech. Of PA—WIPP SAND93-1378; also INL 1st PA SAND93-2330 S1.3.2	Klett 1997b Subseabed PA overview (SAND93-2723) describing the “second iteration”	Cochran et al. 2001 GCD PA	Ho Mixed Waste Landfill PA SAND2002-3131	Yucca Mountain TSPA-LA	Enhanced PA System for Carbon Sequestration
1. Define performance goals	[*implicit*]	[*implicit*]	[*implicit*]	[*implicit*]	2. Develop interim radiological safety standards for subseabed disposal of high-level waste.	1. Define Performance Objectives	[*implicit*]	[*implicit*]	[*implicit*]
2. Characterize System (waste, facility, and site)	1. Characterization of the site and disposal facility to be analyzed (e.g., determination of stratigraphy, hydrologic and geohydrologic properties, resource potential, waste form properties)	1. System Description	[*implicit*]	1. **Disposal-system and regional characterization** entails data collection on waste properties, facility design, regional geology, and regional hydrology.	[3. continued] …in conjunction with the engineering and scientific task groups, using the latest and most complete processes definitions and data.	2. Assimilate Existing Site Information	[*implicit*]	[*implicit*]	[*implicit*]
3. Identify scenarios for analysis; a. Identify, screen relevant features, events, and processes (FEPs) b. Construct scenarios	2. Identification of a collection of scenarios to be analyzed	2. Scenario Development and Screening	1. Screen scenarios to determine which phenomena are to be modeled in the exercise.	2. **Scenario development** identifies and selects features, events, and processes that collectively comprise the scenarios, $S_i(x)$, through which contaminants might be released to the accessible environment	[*implicit*]	3. Scenario Development and Screening	1. Develop and screen scenarios based on regulatory requirements (performance objectives) and relevant FEPs	1 Develop/ Screen Scenarios	1. FEP Evaluation & Screening 1/2. Scenario Definition
c. Estimate scenario probabilities			2. Estimate probabilities of occurrence of those scenarios.	3. **Probability estimation** models likelihoods that the various scenarios will occur, $P(x,S_i(x))$.					
4. Build models and abstractions of relevant FEPs; a. Conceptual models b. Mathematical models c. Computational models	3. Selection of a suitable sequence of codes for use in analyzing each scenario (e.g., repository evolution, radionuclide transport in the geosphere and biosphere, and dosimetry and health effects)	[*implicit*]	3. Choose the conceptual models to be assumed in the modeling of releases.	[*implicit, included as part of scenario development*]	3. Modify and expand the nuclide transport and biological models in conjunction with the engineering and scientific task groups, …	4. Model Development / …	2. Develop models of relevant FEPs	2. Develop Models and Abstractions	2/3. develop appropriate computational models for selected FEPs and scenarios; constrain model input parameters
5.Quantify uncertainty;	4. Selection of code input data (with suitable correlations and measures of uncertainty) for the sequence of codes associated with each scenario	4. Uncertainty Analysis	4. Estimate parameter values and the uncertainties in them.	[*implicit, included as part of consequence analysis*]	[*implicit*]	[4. continued] Parameter Analysis	3. Develop values and/or uncertainty distributions for uncertain input parameters	3. Estimate Parameter Ranges and Uncertainty	4. Uncertainty Quantification
6. Construct Integrated TSPA Model and Perform Calculations	5. Sampling of input data for each scenario according to assigned correlations and distributions 6. Propagation of resultant data and appropriate intermediate calculations through a sequence of models to yield a distribution of potential outcomes (e.g., discharge curves, total integrated discharge, health effects) for each scenario 8. Combining the results for each scenario to arrive at total risk.	3. Consequence Analysis	5. Calculate releases.	4. **Consequence analysis including uncertainty propagation** calculates the potential amounts of contaminants that might be released for a given scenario, $C(x,S_i(x))$, and includes the quantitative evaluation of uncertainties associated with those predictions. 5. Long-term regulatory compliance assessment involves the construction of CCDFs and other performance and uncertainty metrics and …	6. Risk analyses that include transportation, emplacement, an undisturbed repository, and abnormal natural events. Best estimate, bounding, and probabilistic assessments were conducted.	5. Consequence Analysis	4. Perform calculations and …	4. Construct Integrated TSPA Model and Perform Calculations	6. Project Risk Assessment and …

THIS REPORT	**Cranwell, Campbell et al. 1982/1987** "Risk Methodology for Geologic Disposal of Radioactive waste: Final Report" SAND81-2573 NUREG/CR-2452	**SNL Generic PA Methodologies developed for NRC (early 1990s)**	**Barnard et al. 1991 YMP TSPA-1991**	**Rechard 1993a** Intro to Mech. Of PA—WIPP SAND93-1378; also INL 1st PA SAND93-2330 S1.3.2	**Klett 1997b** Subseabed PA overview (SAND93-2723) describing the "second iteration"	**Cochran et al. 2001** GCD PA	**Ho** Mixed Waste Landfill PA SAND2002-3131	**Yucca Mountain TSPA-LA**	**Enhanced PA System for Carbon Sequestration**
7. Uncertainty and sensitivity analyses	7. Analysis of the results generated for each scenario (e.g., sensitivity and uncertainty analyses, …	5. Sensitivity Analysis	6. Interpret results	6. Sensitivity analysis determines the individual parameters and model forms that most influence performance metrics and thereby provides guidance to WIPP project managers on where to direct resources to further evaluate uncertainty of the parameters.	4. Sensitivity analyses to define where the most significant design improvements can be made and identifying the processes and data that have the greatest impact on risk. 5. Attenuation factor analyses to define the risk reduction effectiveness of each component and the entire disposal system, the resilience of the system if component effectiveness deviates, and component stability with changing environments.	7. Sensitivity Analysis	[4. continued] …sensitivity/ uncertainty analyses	5. Evaluate Performance	5. Sensitivity Analysis
8. Evaluate performance	[7. continued] …comparison with environmental standards)	[*implicit*]		[5. continued] [Long-term regulatory compliance assessment involves…] …their comparisons with the relevant long-term environmental regulations.	[*implicit*]	6. Compliance Decision (Acceptable performance?)	[*implicit*]		[6. continued] …and Decision Analysis]
9. As needed, direct science and testing program					7. Interpretation of the results from activities 4-6 to produce revisions in design, site selection, and research requirements. The practicality of the more inclusive interim standards was evaluated. Risks were evaluated using the interim standards and other metrics. 1. Upgrade the reference design based on design requirements developed during the first iteration.	8. Assess Ability to Collect Required Data 10. Data collection and/or Site Alteration	5. Document results and provide feedback to previous steps and associated areas to improve calculations, as needed	6. Iterate	**Inverse Model Components:** • Real-time optimizationSystem performance optimizationData fusion

1973
SDP "Review Strategy"



1979
SDP "Research Approach"

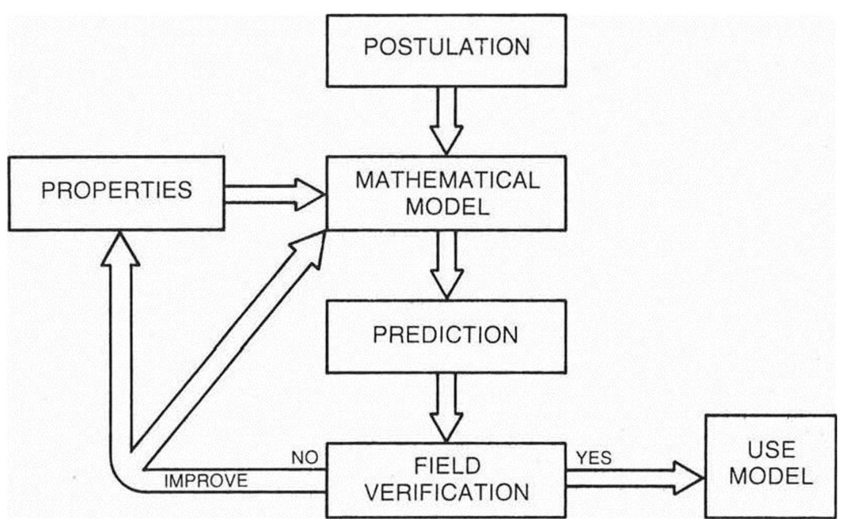


Figure 3. Early PA approach illustrations: 1973 SDP "research strategy" and 1979 SDP "research approach"

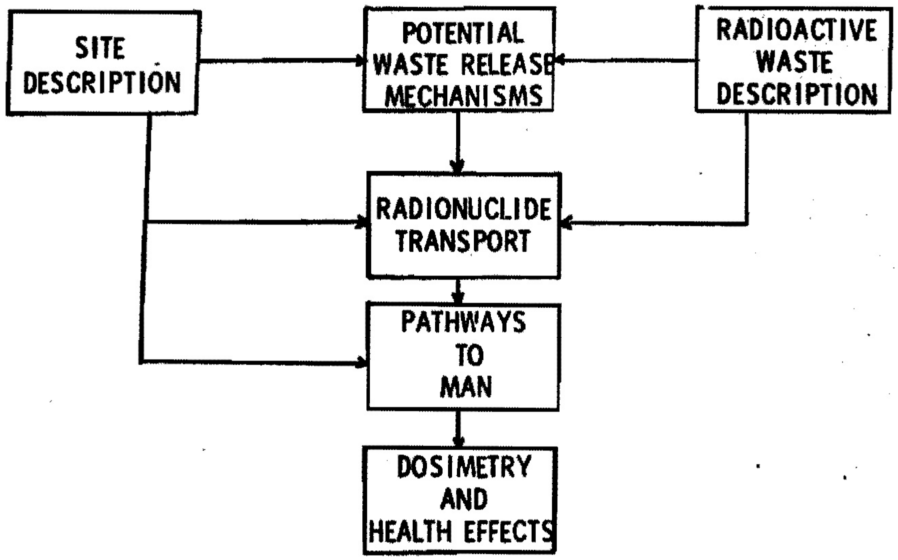


Figure 4. 1978: PA methodology from the Interim Report by Campbell et al. (1978)

The PA methodology outlined by Davis, Price, et al. (1990, Figure 3.1) and Bonano and Wahi (1990, Figure 1), shown in Figure 6, was implemented in each of the PAs for generic HLW repositories in a variety of geologic media (salt, basalt, and tuff) that SNL prepared for the NRC in the 1980s and early 1990s (see Section 3). These representations of the PA methodology gave greater detail to outputs of consequence modeling for comparison to intermediate regulatory performance measures such as waste package lifetime or groundwater travel time. The consequence modeling section of the methodology is divided into several components and assembled in a form that may be used directly in addressing such specific criteria as well as providing input for the estimation of another criterion such as the dose calculation (Davis, Price, et al. 1990). The general structure of the methodology is generic (i.e., irrespective of the host geologic formation). The consequence modeling component of the methodology would accommodate, in principle, capabilities to simulate the source term, groundwater flow, radionuclide transport in the geosphere, radionuclide transport in the biosphere, and dosimetry and health effects. These capabilities were designed to allow the assessment of compliance not only with the containment requirements of EPA's 40 CFR Part 191 standard, but also the individual protection requirements and the groundwater protection requirements as well as the subsystem requirements in 10 CFR Part 60: waste package lifetime, release rate from the engineered barrier system, and groundwater travel time.

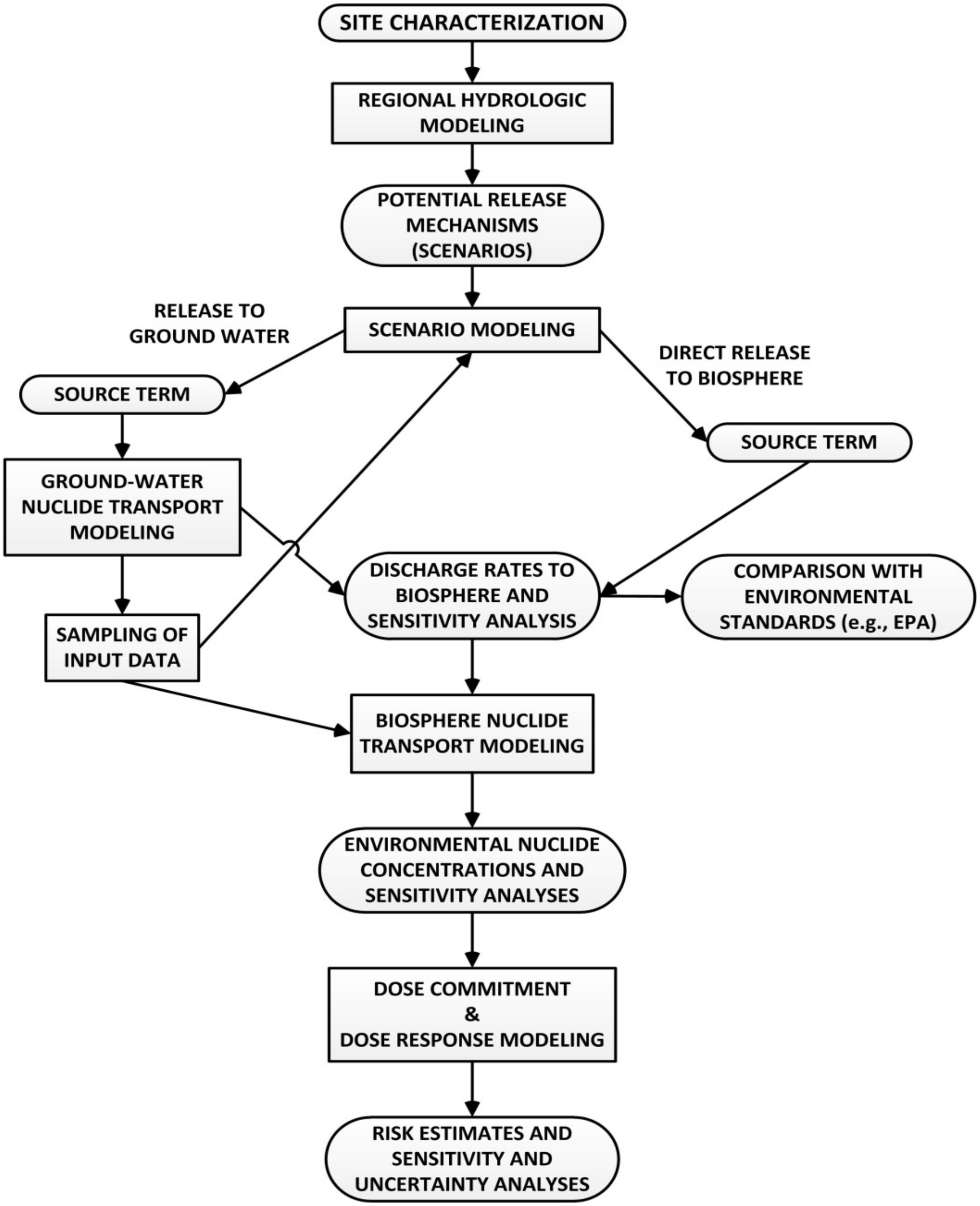


Figure 5. **1982: General PA methodology, outlined in the Final Report by Cranwell, Campbell et al. (1987)**

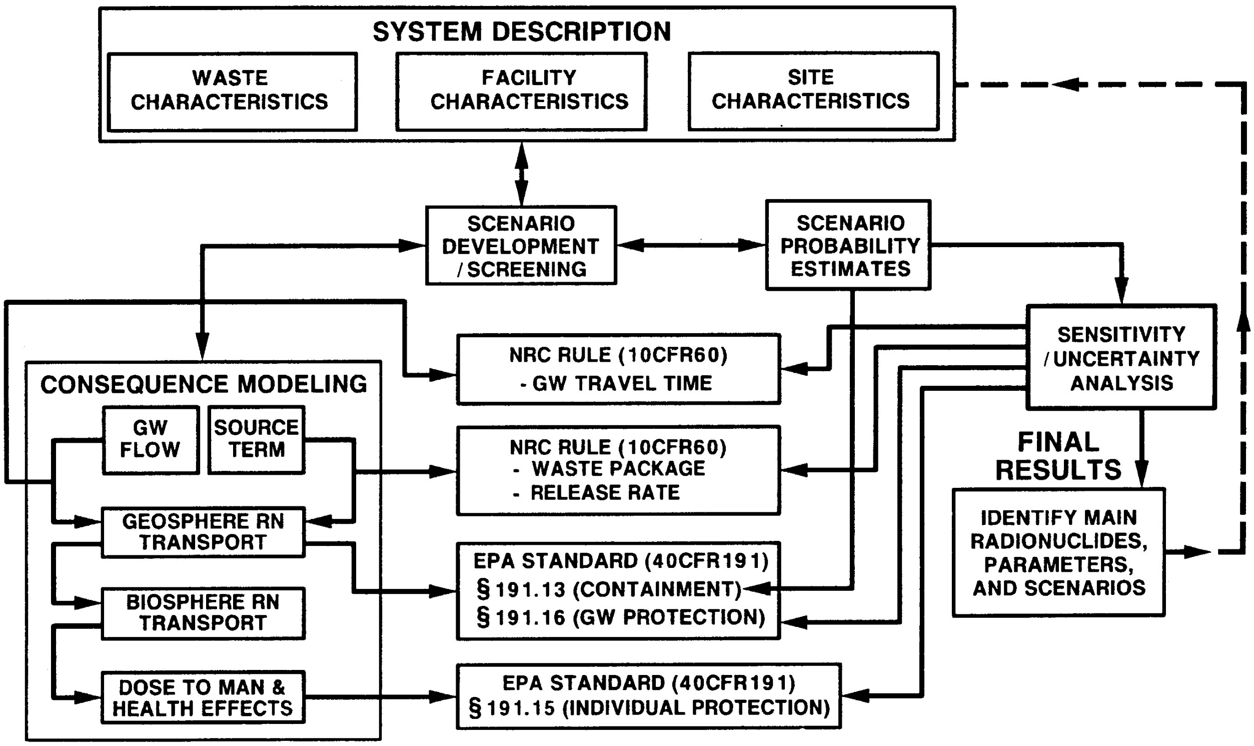


Figure 6. 1990: SNL/NRC methodology applied in PAs for generic repositories in salt, basalt, and tuff

Illustrations developed by Rechard to explain the application of the SNL PA methodology to the WIPP PA (Rechard 1993a, Figure 1.2-1) and to INL HLW PA calculations (Rechard 1993b, Figure 1-1) are particularly helpful in clarifying the general PA methodology in relation to the detailed analyses customized to particular disposal approaches and regulatory contexts. In Rechard's outline of the WIPP PA (provided in Figure 7 and also summarized in Table 1) the methodology is described in six steps rather than the nine steps described in Section 1.2.2 of this report. These illustrations show how the Kaplan and Garret risk triplet is addressed generally by the methodology and in detail by the project specific PA application. Comparison of the illustration for the WIPP PA (Figure 7) with the version from the INL HLW PA (Rechard 1993b, Figure 1-1) reveals how the process is modified to address different regulatory criteria, with the 1993 INL PA process altered slightly to address NRC regulation 10 CFR Part 60 in addition to the EPA regulation 40 CFR Part 191 and to omit consideration of inapplicable Resource Conservation and Recovery Act regulations of 40 CFR Part 286.

Based on this methodology for HLW PA, Bonano and Gallegos (1991) proposed a generic methodology for assessing disposal of all types of wastes (radioactive, hazardous, and mixed), recommending that a probabilistic risk approach that was capable of allowing explicit consideration of uncertainties, which—while inherent in any quantitative estimate of potential environmental and health impacts from waste disposal—is not generally prescribed by regulations for waste disposal except for HLW. Their proposed methodology, outlined in Figure 8, was notable in recognizing the inclusion of economic risks, allowing decision makers to evaluate all available information that will impact the decision and thereby allow for thorough scrutiny and defensibility of the decision.

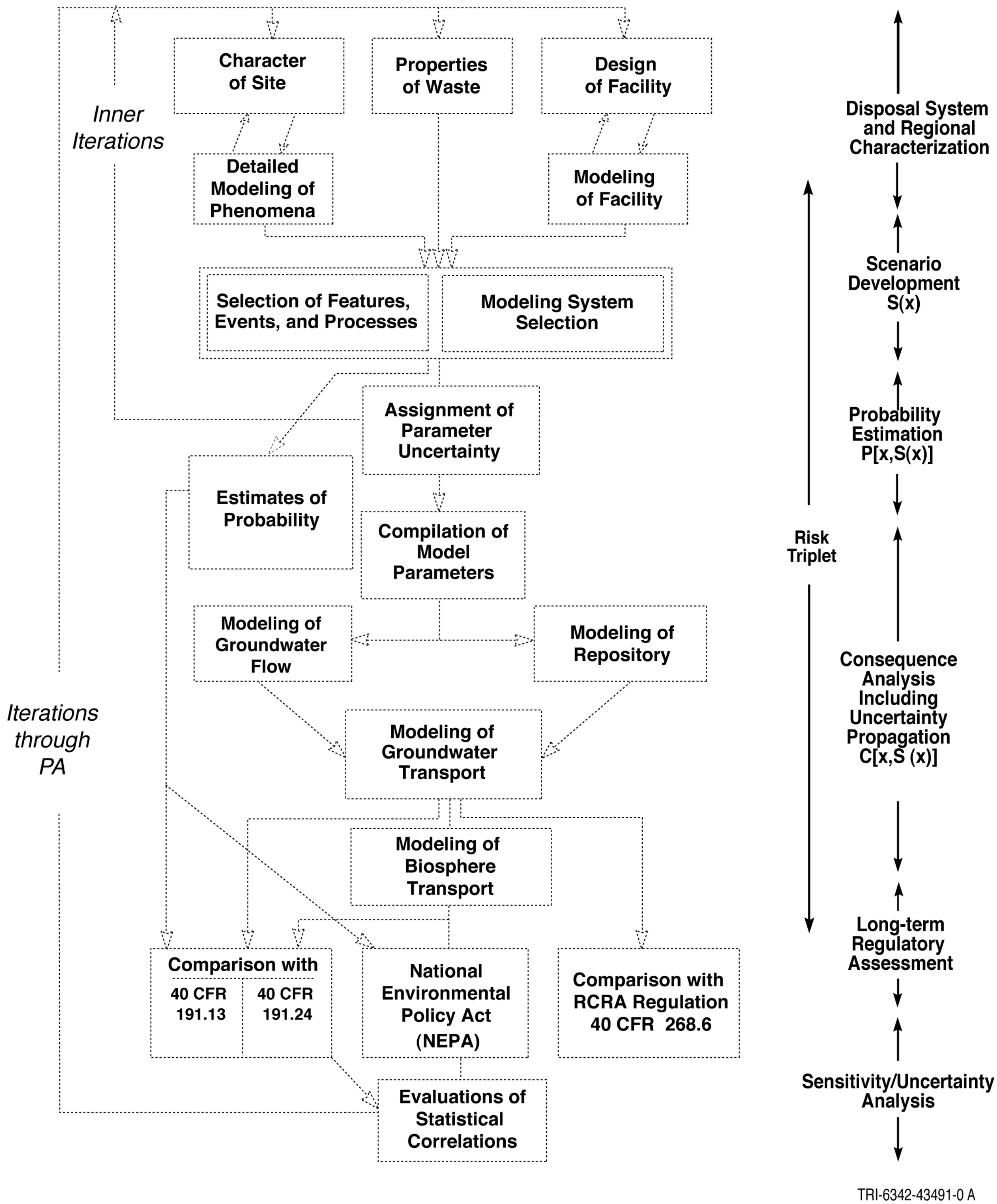


Figure 7. 1993: WIPP PA methodology (Rechard 1993a), showing detailed PA process steps for WIPP in relation to general SNL PA methodology and application of the risk triplet

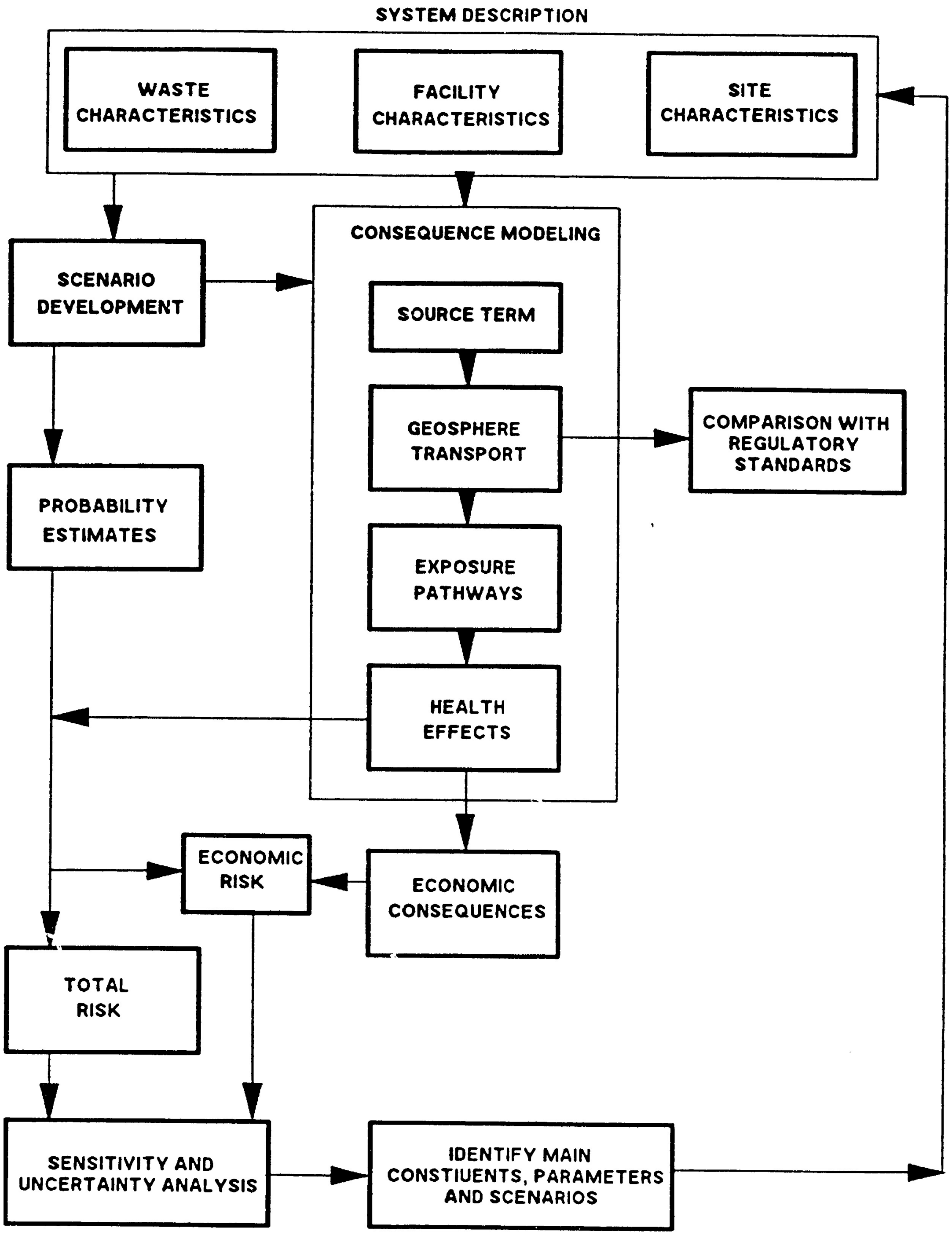


Figure 8. 1991: PA methodology for waste disposal (radioactive, hazardous, and mixed), of Bonano and Gallegos (1991), demonstrating how economic analyses could be included

Considerations of economics and efficient management of large engineering projects also led the development of the Systems Prioritization Method (see Section 4.3) developed and exercised during the WIPP program, described in Section 4.3. The Systems Prioritization Method, illustrated in Figure 9, was designed to: (1) identify programmatic options (testing and technical investigations) and their costs and durations; (2) utilize the PA models to help analyze combinations of activities in terms of their predicted contribution to long-term performance of the WIPP disposal system; and (3) use those results in an analysis of cost, duration, and performance tradeoffs.

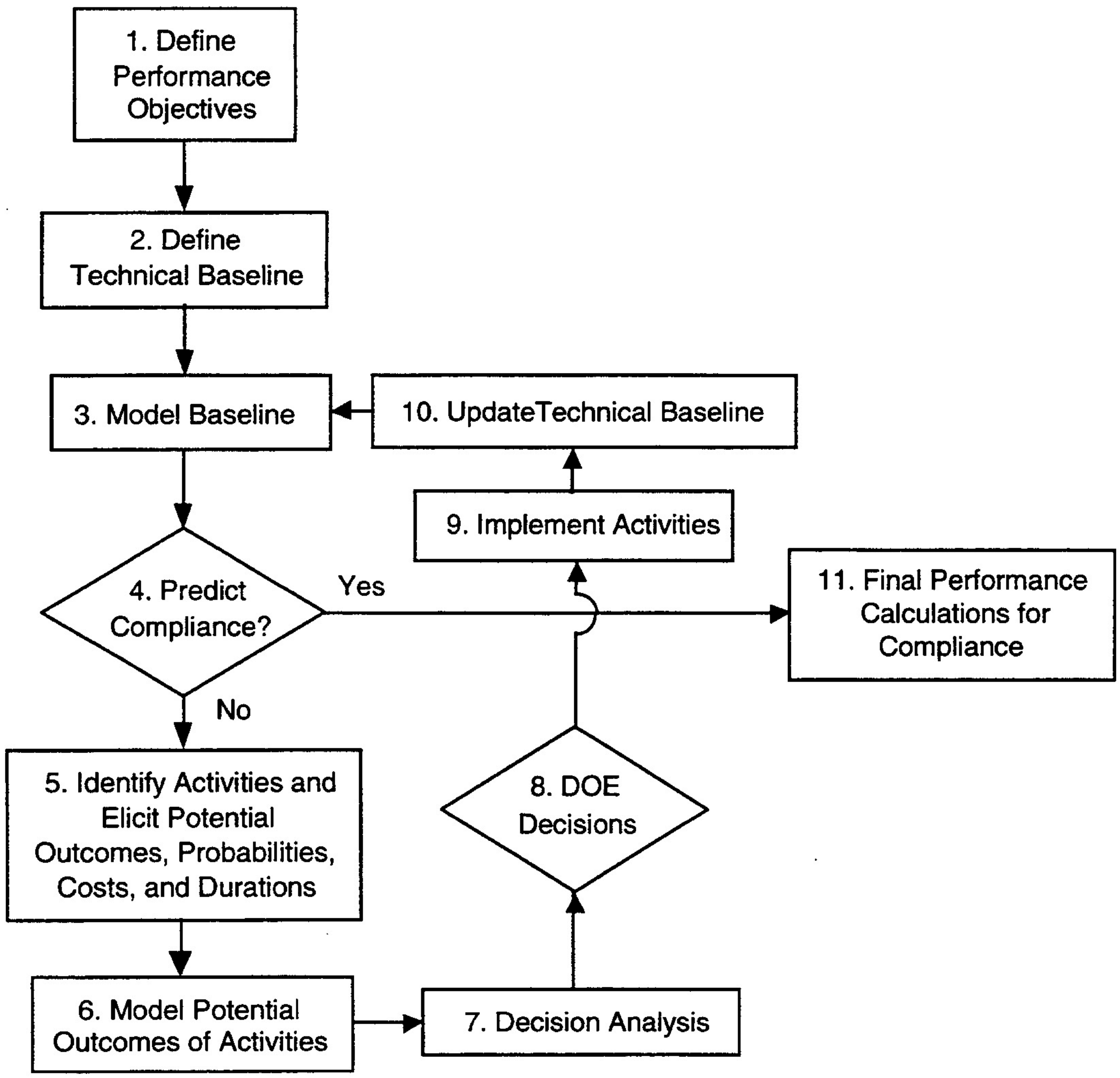


Figure 9. 1995: System Prioritization Method applied at WIPP to support programmatic decision-making

The depiction of the PA methodology in Figure 10, used in describing the application of the PA methodology for the YMP, amplifies lessons learned from WIPP and YMP repository PA programs (Bonano, Kessel and Dotson 2010), emphasizing the iterative nature of the PA process and the information that is produced by and that must be effectively managed by the PA process.

This construction of the SNL PA methodology shows how high-level requirements and objectives are clearly defined, translated into subsystem and research and development requirements, documented, and controlled. The PA method also allowed data and other information to be evaluated and prioritized so that research and development activities could be focused on those activities most important to meeting the performance requirements.

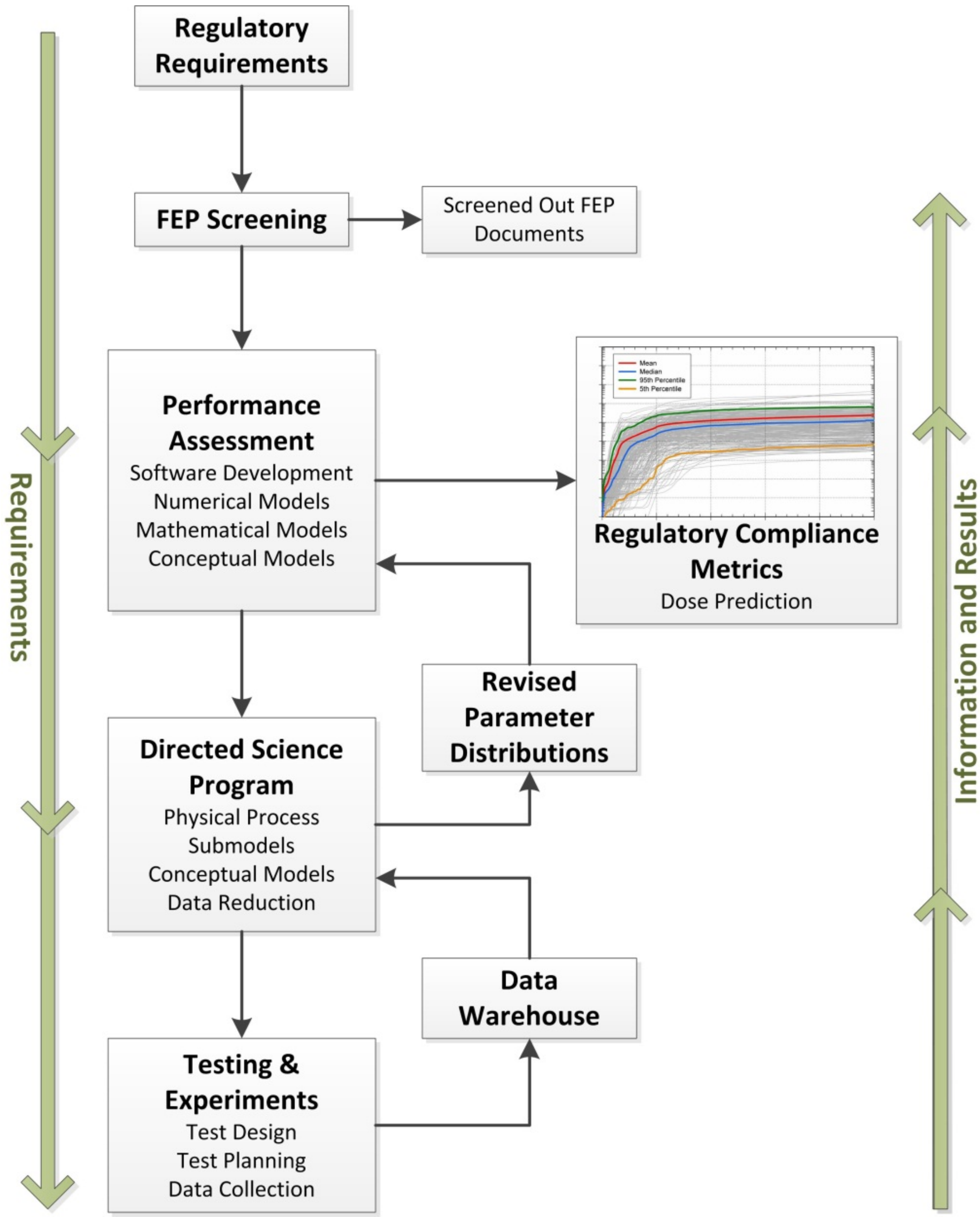


Figure 10. 2008: Illustration of the PA methodology for the YMP, emphasizing information management

In order to maximize the defensibility and credibility of the technical work, a central data warehouse was used on Yucca Mountain and a central parameter database was used on WIPP. Integration and communication was required between the project scientists and the PA analysts. Important lessons learned from these projects were summarized by Bonano, Kessel, and Dotson (2010) as follows:

- All assertions of fact and assumptions require documentation
- All model calculations and decisions require retrievability, traceability, and reproducibility
- All modeling assumptions require traceability to supporting data or evidence
- All data require traceability to methods of collection, calibrations of measuring instruments, and personnel and their qualifications
- All documented work requires sufficient transparency to ensure reproducibility.

The last example, Figure 11, illustrates a potential adaptation of the SNL PA methodology proposed by Wang, Dewers, et al. (2010) to the carbon sequestration in underground reservoirs. This methodology, discussed in Section 8.7, enhances the standard, forward-looking methodology that forecasts long-term behavior of a complex geologic disposal system with real-time monitoring and data analysis, to allow operational changes such as adjusting injection rates of carbon dioxide.

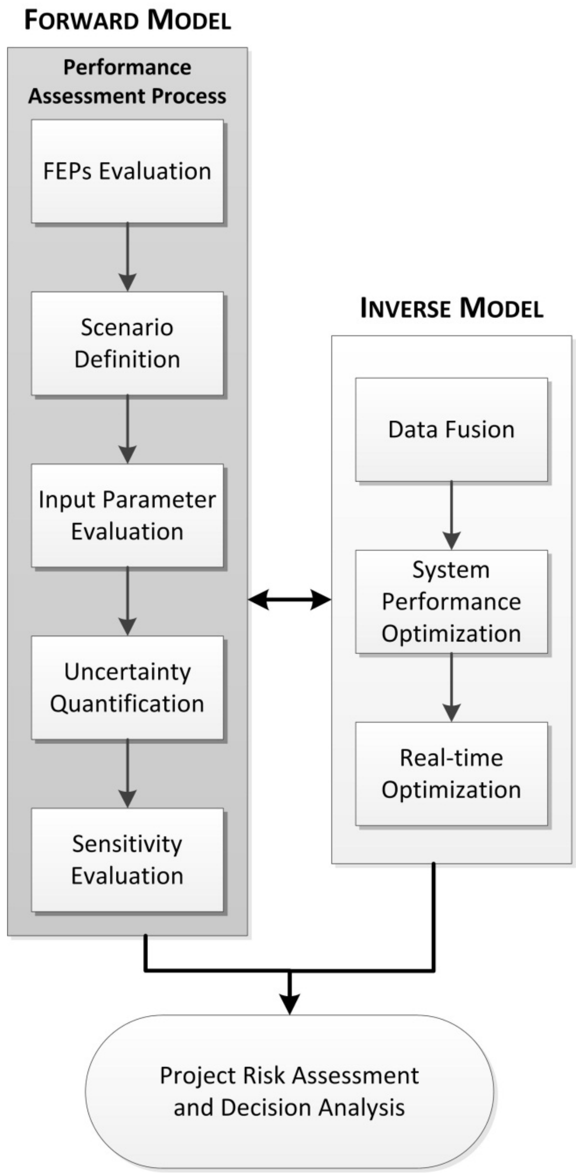


Figure 11. 2010 PA methodology adapted to an enhanced PA system for carbon capture and sequestration

2. INTERNATIONAL SUBSEABED DISPOSAL PROJECT (1973–1987)

In the 1970s and early 1980s, the probabilistic risk assessment methodology was rapidly developing with several concurrent applications that provided the foundation for later work. Reactor safety studies, the international Subseabed Disposal Project (SDP), and technical support for the NRC (see Section 3) provided the building blocks for the SNL PA methodology upon which later refinements were made.

The SDP was begun at SNL in 1973 with a workshop meeting with 14 participants. Within a few years, the effort grew into an international effort involving 200 scientists from 10 countries (Nadis 1996). Between 1976 and 1987, the SDP was part of the international Seabed Working Group of the Organisation for Economic Co-Operation and Development/Nuclear Energy Agency, which oversaw the coordinated multinational investigations into seabed disposal (Anderson and Murray, Feasibility of Disposal of High-Level Waste into the Seabed, Volume 1, Overview of Research and Conclusions 1988). A timeline of the program, shown against the backdrop of other contemporary developments in PA is shown in Figure 12.

The major questions in the feasibility assessment conducted by the SDP were:

- Can an acceptable site for subseabed HLW disposal be found and characterized?
- Can waste canisters be reliably and economically transported and emplaced in deep ocean geologic formations?
- Are radiological risks from emplaced waste, accidents, and abnormal events below the limits for similar waste disposal processes?

The program explored the feasibility of disposal in the clay sediments beneath mid-ocean gyres on the deep abyssal plains near the centers of the tectonic plates, approximately 6,000 m below the ocean's surface. The disposal system considered for high-level radioactive waste in the deep seabed involved the enclosure of the waste in an insoluble solid (borosilicate glass) inside a waste canister. Many emplacement options were considered—15 were illustrated by Anderson and Murray (1988, p. 24)—but eventually only two methods of waste emplacement were considered. The reference emplacement approach used free-falling penetrators weighing several metric tons, dropped from a ship, falling to the ocean bottom, and burying themselves 30 to 70 m into the soft sediments of the seabed. In the second emplacement approach, holes would be predrilled into the sediments, and a column of waste packages would be stacked in the holes, with the remainder of each hole being backfilled and sealed. These disposal and emplacement concepts are illustrated in Figure 13.

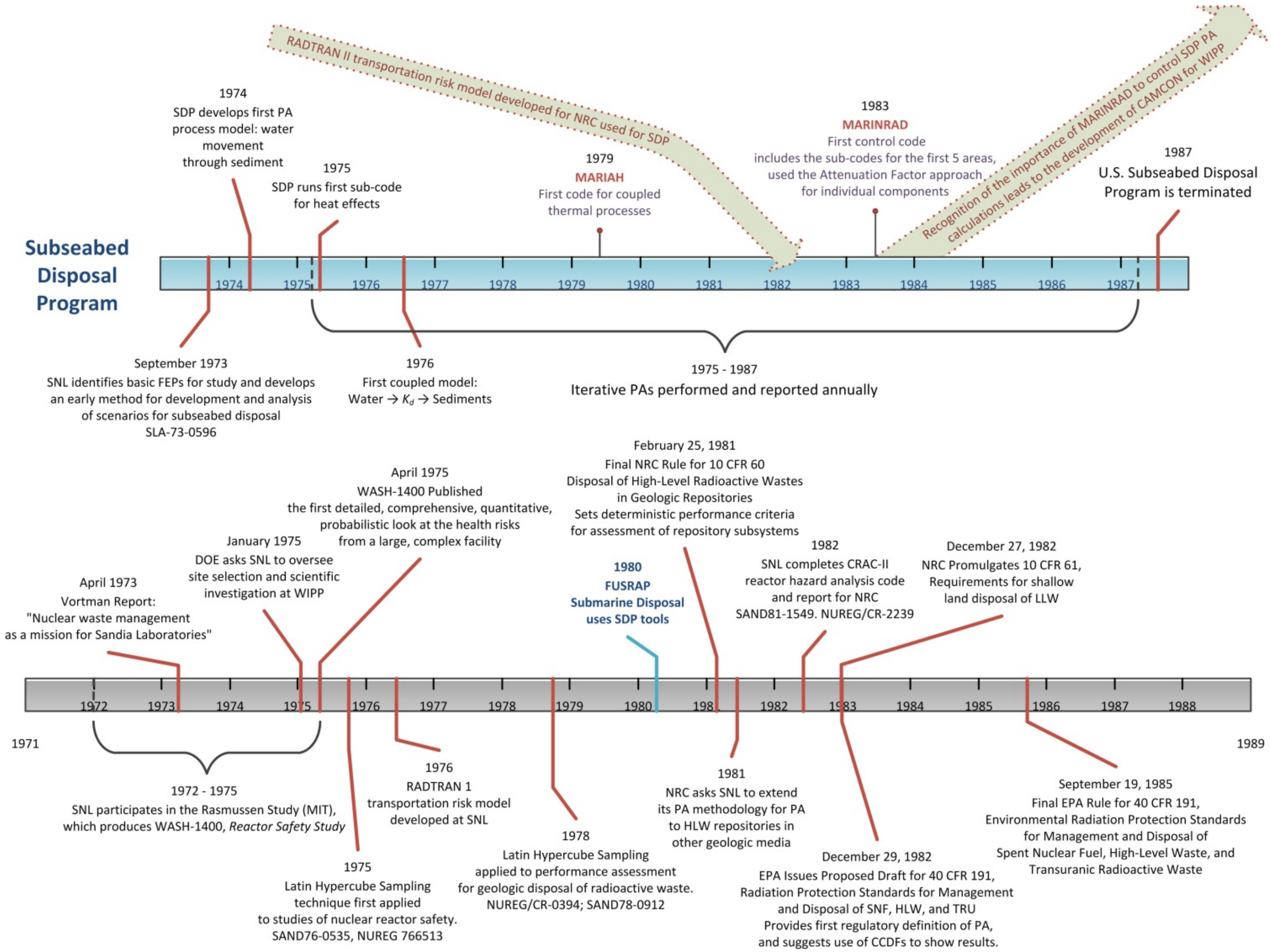


Figure 12. SDP timeline

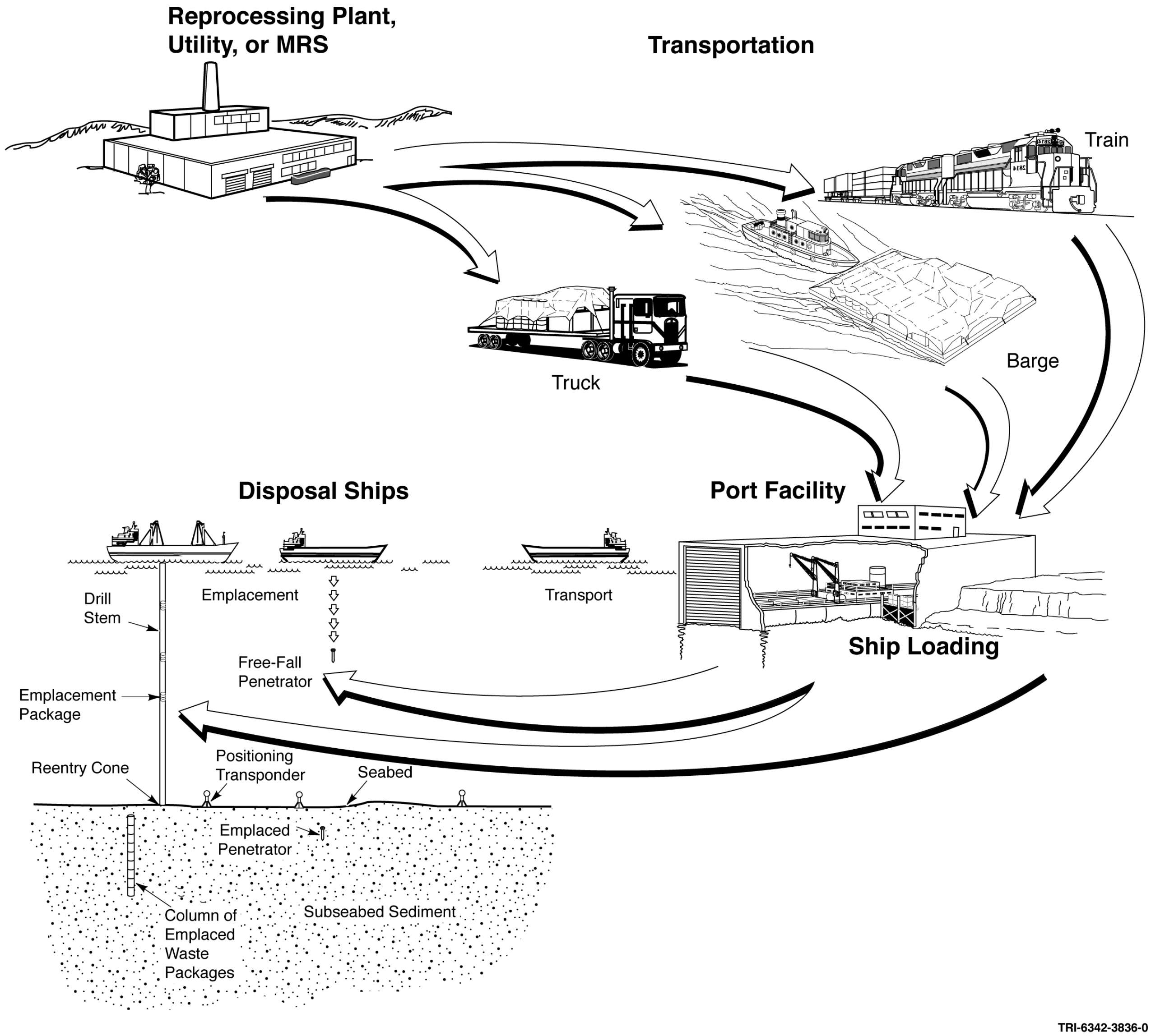


Figure 13. Subseabed disposal concept (1987)

The analytical problems posed by the SDP were first of a kind. Feasibility studies for subseabed HLW disposal required development of new evaluation standards, performance assessment procedures, and models. Annual PA iterations helped to direct and optimize both the national and the international programs. Each year, they were improved and updated as the design developed and as the models and database became more complete. First, the reference system (including transportation, emplacement, the operational repository, and possible abnormal events) was defined. Since no official radiological protection criteria exist for subseabed HLW disposal, interim evaluation standards were developed. Analytic and compartmental models were developed for sensitivity and uncertainty and repetitive risk analyses. These fast and convenient computer models were verified with numeric programs. Analyses began with parametric studies to define the sensitivity of each input parameter. These studies were followed by an attenuation factor analysis to define the effectiveness of each component, and the robustness and limitations of the system. Peak individual doses, peak annual population doses, time integrated world population doses, and biota doses were computed using reference parameters. The risks were bounded using least and most favorable input data, and a stochastic uncertainty analysis was conducted when enough data were available to define probability distributions. Probabilistic accident risks were combined with the risks for the undisturbed emplaced base case to obtain total risks for the system. Abnormal events were analyzed deterministically but not included in the probabilistic risk assessment because realistic probabilities were not yet available. Risk assessments were interactive with other SDP activities and were used in functional analyses for equipment and facility designs, as guidance for research, and in feasibility evaluation. Risk sensitivity and uncertainty analyses showed where the largest gains could be made in site selection, regulation development, radionuclide transport to man, environmental impacts, land transportation of wastes, and designs for the dock, the transport ship, and the repository, and they showed which parts of the database needed to be expanded (Klett 1997b).

2.1 Methodology Development

When the SDP began, two existing methodologies for radiological assessment were available: fault-tree (also known as event-tree) analysis or system performance analysis. Because most of the important processes involved in radioactive waste disposal are slow and continuous, not quickly developing accidents, fault-tree or event-tree analysis was considered not appropriate for the SDP. The performance assessment methodology described in the Section 1 was adopted. To a significant extent, the methodology was developed and tested through its application to subseabed disposal problems, and the general PA methodology described in Section 1 only achieved the mature form described there as a result of the SDP and the contemporaneous technical support to the NRC, described in Section 3, where PA concepts were initially identified, as well as the WIPP program, where PA practices were optimized, as described in Section 4.

The initial workshop on deep ocean basin floors and radioactive materials was hosted by SNL in Albuquerque, New Mexico, on June 4 and 5, 1973. Participants included ten scientists and engineers from SNL and four senior scientists from Woods Hole Oceanographic Institution, Scripps Institution of Oceanography, and Worcester Polytechnic Institute. The study program begun as a consequence of that workshop was designed to assess the technical feasibility and safety of to using a deep ocean basin as a repository for radioactive wastes. The documentation that resulted from that workshop (Bishop and Hollister 1973) identified essential parameters (FEPs, in the nomenclature formalized later), developed a preliminary method for preparation of scenarios from a FEP list, developed a method for assessing scenarios, and began the assembly and organization of a multidisciplinary team of scientists.

The PA methodology illustrated earlier in Figure 2 was implicit in the approach adopted by the SDP, prefiguring the formal nomenclature and structure of PA. This early approach to iterative performance assessment was described by Bishop and Hollister (1973) as follows:

> As the program progresses, possible disposal schemes will need to be assessed in the light of the developing body of information so that research can be directed toward answering technical questions relevant to a decision-making process…

> Such questioning will suggest, during the program, directions in which research should go to supply missing information; and thus a continuing review within the context of radioactive waste disposal will have to be an integral part of the overall program.

Their "review strategy" proposed to iteratively assess subseabed disposal approaches—essentially a framework for subseabed disposal PA—is shown in Figure 14 (Bishop and Hollister 1973, p. 16).

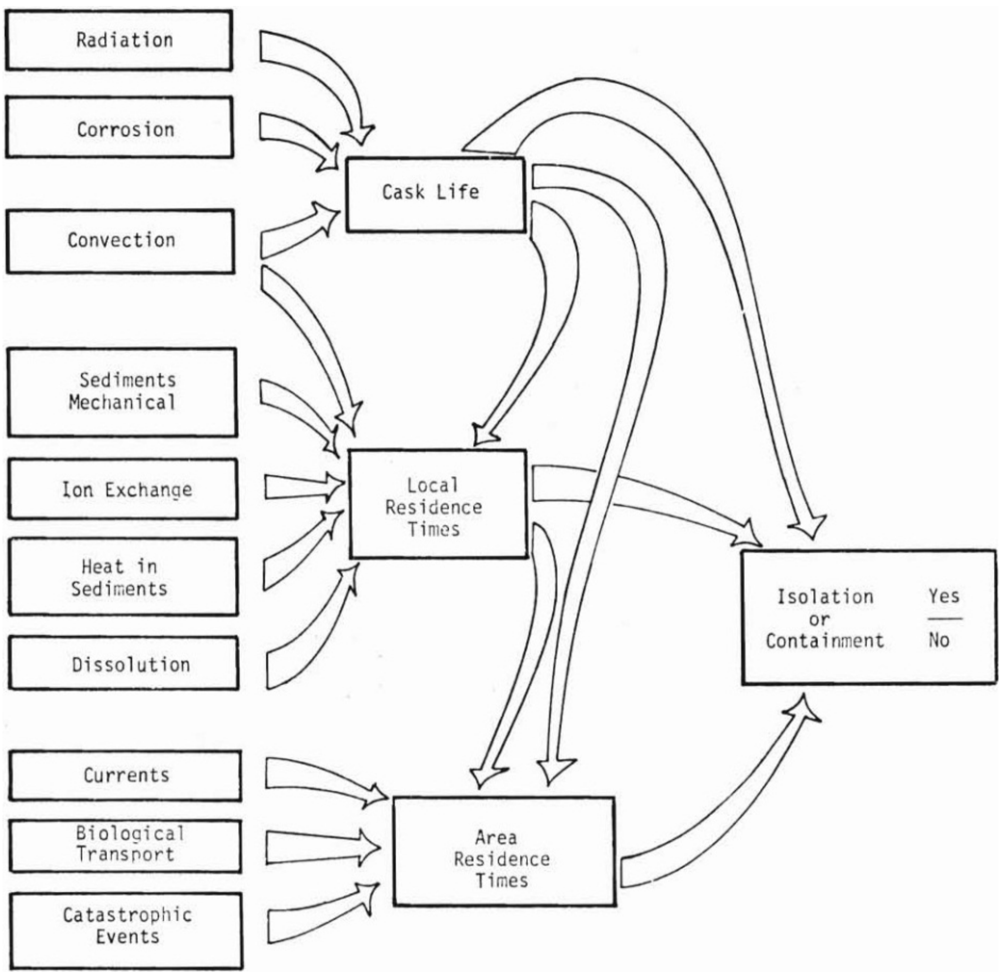


Figure 14. Initial 1973 "review strategy" for subseabed disposal, which would evolve into its performance assessment

This earliest effort also initiated a simple site selection screening process that stressed the necessity of the site to be stable, uniform over great distances, and predictable. A generic comparison of ocean basin floors to deep sea trenches, continental shelf, and submarine deltas, fans, and cones based on 19 parameters likely to be important to suitability for radioactive waste disposal indicated that the centers of ocean geologic plates and the centers of ocean water gyres are the best ocean regions to consider for nuclear waste repositories, and to be preferred over other ocean provinces.

From its outset, the program was focused on challenging its hypotheses in an effort to demonstrate that the subseabed disposal concept would *not* work—i.e., "…with an eye toward identifying any immediately obvious technical reasons for dismissing the possibility of sea disposal," as Bishop and Hollister (1973, p. 6) wrote. This philosophy continued, leading to significant program changes such as the determination in 1976 that, because it lacks the uniformity and predictability of a geologic formation, the water column was not a suitable barrier for waste isolation. The PA would also reveal several weaknesses in the reference ship design that resulted in transportation accidents contributing the highest risk to the system. This information was used in the functional analysis and functional requirements studies for redesign of the ship, and peak individual doses from shipping accidents were reduced by a factor of 1.89×10^6.

2.2 Subseabed Disposal Project PAs

Figure 15 presents the conceptual model for the PA for subseabed HLW disposal. It illustrates the radionuclide pathways and types of nuclide transport and biological computations that were used in the PA. The analyses computed concentrations in the sediment, oceans, and aquatic food; flux from the sediment; and dose rates and doses to biota, the maximally exposed individual, and the world population.

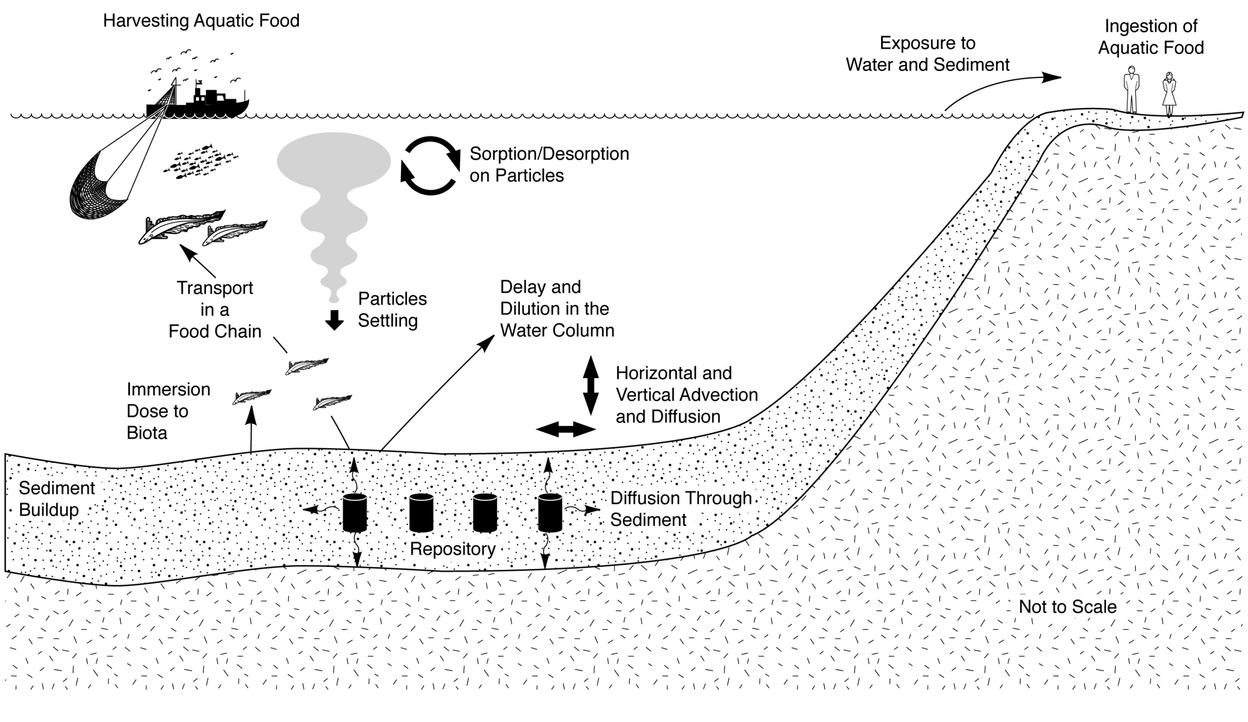


Figure 15. Subseabed disposal conceptual model for performance assessment

Beginning in 1974, the U.S. SDP (and from 1976 on with the cooperation of the international Seabed Working Group) prepared a PA iteration each year, with findings published in an annual report. The early PAs were crude because the development of the field of computer codes used to do transport calculations were just being developed by scientists at that time, necessarily by the SDP group. The calculational capabilities of the computers were very limited. SDP developed the first approach to addressing computational limitations by applying a crude sensitivity/uncertainty analysis to optimize each subsystem as it was developed in order to be able to do the overall calculations. As the SDP became more mature, more subroutines were added and thus more of the science was included.

For convenience of description in this report, the iterative SDP PAs are summarized in three parts rather than describing each annual iteration:

- First, the major developments in the PA are summarized for the period from 1974 to 1983.

- Second, the 1983 SDP PA is presented. The 1983 is selected as a key PA in that it served as the first application of the MARINRAD control code and that the results identified weaknesses in the transport ship and repository designs, and, as a result, function as a transition between the "early" SDP design and PA iterations, and the "mature" SDP design and PAs.

- The 1987 SDP PA, described third, was the last PA conducted before the U.S. and international programs were terminated, despite the promise that the PAs showed, as a result of shifts in national and international policy and an increasing focus on geologic disposal of nuclear wastes (the 1987 amendments to the Nuclear Waste Policy Act focused all further national waste management and research efforts exclusively on Yucca Mountain). The 1987 SDP PA therefore represents the most mature and complete SDP PA conducted, though further iterations would have still been necessary to demonstrate compliance with regulatory standards, though none had been developed specific for subseabed disposal.

2.2.1 PA Development 1974–1983

The first component model for a subseabed disposal repository system, representing flow of water through sediment, was developed in 1974. The first subcode for heat effects was run in 1975, followed by the first coupled model for K_ds in sediments in 1976. The need to examine and compare different models when more than one is available was recognized by 1975, when three different analytical solution models for the temperature fields (finite-difference, closed-form analytical, and closed-form approximate solutions) were compared (Talbert 1976).

As the SDP developed, the concept for the need for an overall control code became increasingly obvious. In 1978, the SDP began the design of a complete system analysis model, MARIAH, and by 1980 the MARIAH code was being applied to do sensitivity analyses to focus the program efforts, minimizing research and development and reducing the number of variables in analyses by eliminating parameters that had little or no effect on the results. Initial sensitivity studies calculated the effects on surface temperature from the waste concentration in the can and the effects on peak temperature from the age of the waste, the distance between canisters, the depth of burial, the sediment conductivity and heat capacity, and the canister radius. Though it was successfully applied in these early analyses, MARIAH was determined to be limited in its scope and was used only for heat and water transport. The lessons learned from the development of MARIAH were used to develop a new control code, MARINRAD, that was very flexible and could easily incorporate all types of subcodes, including codes for dose calculation. MARINRAD was first used in 1983, when it was applied in both the pre-emplacement and postemplacement PA calculations.

In addition to the modeling activities, the SDP research and development program for engineering conducted laboratory experiments (e.g., penetrator and hole closure testing, thermal-mechanical testing of clay sediments and near-field effects), as well as numerous field tests, launching nearly 100 penetrators into ocean sediments in support of model verification and validation, telemetry system testing, design optimization, geotechnical data collection, and hole closure investigations. Research in geoscience characterization included (in order of increasing site-specificity) review of national data archives, reconnaissance cruises, swath mapping, deep tow seismic surveys, and physical and biological oceanographic surveys. Numerous sediment cores were obtained from the mid plate–mid gyre sites under study, showing the sites to be very stable, uniform, and predictable: one core showed a continuous record of slow sediment deposition for three million years.

2.2.2 1983 Subseabed Disposal PA

The radiological assessments in the 1983 subseabed disposal feasibility PA included two general analyses:

- Pre-emplacement operations analyses, including land transportation via truck or rail car to port, storage and handling at the port facility, sea transportation to the disposal site, and emplacement operations.
- Postemplacement performance, considering the barriers and processes that contribute to isolation or eventual release and distribution of radionuclides and resulting radiation doses to humans (and, in this study, also to aquatic biota).

Pre-emplacement operations analyses—The analysis of risks from accidents during pre-emplacement operations, like the postemplacement PA, assumed a 76,000 MTHM repository for canisters of vitrified HLW, with a canister production rate of about 843 per year over a 25.3-year disposal period. The land transportation analysis included normal operations and accidents, both radiological and nonradiological effects, and exposures to works and the general public. Unit risk factors were calculated using the RADTRAN-II code (Taylor and Daniel 1982), developed by SNL initially for the NRC.

The risks calculated for land transportation of HLW would be very similar to those incurred for a land-based repository, and they would be small (less than one fatality per year). The risks from operations at a port facility are much less than from land transportation. Accidents during ship loading and emplacement operations were not considered threats to the integrity of the HLW canisters.

The pre-emplacement analyses focused on transporting HLW by ship from the port facility to the repository location. Several types of accidents (e.g., short-term immersion of undamaged canisters, fire, weather-related phenomena) were shown to be very unlikely to damage canisters. However, collision accidents and grounding accidents (or a combination) could result in radionuclide releases. The outcome of these accidents could range from a floating ship with all canisters being recoverable intact to a sunken ship with recovery of the canisters being practically impossible. Consequences of releases in the water were modeled with MARINRAD, the same model used in the postemplacement analyses, but with a compartment model developed for coastal waters used only in the pre-emplacement analyses.

The analyses showed that the probability of an accident that leads to some degree of release within the 25.3-year period of emplacement operations was very small (about six chances in 10,000). The most probable release accident leads to a peak dose of 2.3×10^{-3} rem/yr to a maximally exposed individual, approximately 2% of doses received from natural background. The most serious release scenario was shown to be the HLW transport ship sinking in coastal waters, without any recovery of the canisters. It was calculated to be extremely unlikely, about six chances in 1 million, but the resulting peak dose was 0.7 rem/yr.

Postemplacement analyses—The postemplacement analyses (Kaplan, Koplik and Klett 1984, SNL Seabed Programs Division 1983) considered subseabed disposal of vitrified HLW at two locations (one in the mid plate–mid gyre in the Pacific, and on in the Nares Abyssal Plain in the Atlantic). The 1983 PA was a deterministic consequence analysis for each site, modeled as a 76,000 MTHM repository of canisters emplaced 20 m deep in the ocean sediments. Sensitivity analyses were performed for the transport parameters. The waste inventory was based on vitrified, mixed-oxide HLW, and the analyses considered the 22 most radiotoxic nuclides in the waste. For the Subseabed Disposal PA in 1983, the initial "review strategy" concept for postemplacement analyses from 1973, illustrated in Figure 14, which guided research and program management and structured the PA itself, had evolved to the relationships illustrated in the simplified systems component model diagram given in Figure 16 (SNL Seabed Programs Division 1983, Figure 1.3), which shows how program tasks are interrelated and how the PA process was designed to lead to an assessment of scientific and environmental feasibility. Figure 17 (SNL Seabed Programs Division 1983, Figure 3.1) provides an example of the detailed coupling within the models, emphasizing the coupled models of thermally driven processes that provide input to the biological, oceanographic, transport, and radiological exposure models within MARINRAD.

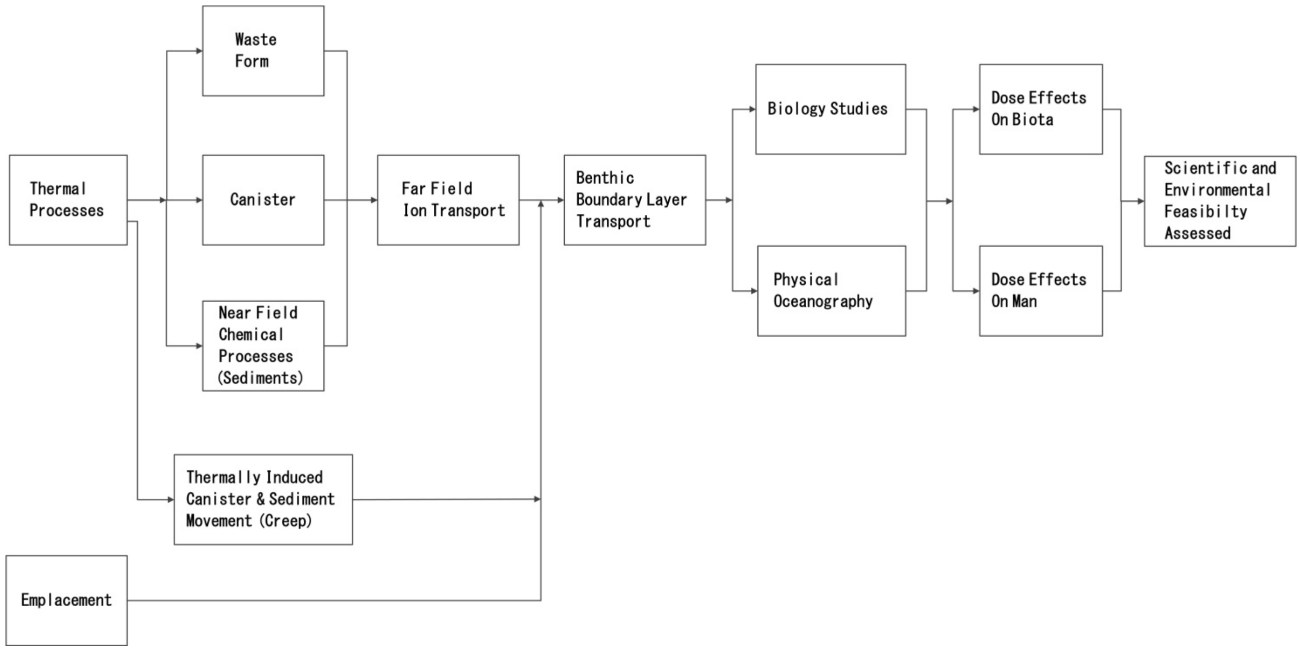


Figure 16. 1983 Subseabed Disposal Project PA simplified systems component model diagram

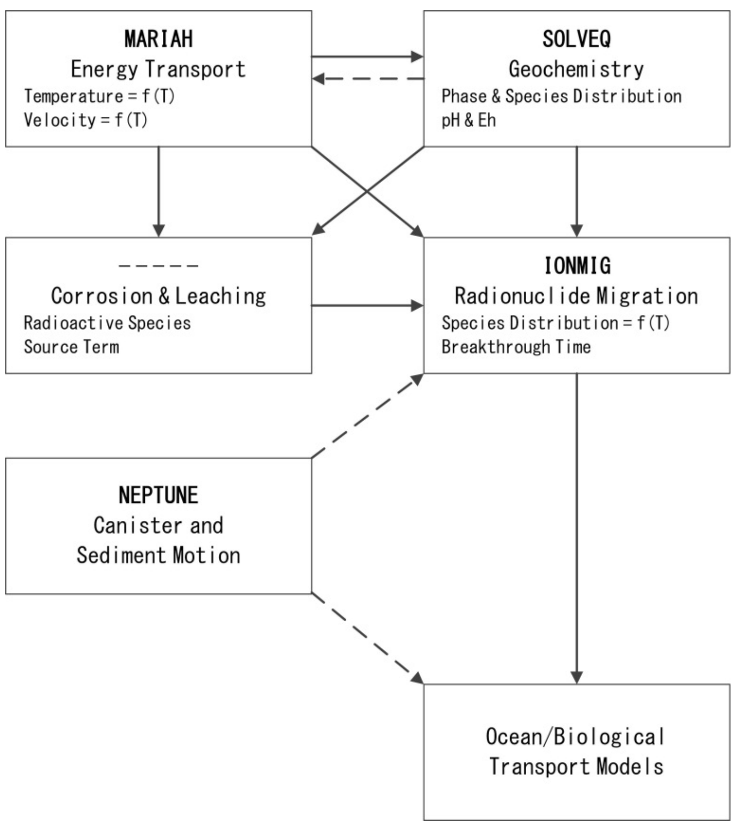


Figure 17. Relationships between thermally driven interactions considered in the 1983 Subseabed Disposal PA (and showing application of computer codes MARIAH, SOLVEQ, IONMIG, and NEPTUNE)

Lacking applicable regulatory safety criteria, the SDP adopted two current performance measures for use in the PA:

- Radiation dose to a maximally exposed individual, compared to natural background radiation levels or to generally relevant federal and international standards (in addition, similar approach was applied to aquatic biota)
- Cumulative risk in terms of the number of fatal cancers and heath affects predicted to results from releases during the first 10,000 years after emplacement, following the EPA method for its then-proposed standards for mined geologic repositories (40 CFR Part 191).

Using the EPA method allowed direct comparison to the EPA's 40 CFR Part 191 (which set limits for health risk to no more than 0.01 fatal cancers and first generation genetic defects per MTHM in 10,000 years). Applying the EPA method to subseabed disposal was recognized as very conservative, because the incremental risk for every individual affected is added together, regardless how small that individual risk might be. With ocean disposal, the potentially affected population from even a small release of radioactivity is very large (due to mixing throughout the world's oceans), even though the impact on any single individual may be extremely small. The 1983 analysis showed that, for a repository that just meets the EPA measure, when analyzed by the other performance measure, the dose to a maximally exposed individual calculated dose to the reasonably maximally exposed individual was less than 10^{-5} rem/yr, which is less than 0.01% of the dose from natural background radiation. Applying the EPA standard to a subseabed repository was thus shown to be highly conservative (SNL Seabed Programs Division 1983, p. 135).

MARINRAD was used to provide a framework for integrating the elements of a safety assessment for subseabed disposal of HLW. MARINRAD is a system of codes developed, tested, and documented by SNL (Koplik, et al. 1984) to address (1) the rate of release of radionuclides into the water column; (2) oceanographic transport of radionuclides; (3) biological transport of radionuclides; (4) exposure pathways to humans; and (5) dose, fatal cancers, and first generation genetic effects calculations (including doses to biota).

In the analysis of radionuclide releases to the water considered the effects of canister lifetime, waste form release rate, and transport through the sediment. For modeling purposes, the canisters are assumed to have a lifetime of 100 years unless they are damaged in an accident, and the waste form is assumed to leach at a fractional rate of 0.1% per year. Both assumptions were recognized to be conservative for the materials and waste forms being considered for the repository design. Transport through the seabed sediments was calculated using equations for single-boundary, surface-integrated fluxes from a line source. Because site selection criteria will rule out any sites with significant natural porewater convection, diffusion was considered the only significant means of transport. Sorption was modeled as an instantaneous, linear, reversible process.

The oceanographic transport models divided the ocean into compartments based on spatial scales and dimensions of mixing within the ocean, ranging from very small scale (representing mixing near the point of release) to ocean-basin scale (representing global ocean circulation). Ocean processes including advection, dispersion, sorption onto suspended and sea-bottom sediments were modeled by transfer coefficients between the compartments, and radionuclide concentrations were calculated by performing a mass balance on each compartment. A separate deep-water model was created for the generic northwest Pacific location and for the Atlantic location in the Nares Abyssal Plain. In addition, a generic coastal model with three smaller-scale compartments was developed to assess transportation accident risks from the sinking of a ship carrying waste to the disposal site.

The dose calculations for the 1983 subseabed disposal PA included several major exposure pathways:

- Ingestion of various aquatic biota (fish, seaweed, crustaceans, mollusks, and plankton) directly contaminated by the water or the sediment
- Ingestion of fish from a contaminated food chain
- Ingestion of contaminated sea salt and contaminated desalinated water.
- Immersion in contaminated water
- Exposure to shore sediments
- Inhalation of sea spray and shore sediments.

Any releases in the near future would be from accidents occurring in US coastal waters, so the calculations of near-future doses model the individual receptor using a diet modeled on present-day American consumption patterns. However, for releases that take place far in the future, the receptor's diet is based on the diet of a contemporary Japanese fisherman to reflect the maximum credible intake of food from the sea. Similarly, for calculating population doses, fatal cancers, and first-generation genetic effects, near-future releases are paired with present-day harvest rates, with far-future releases are paired with estimated maximum potential harvest rates. Calculations of the dose from aquatic food included concentration factors reflecting contamination both from exposure as well as from predation by feeding on prey from areas of potentially higher contamination.

The calculated peak doses to the maximally exposed individual were roughly five orders of magnitude below background radiation dose, at 1.1×10^{-6} rem/yr at the Pacific site and 5.3×10^{-7} rem/yr at the Atlantic site. At both locations, peak dose was calculated to occur at about 100,000 years, primarily coming from ^{237}Np in seaweed. Doses in the early period (before 10,000 years) were dominated by ^{98}Tc and ^{129}I coming from seaweed and fish.

In comparison to a performance standard like EPA 40 CFR Part 191 criterion, proposed at the time of this PA, the results showed a safety factor of more than 200, with only 2.9 premature deaths projected over 10,000 years. Biota living in the area would not appear to be endangered. In both locations, calculations of radiation doses to biota living at the ocean bottom indicated that the biota would receive a million times more radiation from background sources than from radionuclides escaping from the sediments at a subseabed repository, with calculated doses ranging from 10^{-6} to 10^{-5} rad/yr (SNL Seabed Programs Division 1983, p. 151).

Sensitivity studies in the 1983 PA included all major segments of the systems model for postemplacement repository performance. The analysis of the effect of burial depth on performance indicated that the safety requirements similar to EPA's proposed 40 CFR Part 191 limits would likely be met even with canisters buried at only 2 m in the sediments, and that the margin of safety rapidly increases, with releases calculated to be less than 0.1% of the EPA limit at depth of 30 m. An analysis of the distribution coefficients (K_ds), the most important sediment parameter controlling release of radionuclides from the sediment, showed that setting K_ds at 10% of their reference values resulted in peak doses three to four times higher than the base case (which is still very low, at 3.3×10^{-6} rem/yr at the Pacific site and 2.2×10^{-6} rem/yr at the Atlantic site), with the number of premature deaths increasing only slightly.

Summary—The 1983 SDP PA demonstrated the successful application of PA models for a subseabed repository for HLW. Though regulatory criteria had not been developed at that time, the preliminary analyses from this PA showed that properly emplaced waste canisters would likely meet proposed criteria for geologic repositories being developed at the time with large margins of safety. The analyses identified dominant radionuclides and pathways and areas where more site-specific data are required. After the 1983 SDP PA exercised the MARINRAD control code for the first time, Canada's SYVAC code would be used to make control code intercomparisons with MARINRAD and to carry out a series of preliminary safety assessments (Anderson 1986).

The pre-emplacement safety analysis identified the types of accidents that could lead to releases of radioactivity and calculated their probabilities, with the results showing probabilities in the range from about 10^{-6} to about 10^{-4}. The consequences for these transportation and operational accidents all were below fluctuations in natural background radiation with one important exception: the sinking of an entire transport ship in a coastal area, without recovery. The consequence of this accident was calculated to result in doses to a maximally exposed individual at seven times the natural background radiation. This indicated the importance of increasing safety of the HLW ship.

2.2.3 1987 Subseabed Disposal PA

The 1987 PA (Anderson and Murray, Feasibility of Disposal of High-Level Waste into the Seabed, Volume 1, Overview of Research and Conclusions 1988, de Marsily, et al. 1988) assumed a 100,000 MTHM repository, with wastes emplaced 50 m deep in the ocean sediments after 50 years in storage. Site selection focused on three study areas where geologic characterization was performed, one eastern Atlantic site at the Great Meteor East area in the Madeira Abyssal Plain, one western Atlantic site on the southern Nares Abyssal Plain, and one northwestern Pacific site east of the Shatsky Rise (Shephard, et al. 1988). Due to resource limitations, the PA focused on the Atlantic sites, and used the western site on the southern Nares Abyssal Plain as its base case (analyses of the eastern Atlantic site suggested that the results would be very similar). The risk analyses included:

1. A deterministic analysis of the base-case scenario with properly emplaced waste in the reference (penetrator emplacement) repository
2. Probabilistic analysis of transportation and emplacement accidents.
3. Bounding values for 1 and 2 using the most and least favorable input data.
4. Probabilistic analyses of 1 and 2 when there were enough data to define the input variable distributions.
5. Consequence analyses of abnormal scenarios.
6. Total probabilistic risk for the system.

Ocean circulation models were assembled using a nested approach from bottom sources in the deep bottom boundary layer through regional-scale open-ocean models to ocean-basin scales to describe oceanic dispersion of radionuclides (Marietta and Simmons 1988). Field studies provided physical data that was synthesized and assimilated into ocean circulation models. Simplified models for PA purposes were abstracted and focused on bottom sources in the southern Nares Abyssal Plain as a reference location for risk assessment calculations. Due to the low permeability of the oceanic sediments, the primary radionuclide-transport mechanism is diffusion, which for cations is slowed by the highly sorptive properties of the fine grains. At these depths, low temperatures reduce chemical reaction rates, except in the immediate vicinity of waste canisters, and high pressure prevents boiling.

The 1987 PA consisted of a base-case, where the repository behaves as anticipated (i.e., undisturbed by geologic forces that would expose the waste packages to the seawater and free from water movements within the sediments), a number of abnormal performance scenarios, a set of sensitivity analyses, and an attenuation factor analysis. The calculations were designed so the results could be roughly scalable on a per-MTHM unit basis, with some of the results presented in dose or releases per MTHM. The base-case assumed the penetrator emplacement method, with the 100,000 MTHM of HLW emplaced with 14,667 free-fall penetrators reaching a nominal depth of 50 m into the ocean sediments. The 12 abnormal scenarios were:

1. An undamaged penetrator lying on the seabed in deep water immediately after emplacement
2. A damaged penetrator lying on the seabed in deep water immediately after emplacement
3. A partly buried, undamaged penetrator lying only 10 m deep in the sediment, immediately after emplacement
4. A partly buried, damaged penetrator lying only 10 m deep in the sediment, immediately after emplacement
5. A properly emplaced, undamaged penetrator, with enhanced pore water velocity in sediments
6. A properly emplaced, damaged penetrator, with enhanced pore water velocity in sediments
7. A properly emplaced, undamaged penetrator, with enhanced leach rate of glass (modeled to instantly release radionuclides to sediments at the time of canister failure)
8. A properly emplaced penetrator, with enhanced corrosion rate (modeled as instantly failed or corroded)
9. A change in contamination pathway model (an alternative, hypothetical food-chain model representing a "short-circuit" of the base-case contamination pathway model)
10. A change in ocean circulation due to glaciation and climatic effects
11. An importance evaluation of sorption, with base-case conditions but assuming all K_ds were zero
12. An sensitivity analysis extending Case 11, setting K_ds at zero for three specific radionuclides, ^{99}Tc, ^{231}Pa, and ^{227}Ac.

Excepting scenarios 9, 11, and 12, which are analyses of alternative models, parameter importance, or system sensitivity, these scenarios represent the results of low-probability events and processes including poor emplacement of the penetrator, rocks hit by the penetrator, improper hole closure, sabotage or human error, glass quality control failure, human exploitation or natural erosion of the seabed, faulting of the sediments, or unexpected chemical processes such as microbial activity.

In addition, a study by the UK National Radiological Protection Board reported as part of the 1987 PA (de Marsily, et al. 1988, p. 115) examined a scenario for the option of drilling with mechanical emplacement in sediment 200 to 800 m beneath the ocean floor. Finally, even though there were no manganese nodules in the selected sites, a scenario representing doses resulting from activities related to mining of manganese nodules was evaluated, since they are commonly present in other parts of the ocean, and, in a few areas, may contain enough manganese, copper, nickel, and cobalt to potentially be of economic interest. This model considered doses to worker standing close to a conveyor belt carrying nodules irradiated from a subseabed repository.

In the base case, waste canisters were to be emplaced with free-fall penetrators reaching a nominal depth of 50 m below the sediment-water interface (Hickerson, et al. 1988). Laboratory investigations of radionuclide migration through the sub-bottom deep sea sediments included sorption and diffusion experiments that provided data for risk assessment calculations (Brush 1988). Studies of processes near the buried waste canisters included thermal effects, induced pore water flow, canister corrosion, waste form degradation, seawater/sediment interaction experiments, and modeling also provided abstractions and data to PA (Lanza 1988). Extensive field and modeling studies of deep-ocean biological processes were performed to understand the role of living organisms and the carbon cycle in dispersing radionuclides; seafood pathways to humans and impacts on the ocean ecosystems provided data and model abstractions to PA, also (Pentreath 1988).

The 1987 PA results indicated that with either emplacement method (i.e., freefall penetrators or drilled with mechanical emplacement), the waste package would protect and contain the wastes during transportation and emplacement and for 500 to 1,000 years after emplacement. For the next 2,000 years, waste would slowly be released to the sediments, which would be the primary barrier to the release of radionuclides. The small amount of waste that would eventually escape from the sediments would be dispersed and diluted by the oceans.

Risks from sea transportation with the newly designed emplacement ship were shown to be negligible compared to the risk from emplaced waste. Mean values for the probabilistic peak individual dose per MTHM from shipping were 4.4×10^{-20} Sv/yr compared to 5.3×10^{-15} Sv/yr for emplaced waste. The collective population dose was calculated to be 4.4×10^{-12} person-Sv from shipping versus 2.2×10^{-2} person-Sv from the emplaced waste in the subseabed repository (Klett 1997b).

Figure 18 (Klett 1997b, Figure 6) shows the total dose to the maximally exposed individual from emplaced waste as a function of time, along with the dose contribution from each nuclide to the maximally exposed individual from emplaced waste as a function of time, using best estimate input data. It shows that only a few nuclides—those with low distribution coefficients—significantly contribute to peak dose rates. It also shows that that the dose should be calculated out to at least 100,000 years to obtain meaningful collective doses. Similar curves were generated for each of the above analyses. The results of peak individual dose studies for emplaced waste are summarized in Table 2.

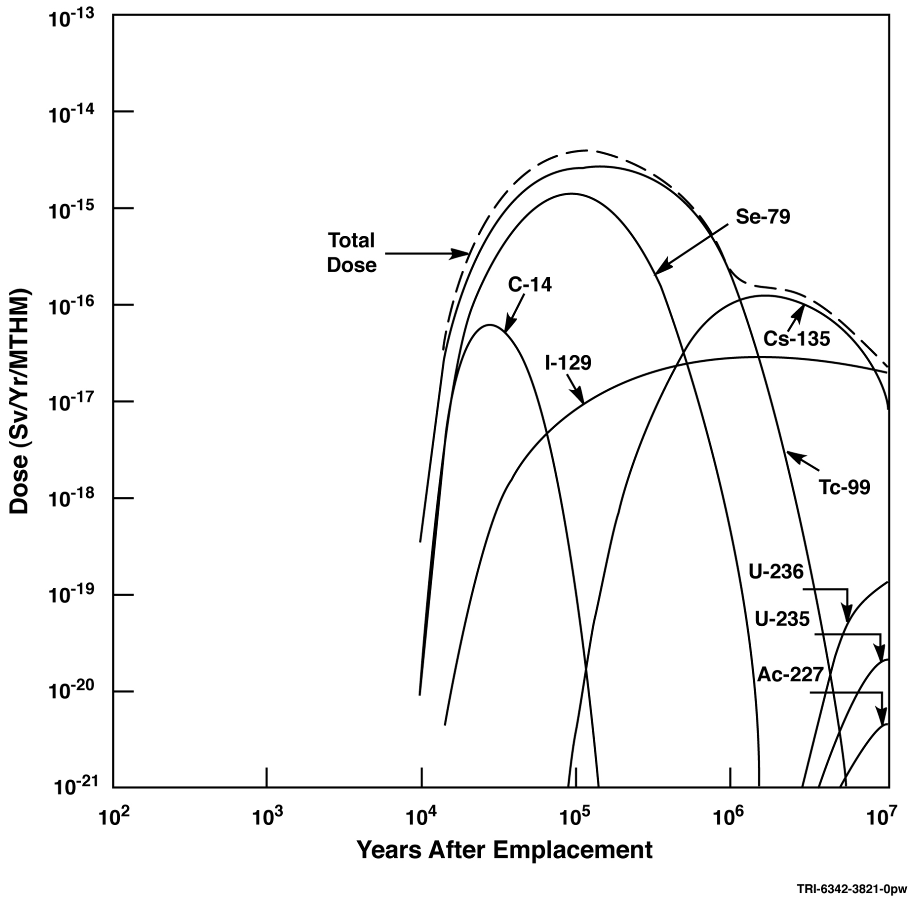


Figure 18. 1987 Subseabed Disposal Project PA results: 10-million-year individual dose histories, by radionuclide (Top N. American compartment, best estimate data, per MTHM)

Table 2. Subseabed Disposal Project PA results: peak individual dose—postemplacement release (undamaged canisters, 50-year-old waste)

Input Data	Peak Individual Dose (Sv/yr per MTHM)	Time of Peak Dose	Principal Radionuclides and % Contribution	Principal Pathways and % Contribution
Most favorable	1.0×10^{-18}	7 million years	^{129}I (91%) ^{135}Cs (9%)	Seaweed (94%) Fish (4%)
Best Estimate	5.2×10^{-15}	150,000 years	^{99}Tc (71%) ^{79}Se (29%)	Mollusk (29%) Crustacean (28%) Seaweed (24%) Fish (19%)
Least Favorable	7.3×10^{-13}	7,000 years	^{126}Sn (55%) ^{99}Tc (40%)	Fish (31%) Seaweed (30%) Crustacean (23%) Mollusk (16%)

The results of analysis of abnormal scenarios are shown in Figure 19 (de Marsily, et al. 1988, Figure 4.6), where each scenario is normalized against the base case results. Of the abnormal scenarios investigated, damaged emplaced canisters (i.e., "enhanced corrosion rate"), enhanced leach rate, and changes in ocean currents had almost no or very little effect. If all distribution and partition coefficients were zero, individual doses would increase by a factor of about 300. The greatest affect was shown by the unrealistic sensitivity case, increasing sediment vertical pore water velocity to 1 m/yr with all the canisters failed, which would result in peak individual doses increasing by six orders of magnitude. Since realistic probabilities could not be assigned to abnormal events, they were not included in the total probabilistic risk. Figure 19 also includes the peak dose results from the analysis of the emplacement option of using drilled holes with stacked, mechanical emplacement of canisters rather than the freefall penetrators; that method is estimated to reduce peak doses by three orders of magnitude compared to the base case.

Summary—The PA results indicated that subseabed would be a safe method of HLW disposal and, moreover, that such predictions based on PA could be made with a high degree of confidence. Individual doses from a subseabed HLW repository would be low compared to average individual dose levels (natural background, food, water, inhalation, and medical), the contemporary ICRP (International Commission on Radiological Protection) limit, and an assumed *de minimis* level (below regulatory concern), as shown in the results from Klett (1997b, p. 11) in Figure 20, which presents the deterministic PA results from the best estimate data as well as the results from the most and least favorable data (the curve labeled "Best Estimate" is the same as the "Total Dose" curve shown in Figure 18. The present radioactivity in the oceans is 1.9×10^{22} Bq. All the release to the ocean from a 100,000 MTHM HLW repository would increase the radioactivity in the oceans by only 0.000004%.

In 1987, the Nuclear Waste Policy Amendments Act designated Yucca Mountain as the single site for further characterization as a potential geologic repository. In addition to terminating all other site investigations for land-based geologic repository programs, further study of subseabed disposal options for radioactive waste was canceled, as well.

2.3 Significance of the Subseabed Disposal Project in the Historical Development of the PA Methodology

The SDP was initiated as a fundamentally interdisciplinary endeavor, having sparked from a conversation in 1973 between a geologist from the Woods Hole Oceanographic Institution and a Sandia Laboratories chemist—a discussion of the failure of the AEC's nuclear waste repository program at Lyons, Kansas, in the previous year (Nadis 1996). The SDP was launched quickly into a focused research program by risk-informed management approach based on an embedded PA framework and based on anticipated regulatory requirements. It serves as an early example of managing a multidisciplinary, multinational complex project using what is now called a systems engineering approach. The science and engineering program for the SDP was managed, integrated, documented, and its records and results archived using an iterative management process that is as important as the many technological and scientific advances that resulted from this PA program and others that were to follow.

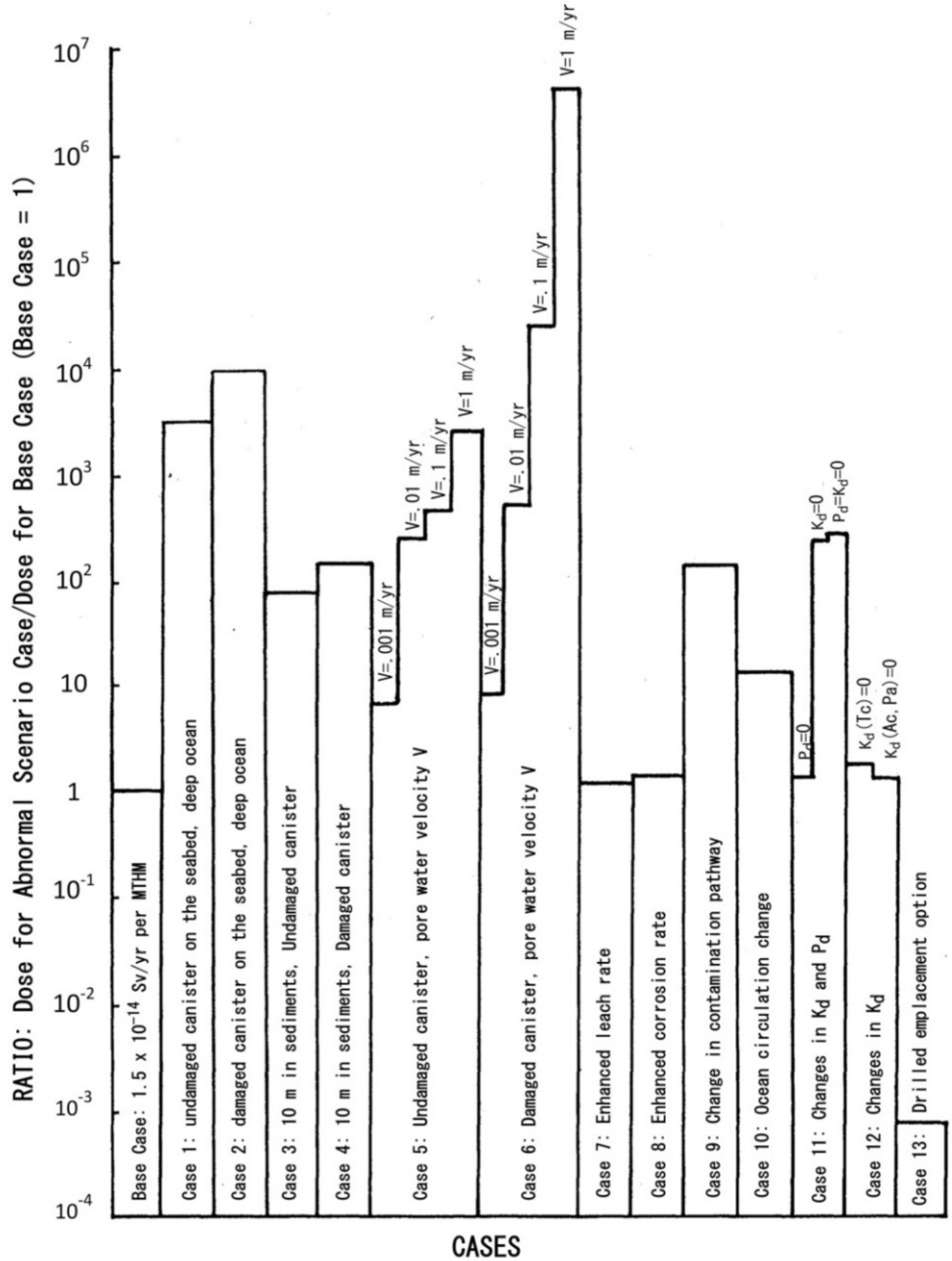


Figure 19. 1987 SDP PA: Comparison of peak dose results from abnormal scenarios and the drilled emplacement option versus peak dose results from the base case (for 1 MTHM)

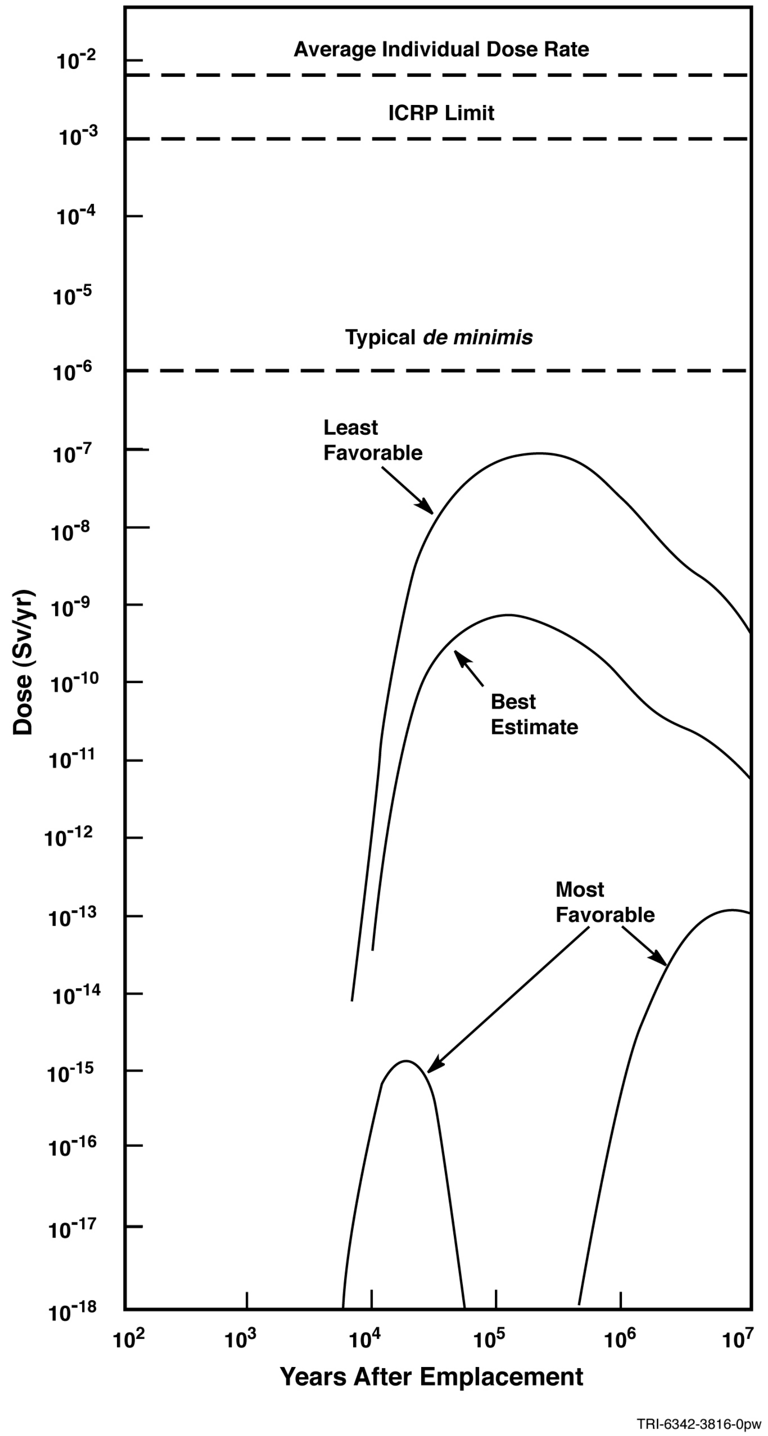


Figure 20. 1987 Subseabed Disposal Project PA results: 10-million-year dose history to the maximally exposed individual from a 100,000 MTHM repository, compared to potential regulatory levels and average individual dose rate

The SDP was instrumental in laying the foundation for the PA methodology for waste disposal, so its list of important first accomplishments is long. Among its significant contributions to PA methodology, the SDP:

- Developed the first component model, representing flow of water through sediment in 1974, followed by the first subcode for heat effects in 1975 and the first coupled model for K_ds in sediments in 1976;
- Recognized the need to examine and compare different models when more than one is available in 1975, when three different analytical solution models for the temperature fields were compared (Talbert 1976);
- Realized that if more than one code is available, the codes must be tested against each other to develop the best understanding and the identification of the best code to use;
- Developed assumed performance standards, since no regulatory criteria existed, thereby recognizing the fundamental importance of defining performance goals in the PA process and demonstrating the potential use of PA as a tool in development of regulatory standards for waste disposal;
- Identified the need for defining and including low probability events;
- Identified the need for a complete FEP list as an important first step in iterative PAs;
- Demonstrated how to develop scenarios from the identified FEPs;
- Identified the need for and demonstrated the use of a total system control code, MARINRAD, that compiled nested or coupled models, allowing iterative deterministic and probabilistic calculations;
- Developed the first approach to addressing computational limitations by applying a crude sensitivity/uncertainty analysis to optimize each subsystem model as it was developed in order to be able to perform the overall calculations;
- Adopted the Latin Hypercube sampling method to optimize the number of calculations
- Used correlation coefficients to rank the importance of parameters; and
- Demonstrated the use sensitivity analyses to identify parameters for future study.

Important or unique features of the SDP analyses included:

- Radiological assessment for pre-emplacement operations (including land transportation, port facility activities, and ocean transportations) as well as postemplacement repository performance;
- Modeling of both atmosphere and marine pathways;
- In addition to calculating risks in terms of human radiological doses, calculation of biota doses (i.e., doses to fauna) as a performance measure for potential environmental impacts; and
- Cost–benefit analyses.

The SDP showed that subseabed disposal was a very safe disposal option, with a peak mean dose of 2.8×10^{-9} Sv/yr at 125,000 years. Among important lessons learned for other waste disposal approaches, it clearly demonstrated the advantages of a uniform and predictable geology in making the calculations transparent.

3. NRC LICENSING AND REGULATORY SUPPORT (1976–1993)

3.1 Background

In 1974, SNL participated in a 60-member team led by Massachusetts Institute of Technology Professor Norman Rasmussen that evaluated the potential health risks associated with accidents from a commercial nuclear power plant. This work led to the publication of the Reactor Safety Study (WASH-1400) in 1975 (NRC 1975), which set the stage for the probabilistic risk methodology used to evaluate nuclear power plants. By the late 1970s and early 1980s, SNL was advocating a probabilistic approach to the modeling of geologic waste repositories (Campbell, Dillon, et al. 1978, Runkle, Cranwell and Johnson 1981, Cranwell, Campbell and Helton, et al. 1987), which was influenced by SNL's investigations and PA for the SDP as well as the involvement on WASH-1400 (NRC 1975) and ongoing consequence analyses that SNL developed for the NRC, specifically CRAC-II (Aldrich, et al. 1982) and NUREG-1150 (NRC 1990). A timeline of the SNL support to NRC, shown against the backdrop of other contemporary developments in PA, is shown in Figure 21.

3.2 Regulatory Development

In 1976, under the Reactor Safety Study Method Application Program, NRC funded SNL to apply event tree methodology to Calvert Cliffs-2, Grand Gulf-1, Sequoyah-1, and Oconee-3 nuclear power plants but did not include funding for any new consequence modeling (Rechard 1999a, Figure 6). Also in 1976, NRC funded SNL to develop a probabilistic PA methodology for deep geologic repositories that could demonstrate compliance with the requirements contained in the proposed NRC and EPA regulations, 10 CFR Part 60 and 40 CFR Part 191, respectively. In 1978, NRC funded Sandia to work on probabilistic PA development and apply it to a hypothetical bedded salt repository, which resulted in abandoning the fault tree methodology and using simple event trees (Rechard 1999a).

Then in November of the same year, EPA published "Criteria for Radioactive Wastes" as guidance and sought comments. Later, in March 1981, EPA withdrew their proposed "Criteria for Radioactive Wastes" because it considered the implementation of generic disposal guidance too complex given the many different types of radioactive waste (46 FR 17567). Subsequently, based on a working draft of EPA's regulation for 40 CFR Part 191, the SNL Fuel Cycle Risk Analysis Division prepared a six-volume report, *Technical Assistance for Regulatory Development: Review and Evaluation of the Draft EPA Standard 40CFR191 for Disposal of High-Level Waste* (Ortiz and Wahi 1983, Pepping, Chu and Siegel 1983a, 1983b, Siegel and Chu 1983, Helton 1983, Runkle 1983), that analyzed the draft EPA standard to provide the NRC information for use in evaluating the rationale for the technical requirements in the proposed Rule 10 CFR 60, to respond to public comments on the proposed rule, and to analyze the benefits of alternative criteria for the final rule. A series of parametric analyses were performed on the potential releases of radionuclides to the accessible environment in order to determine the impact on compliance with the draft EPA standard.

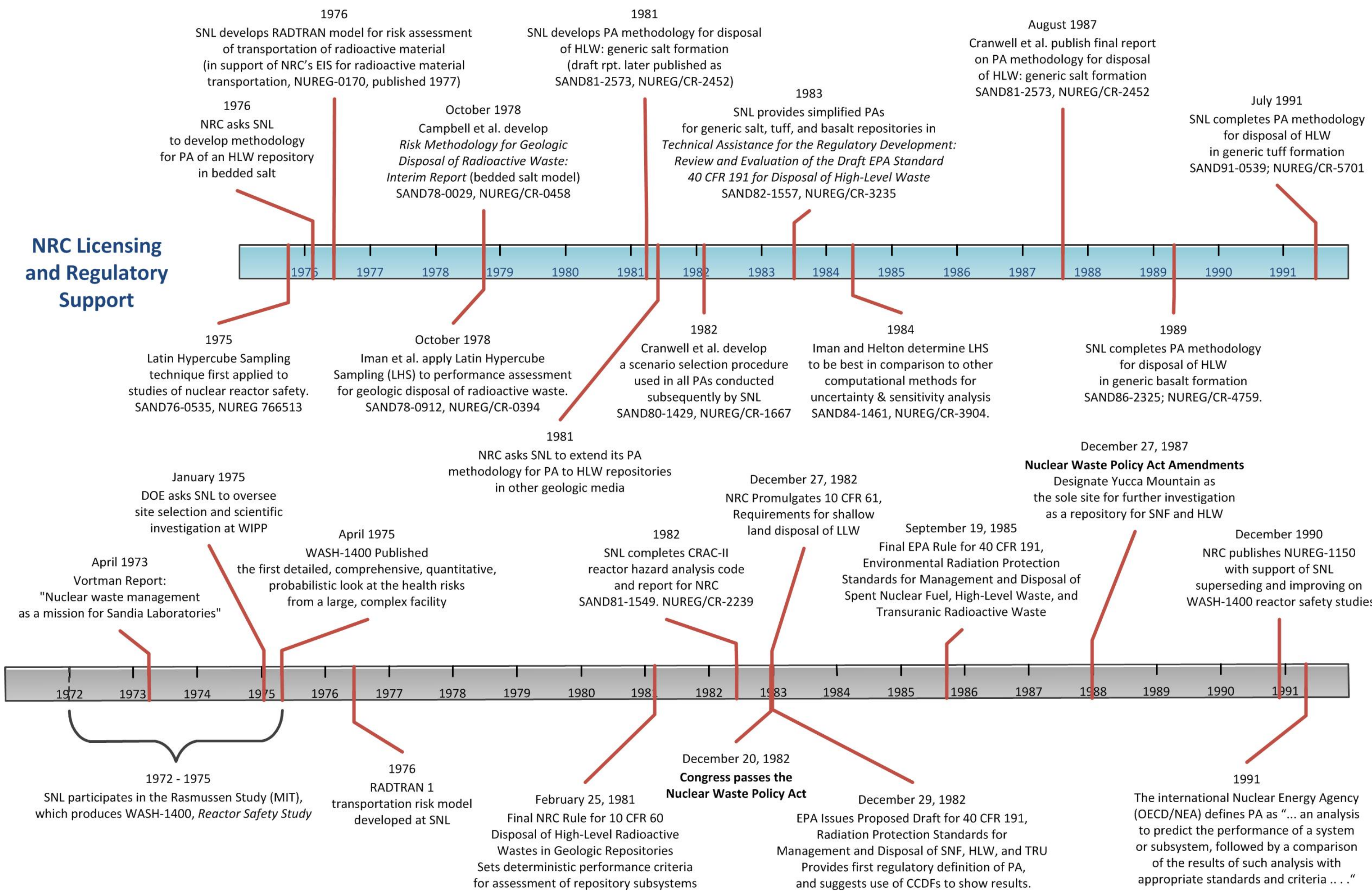


Figure 21. NRC PA support timeline

In 1981, the NRC initiated a program at SNL with the primary objective of modifying and extending the bedded-salt methodology to alternative geologic media including basalt, welded tuff, granite, and domed salt. At that time, these other geologic formations were also being considered as potential host formations for an HLW repository in the U.S. The analyses were performed for three geologic media: basalt, bedded salt, and tuff, which are described in the next section of this report. These three examples identified issues that needed to be addressed in the draft regulations, effectively aiding in the development of those regulations. (Before analyses of granite and domed salt were conducted, the number of potential host formations under consideration was reduced to three sites in bedded salt, basalt, and tuff and—still later—to one: the welded tuff site at Yucca Mountain.)

In 1982, Congress passed the Nuclear Waste Policy Act (NWPA) (Public Law 97-425), requiring the U.S. Environmental Protection Agency (EPA) to promulgate environmental standards for the management, storage, and disposal of SNF, HLW, and TRU waste. The NWPA also required the NRC to implement environmental standards set by EPA as part of the repository licensing process. EPA promulgated 40 CFR Part 191, Environmental Radiation Protection Standards for Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes, in 1985, setting probabilistic criteria indirectly based on population health risk. The EPA standard required that results be expressed in terms of probabilities of release and integrated radionuclide releases of radioactivity to the accessible environment over 10,000 years following closure of the repository. Uncertainty in the cumulative normalized release was displayed as a complementary cumulative distribution function (CCDF) for comparison to limits defined in 40 CFR Part 191. (A CCDF indicates the probability of exceeding various levels of cumulative release.)

In 1983, NRC promulgated 10 CFR Part 60, Disposal of High-Level Radioactive Wastes in Geologic Repositories, prescribing the rules governing the licensing of geologic repositories. The requirements in 10 CFR Part 60.112 set an overall system performance objective that amounted to meeting the EPA's containment requirements in 40 CFR Part 191.13, while other sections set forth subsystem performance objectives.

In 1986, Hunter, Cranwell, and Chu (1986) built upon the work of done for the NRC in Technical Assistance for Regulatory Development: Review and Evaluation of the Draft EPA Standard 40CFR191 for Disposal of High-Level Waste and provided an overview of the techniques that could be used to determine whether a repository would meet the containment requirements in 40 CFR Part 191.13. This work divided a PA into four main parts: scenario development and screening, consequence assessment, sensitivity and uncertainty analysis, and regulatory-compliance assessment. Hunter, Cranwell, and Chu (1986) used a hypothetical repository set in basalt of the Columbia Plateau as an example for the PA implementation due to the wealth of data and information that existed for basalt, discussed in the next section.

Later, in 1990, Bonano and Wahi (1990) incorporated recent advances in the PA methodology and described the role of PA in assessing compliance with 40 CFR 191.13. By this time, the PA methodology had expanded to include five main components: system description, scenario selection and screening, consequence analysis, uncertainty analysis, and sensitivity analysis.

3.3 PA Methodology Advancement

Concurrent with the generic PA evaluations and regulatory development being conducted for the NRC, SNL prepared several studies for the NRC that advanced the general state of the PA methodology and provided guidance on its general implementation. The PA process illustrated previously in Figure 6 was described by multiple SNL researchers including Cranwell, Campbell, et al. (1987) and Davis, Price, et al. (1990). Several SNL researchers focused on specific steps in the PA process, such as FEPs (Bingham and Barr 1979)[4] and scenario development (Bingham and Barr 1979, Cranwell, Guzowski, et al. 1990); FEP probabilities (Hunter and Mann 1989, Apostolakis, et al. 1991), treatment of uncertainty (Bonano and Cranwell 1988, Cranwell 1985, Davis, Bonano, et al. 1990), including data and parameter uncertainties (Zimmerman, et al. 1990) and conceptual model uncertainties (Bonano, Davis and Cranwell 1988); uncertainty and sensitivity analysis techniques (Iman and Helton 1985), including LHS sampling (Iman, Davenport and Zeigler 1980) and various techniques for sensitivity analyses (Siegel, Leigh, et al. 1989, Iman and Conover 1980), model validation (Davis, Olague and Goodrich 1991), and expert elicitation (Bonano, Hora, et al. 1990).

FEPs and scenario development—Some of the earliest work related to FEPs and scenario development was conducted by Bingham and Barr (1979) for the WIPP EIS and Cranwell, Guzowski et al. (1990), whose work on a scenario selection procedure as part of the NRC risk assessment methodology was initially submitted to NRC in 1981, and published later. Both studies focused on bedded salt as the medium in which to develop and test their methodology. Cranwell, Guzowski et al. (1990) developed the procedure for the development of scenarios described in Section 3.4.1.

Predictability of low-probability events—Hunter and Mann (1989) reviewed the literature on techniques for predicting the probabilities of events and processes for geologic repositories in five areas: human intrusion, climate change, tectonics, seismic hazard assessment, and volcanology. Building upon this information, Apostolakis et al. (1991) demonstrated a method for estimating the probability of human intrusion, climate change, and tectonics, based on decision theory, which involves Bayesian probability techniques.

Treatment of uncertainty—Implementation of uncertainty in PA has been described by Bonano and Cranwell (1988), Cranwell (1985), Davis, Bonano et al. (1990), Zimmerman et al. (1990), and Bonano, Davis, and Cranwell et al. (1988). The effect of uncertainties propagates throughout the PA analysis. Three major sources of uncertainty exist in PA: uncertainty in future state of the disposal system (scenario uncertainty), modeling uncertainty (e.g., conceptual model uncertainty, mathematical model uncertainty, and computer code uncertainty), and parameter and data uncertainty.

Bonano and Cranwell (1988) described the sources of uncertainty in scenario development: uncertainty associated with the "completeness" of scenarios, uncertainty associated with the probability of occurrence of a scenario, and uncertainty associated with the estimation of the

[4] The Bingham and Barr study was conducted in support of the EIS for WIPP. It was contemporaneous to but not part of the program for development of the SNL/NRC risk analysis methodology.

consequences of scenarios, and they described techniques for quantitatively estimating the probability of occurrence of scenarios:

- Axiomatic: the event or process is represented by a probability model; available data are used as input to the model; probabilities are assessed based on the output of the model.
- Frequentist: data on the event or process are examined for frequency patterns; probabilities are assessed based on the frequency of the data; experiments may be used to obtain the data.
- Modeling: conceptual and mathematical models are developed; repeated simulations of the mathematical model are preformed; probabilities are assessed based on the outcome of the simulations.

Techniques for uncertainty and sensitivity analysis—The Latin hypercube sampling (LHS) method was developed in 1975 by W.J. Conover for Los Alamos National Laboratory as a method of improving the computational efficiency of Monte Carlo sampling and then, in the same year, applied by SNL to computer modeling approaches being developed for the NRC nuclear reactor safety analyses (Steck, Iman and Dahlgren 1976). Software for LHS was developed by R.L. Iman in 1975 and published by SNL in 1980 (Iman, Davenport and Zeigler 1980) and subsequently revised by Iman and Shortencarier (1984), under an SNL contract to NRC. It was also under NRC contact that Iman, Helton, and Campbell (1978), first outlined the application of the LHS method to analyses of geologic repositories for radioactive waste.

Iman and Helton (1985) performed a comparison study of several widely used computational techniques for uncertainty and sensitivity analysis by examining three models (two used in radioactive waste repository PAs, and one in reactor accident assessments) having large uncertainties and varying degrees of complexity. The technique using LHS and regression analysis gave the best overall results. Zimmerman et al. (1990) provided a thorough review of techniques for predicting data and parameter uncertainties in HLW PA models. Four categories of uncertainty analysis methods were described: (1) Monte Carlo simulation, (2) replacement models (response surface techniques), (3) differential techniques (direct, adjoint, and Green's function technique), and (4) geostatistical techniques (stochastic modeling using Monte Carlo simulation and spectral analysis). The advantages, disadvantages, and applications of each technique were presented, as well as propagation of those uncertainties through multiple, linked models and application of those techniques to sensitivity analysis.

On the basis of its computational efficiencies, confirmed by comparative studies of alternative methods, the LHS approach was used at SNL for performance assessments of hypothetical disposal systems in bedded salt (Cranwell, Campbell and Helton, et al. 1987) and basalt sites (Bonano, Davis and Cranwell 1988), developed for the NRC, as described later. LHS has been adopted for use for all SNL probabilistic PAs of radioactive waste disposal, including the WIPP and YMP PA models and the GCD and INL HLW PAs, as well (as described in Sections 4, 5, 6, and 7).

Expert elicitation—Expert elicitation has been proposed to address uncertainties in scenarios and input parameters. Early descriptions of techniques and implementation of expert elicitation in PA were described by several SNL researchers including Bonano and Cranwell (1988) and

Bonano, Hora et al. (1990). Bonano and Cranwell (1988) described the use of expert elicitation in defining unknown data and parameter values, and suggested the following approach:

1. Identification of areas in which expert opinion is needed or recommended in uncertainty analysis.
2. Identification and screening of important issues to be considered by experts in each of these areas.
3. Compilation of available techniques for elicitation and use of expert opinion that are appropriate for identified issues.
4. Classification of issues according to acceptable elicitation techniques that are recommended for each category (e.g., single expert vs. multiple experts).
5. Identification of areas for which decomposition is likely to be more useful than direct assessment of the complete problem.
6. Elicitation and use of expert opinion to address the issues identified above.

Bonano, Hora et al. (1990) provided a much more detailed analysis of the role of expert judgment in PA and described the process for the formal elicitation and communication of expert judgment and its potential application in PAs for HLW geologic repositories.

Uncertainty also exists in the codes and models used in HLW PAs. This uncertainty is introduced due to uncertainty in the theoretical description of the process being modeled, coding errors, and errors in numerical algorithms used in the computer code. Validation can provide confidence in the ability of the model to adequately describe the system, and Davis, Olague, and Goodrich (1991) provided general approaches and concepts that can be applied in validation of models used in PA of HLW repositories.

Software code development—As part of the development of a risk assessment methodology for NRC, SNL developed important software codes that would be broadly applied in analyses of geologic repositories or serve, later, in the development of other important codes. Among them, the series of groundwater flow and transport codes developed by SNL was particularly important to regulatory development. The codes developed in support of the SNL/NRC PA methodology program included DNET (Cranwell, Campbell and Stuckwisch 1982), a quasi-two-dimensional network code that was developed primarily for investigating the combined near-field effects of thermal expansion, subsidence, salt dissolution, and salt creep; Pathways (Helton and Kaestner 1981) three groundwater flow and transport codes:

- The Sandia Waste Isolation Flow and Transport (SWIFT) code, developed and described by Dillon, Lantz, and Pahwa (1978) and Reeves and Cranwell (1981),
- The Network Flow and Transport (NWFT) code (Campbell, Kaestner, et al. 1980)
- The Network Flow and Transport code with Distributed Velocity Method option (NWFT/DVM) (Campbell, Longsine and Cranwell 1981).

The SWIFT code, the first groundwater modeling code developed as part of the PA methodology program, was a very flexible, coupled, transient, finite-difference model in one, two, or three dimensions that was used to model regional hydrology of the reference site. NWFT was developed to provide a simpler, more efficient code to modeling groundwater flow and transport. It models the flow system as a network of one-dimensional segments. Fluid discharge and velocity are determined by requiring conservation of mass at the segment junctions. Once the flow system is established, the radionuclide migration pathway from the repository to the discharge point is determined. Radionuclide discharge is then calculated from an analytic solution by assuming that transport occurs along a single, one-dimensional path having length equal to the total migration path length and using the average isotope velocity. NWFT/DVM incorporated the Distributed Velocity Method, removing some important limitations of the original code. NWFT/DVM provided considerable capability for groundwater flow and transport calculations in sensitivity and risk analysis applications, but was best used in conjunction with a code such as SWIFT that provides a realistic description of the fluid flow field to determine the radionuclide migration path for any particular repository breach scenario. NWFT/DVM could then be applied to reproduce the migration path and to evaluate the effects on radionuclide discharge of variables that alter the radionuclide source rate or migration time.

Code development continued in the ongoing development of PA models of repositories in various media. SWIFT II (Reeves, Ward, et al. 1986) was developed as part of modeling the basalt repository and later applied to the tuff repository PA; it updated the SWIFT model to add the capability to model flow in transport in fractured media. NWFT/DVM evolved into the NEFTRAN model (Longsine, Bonano and Harlan 1987) and, later, NEFTRAN II (Olague, et al. 1991) and NEFTRAN-S (Campbell, Leigh and Longsine 1991), which were developed in parallel. The codes have evolved to retain the capabilities of its predecessor and add new features to enhance the modeling capability. The NEFTRAN code expanded the capability to simulate transport through saturated, dual-porosity fields or fractured media. NEFTRAN-II included a piecewise-steady-state option allowing for transient variations in flow and transport conditions that was not included in NEFTRAN-S. The NEFTRAN-S version (developed for EPA, not under contract to NRC) further enhanced the NEFTRAN code capability by including probabilistic analysis of radionuclide transport.

In addition to supporting NRC's regulatory risk analysis capabilities, the NWFT/DVM code was used by EPA to support the 1985 promulgation of 40 CFR Part 191, and NEFTRAN-S was used to support EPA's amendments to 40 CFR Part 191 in 1993 (EPA 1993). Both the SWIFT and NEFTRAN code series were applied in subsequent SNL PAs.

3.4 PAs for Generic HLW Repositories

Beginning in 1976 and continuing until 1991, in support of the NRC, SNL evaluated three geologic media for the deep geologic disposal of HLW: bedded salt, basalt, and welded tuff.

SNL's initial PA work was applied to a hypothetical HLW repository in a generic bedded salt formation (Campbell, Dillon, et al. 1978, Cranwell, Campbell and Helton, et al. 1987, Pepping, Chu and Siegel 1983b, Cranwell, Guzowski, et al. 1990). As SNL's PA analyses for bedded salt progressed, SNL began to investigate whether the PA could be applied to the analysis of a HLW repository in other geologic media, specifically basalt (Pepping, Chu and Siegel 1983a, Guzowski and Cranwell 1983, Hunter 1983, Bonano, Davis and Shipers, et al. 1989) and welded tuff (Siegel and Chu 1983, Gibbons and Guzowski 1989, Parsons, Olague and Gallegos 1991, Gallegos 1991). As a result of this work, SNL successfully demonstrated that the PA methodology was independent of geologic media and could be used by NRC to examine compliance with the regulatory requirements in 10 CFR Part 60.

3.4.1 HLW Disposal in Generic Salt Repository

In 1976, during the same time period that the Subseabed Program was getting underway, the NRC instituted a program with SNL to develop a comprehensive PA methodology for the evaluation of deep geologic disposal of HLW in bedded salt formations. The ensuing methodology followed the same steps outlined in Figure 6. Models for groundwater flow and radionuclide transport through bedded salt (assumed to be a saturated, porous medium) were developed, as well as models for biosphere radionuclide transport, and dosimetry and health effects. Techniques for sensitivity and uncertainty analyses were also developed under that program.

The first study completed in 1978 (Campbell, Dillon, et al. 1978), an interim report on the development of a comprehensive methodology for risk assessment of geologic disposal of radioactive wastes, outlined a preliminary risk methodology for the deep geologic disposal of wastes using a generic bedded salt as the host medium. This study considered HLW, intermediate-level wastes, cladding wastes and TRU waste for storage in the reference repository. Twenty events and processes that could influence the stability of the disposal system were identified, including natural events such as faulting and subsidence; inadvertent intrusion such as by drilling, mining, or hydrofracture, other man-made disturbances such as hydrologic stresses from irrigation and dams; and thermal, chemical or other physical effects. Waste release modes were identified, probabilities of release were calculated, and failure rates were calculated for the events and processes identified. Conceptual models (shown in Figure 22) and mathematical models were constructed, and, finally, the study calculated transport of radionuclides by groundwater, transport to man, and dosimetry and health effects. However, this first study did not include sensitivity analysis.

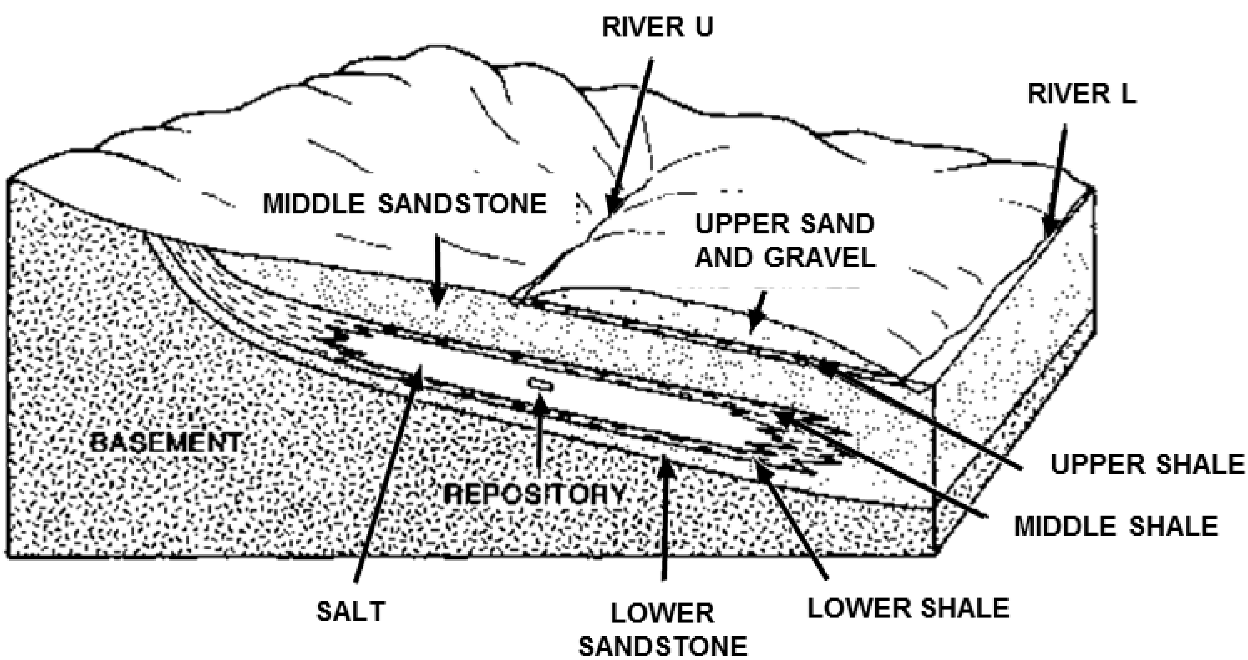


Figure 22. Conceptual model for a generic HLW repository in a geologic salt formation

In 1981, Cranwell, Guzowski, et al. (1990) developed a procedure for the development of scenarios and applied the procedure to a hypothetical repository in bedded salt. (This report was submitted to NRC in 1981, but NRC did not publish it until 1990.) The scenario selection procedure, which is still used today, consists of the following steps:

1. Comprehensive identification of events and processes important to long-term waste isolation,
2. Classification of events and processes to aid in completeness arguments,
3. Screening events and processes based on well-defined criteria,
4. Formation of scenarios using specific combinations of events and processes remaining after screening,
5. Screening of scenarios, and
6. Selection of a final set of scenarios for use in evaluating a potential disposal site.

Screening criteria for events and processes was based on physical reasonableness, probability of significant release of radionuclides, and potential consequences (Cranwell, Guzowski, et al. 1990). Only 12 scenarios remained for the hypothetical bedded salt repository after screening based on physical reasonableness and consequence. The scenario selection procedure developed by Cranwell, Guzowski, et al. (1990) has been used for all subsequent PAs conducted by SNL.

Also in 1981, continuing the work begun in the Interim Report by Campbell et al. (1978), Cranwell, Campell, and Helton et al. (1987) submitted the draft of the Final Report documenting the SNL PA methodology for geologic disposal of radioactive waste. (Though it was submitted in 1981, the NRC did not publish the Final Report until 1987.) This methodology report included a PA of a generic salt repository, applying more fully developed models to evaluate the regional hydrology, waste–host rock interactions, groundwater flow and solute transport, biosphere radionuclide transport, and dosimetry and health effects. This demonstration included the evaluation of the 12 different scenarios identified previously by Cranwell, Guzowski, et al. (1990), including an undisturbed (or base-case) scenario, as well as multiple disturbed repository scenarios involving various combinations of boreholes and groundwater withdrawal wells. Scenario screening was performed to reduce the number of scenarios to a manageable number. Random samples of uncertain parameters were obtained from their respective probability density functions (PDFs) using LHS sampling, and input vectors were constructed for each scenario. The results violated the draft EPA standard, and owing to the structure of the calculation it was possible to identify the individual vectors that caused the limit to be exceeded (Cranwell, Campbell and Helton, et al. 1987).

In 1983, SNL specifically evaluated the total integrated discharges that could be expected from “man-induced” disruptions in accordance with the limits described in a draft version of EPA’s 40 CFR Part 191 (Pepping, Chu and Siegel 1983b). The reference site for the 1983 PA was an unnamed bedded salt formation in a similar geologic setting. The repository was assumed to be located at a depth of approximately 700 m (2,300 ft) and the waste consisted of 86,000 MTHM. Six scenarios were developed, including four “U-tube” drilling scenarios, one direct canister hit, and one brine pocket penetration. Drilling was assumed to be a Poisson process. Three source-term assumptions were used for each of these scenarios. Uncertain parameters were sampled using LHS for the flow and transport calculations. The results indicated slight violations of the draft EPA standard. The assessment identified several very important issues: (1) brine pockets could pose a significant problem, (2) choice of source model was very important, and (3) assigning reliable numerical values to the scenario probabilities was difficult.

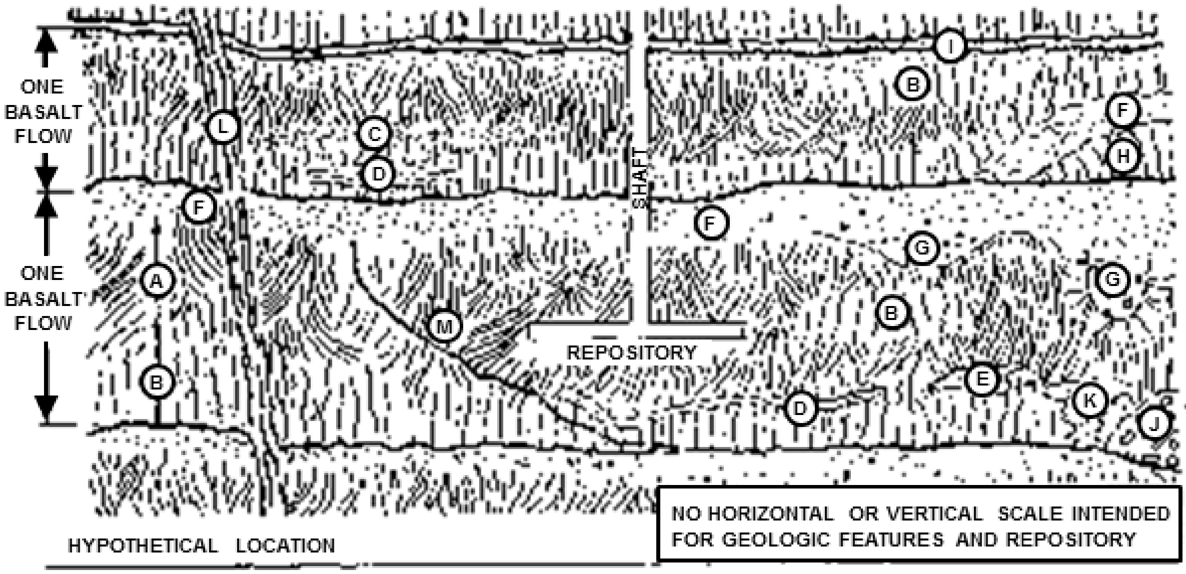
The bedded salt PAs successfully demonstrated the probabilistic risk assessment methodology for a deep geologic repository and its use in assessing compliance with the NRC and EPA regulations and advanced the art of scenario development and screening.

3.4.2 HLW Disposal in Generic Basalt Repository

In 1981, the NRC initiated a program with SNL with the primary objective of modifying and extending the bedded-salt methodology to alternative geologic media. The structure of the SNL PA methodology remained the same with the main difference being the conceptualization of groundwater flow and transport. Bedded salt was assumed to behave as a porous medium whereas basalt, which is dominated by fractures, required a different set of flow and transport models and computer codes to assess consequences.

The first basalt analysis was completed in 1983 (Pepping, Chu and Siegel 1983a), and the results were compared to the draft EPA standard; however, at that time, “release” was not defined in the draft standard and was left to interpretation. Questions remained as to whether “release” applied to a unique scenario or to all events or processes that may result in discharges to the environment. This first PA set out to evaluate the implications of alternative interpretations of the draft EPA standard.

In the conceptual model for the first basalt study, shown in Figure 23, the reference site (Pepping, Chu and Siegel 1983a) was an unnamed saturated fractured basalt formation with mountains on two sides, encompassing most of the southwestern side and part of the northeastern side, as well as a major river encompassing the entire northeastern side of the reference site. The geology and hydrogeology consisted of a shallow unconfined aquifer, several basalt flows, and intermittent interbeds. The repository was placed at a depth of 90 meters (300 feet), and was overlain by a dense basalt layer. The waste consisted of spent fuel from 46,800 MTHM. The canisters were assumed to have a life of 1,000 years. The analysis considered radionuclide retardation in fractures where secondary mineralization had occurred. The radionuclide solubilities were based on those from bedded salt, which was considered conservative, and matrix diffusion was conservatively assumed to be negligible (Pepping, Chu and Siegel 1983a).



FLOW INTERIOR DISCONTINUITIES

A ENTABLATURE JOINTS
B COLONNADE JOINTS
C VESICULAR ZONE
D PLATY ZONE
E LOCAL FRACTURED ZONE

FLOW CONTACT

F FLOW TOP
G LOCAL THICKENING OF FLOW-TOP BRECCIA
H FLOW TERMINATION
I SEDIMENTARY INTERBED
J PILLOW BRECCIA
K SPIRACLE OR SPIRACLE-LIKE FEATURE

BEDROCK STRUCTURAL DISCONTINUITIES

L FAULT OR FRACTURE ZONE, HINGE OF FOLD, OR SHEAR ZONE
M LOCALIZED TECTONIC FRACTURE

Figure 23. Conceptual model for a generic HLW repository in a basalt formation

Three scenarios were developed including an undisturbed (base case) and two disruptive scenarios, one with repository-induced fracturing of the overlying dense basalt and one with an intrusion borehole. The intrusion scenario also examined different source term models: a mixing cell source model and two different leach-limited source models. Groundwater transport was calculated using a quasi–2-D model, which obeyed Darcy's Law. Large uncertainties associated with some input variables were assumed to be distributed according to user-specified probability distributions and sampled using LHS (Pepping, Chu and Siegel 1983a).

There were several violations of the draft standard; however, the results of the analyses were not the important conclusion, but rather the issues that it raised, such as two possible interpretations of the draft EPA standard that could significantly influence a regulatory compliance decision. The importance of the source term assumption was demonstrated through use of the different source term models in the intrusion scenario. In addition, it was recommended that future analyses address sampling error (Pepping, Chu and Siegel 1983a).

In 1989, SNL published a more detailed demonstration of the 1983 basalt PA (Bonano, Davis and Shipers, et al. 1989). The 1989 basalt PA utilized a comprehensive database of geologic, hydrologic, thermal-mechanical, and geochemical data for basalt deposits in the Pasco Basin in the State of Washington (Guzowski and Cranwell 1983) and a more in-depth evaluation of scenarios (Hunter 1983). For this analysis, SNL chose the Columbia Intermontane Province near Hanford, Washington, as its reference site, although the geohydrology was greatly simplified for this demonstration. Three performance measures were examined: (1) integrated discharge of each of the 30 radionuclides in the transport model, (2) normalized EPA sum of radionuclide discharge to the accessible environment in 10,000 years (i.e., the containment requirements in 40 CFR Part 191), and (3) groundwater travel time from the edge of the repository to the accessible environment (10 CFR Part 60). The objectives of the study were to demonstrate how the PA methodology could be used to assess compliance with relevant regulatory criteria, to identify potential limitations of the PA methodology, to direct future research efforts, and to facilitate transfer of the methodology to NRC and its review by the technical community.

A total of 318 credible scenarios were developed (Hunter 1983), which were then screened based on probability and consequence, reducing the number of scenarios to just seven. Seventy input vectors were constructed based on the LHS sampling of 57 uncertain parameters, including hydraulic, geochemical, source-term, and other transport parameters. These 57 parameters were assigned a range of values and a PDF using information from the literature or analyst judgment. For each sample, a regional flow model, a local flow model, and a radionuclide transport simulation was performed. The consequence modeling consisted of (1) source term, (2) groundwater flow, (3) radionuclide transport, (4) biosphere transport, and (5) health effects. Each of the transport simulations yielded the integrated discharge over 10,000 years for each of 30 isotopes. However, only groundwater flow and transport in the geosphere bounded by the thermally disturbed zone and the accessible environment was modeled; therefore, there was a gap between the source term of radionuclides to the undisturbed zone and the simulation of radionuclide migration to the accessible environment.

Improvements were made to the LHS and STEPWISE computer codes and a new 2-D code was developed to solve the groundwater flow inverse problem. Sensitivity analysis was performed on the results of the base-case scenario (pre-emplacement groundwater flow), and the sensitivity analyses demonstrated that performance criteria are dependent on the physical processes taking place within the repository. Data limitation was the most critical problem encountered during this PA analysis, and limited data impacts the development of the conceptual model. In addition, the approximations in the dual-porosity model were deemed worthy further investigation (Bonano, Davis and Shipers, et al. 1989).

These PAs further advanced the state of PA by demonstrating the methodology in a different geologic medium with different conceptual models and different codes. In addition, uncertainty and sensitivity analyses were effectively implemented in the PA, and a set of fully documented codes were developed for saturated fractured media.

3.4.3 HLW Disposal in Generic Tuff Repository

In 1983, SNL published a simplified analysis of a hypothetical repository in a tuff formation (Siegel and Chu 1983) to demonstrate how the PA methodology could be used to determine compliance with the “working” draft EPA standard, 40 CFR Part 191. The reference site was an unnamed tuff formation with alternating layers of tuff of varying degrees of welding, a shallow carbonate aquifer, and block faulting. The repository was located in a densely welded tuff aquitard, approximately 900 meters (3,000 feet) below ground surface. The waste was 46,800 MTHM from spent fuel. It was assumed that retardation of radionuclides occurred in the layers of zeolitized tuff and that groundwater obeyed Darcy’s Law.

Six scenarios were developed representing different combinations of potential repository characteristics and environments (i.e., release rates and retardation of radionuclides, matrix diffusion, flow paths, location of the water table, and distances to the accessible environment). Uncertainties in geochemical and hydrogeological parameters were represented by assigning realistic ranges and probability distributions to these variables. Over 100 input vectors were constructed based on the LHS sampling of uncertain parameters, and integrated discharges were calculated one mile downgradient in a shallow aquifer. Radionuclide releases were estimated in 10,000-year increments over a 50,000-year timeframe for each scenario. All violations of the EPA draft standard in the base case were due to discharges of ^{99}Tc and ^{14}C. However, violation of the standard occurred only when the most conservative assumptions were used or when combinations of input data produced groundwater flow rates that were unrealistically high. The 1983 study recommended (1) detailed calculations of solubilities of uranium, neptunium, and radium under the geochemical conditions expected in the tuff site; (2) calculations of potential retardation due to matrix diffusion; (3) calculations of sensitivity of radionuclide discharges to assumptions about speciation; and (4) a study of the frequency of oil and water drilling and mineral exploration in areas like Yucca Mountain (Siegel and Chu 1983).

Later, Gibbons and Guzowski (1989) developed representative disruptive scenarios derived from the current site-specific understanding of the Yucca Mountain region, discussing the structural framework of the Yucca Mountain region in the context of scenario probability and describing the tectonics of this area in greater detail. In 1991, SNL recognized that existing flow and transport models were inadequate for PAs in unsaturated fractured tuff, and Parsons, Olague, and Gallegos (1991) provided guidance for new models that should be developed for PAs in such media.

Also in 1991, SNL published a proposed a methodology for conducting a PA for a HLW repository in an unsaturated tuff (Gallegos 1991). The proposed methodology was consistent with the structure for other PAs SNL had conducted, and it provided recommendations for conceptual models and numerical codes for the consequence analysis. The report identified several outstanding issues, including (1) the validity of using a continuum approach, especially those based on Darcy's Law, to model groundwater flow in unsaturated, fractured media, (2) the applicability of using the convective-dispersion equation with a K_d-based retardation factor for modeling radionuclide transport, (3) the development of efficient numerical techniques for modeling groundwater flow and radionuclide transport, and (4) the treatment of conceptual model uncertainty in PA.

The tuff PA methodology report (Gallegos 1991) concluded SNL's contract with the NRC for developing a PA methodology for the deep geologic disposal of HLW, and the tools identified and developed by SNL were transferred to the NRC and its contractors.

3.5 PAs for Generic LLW Near-Surface Disposal

SNL then developed a generic PA methodology for the NRC for evaluating license applications for LLW disposal facilities (Kozak, Chu and Mattingly 1990, Shipers 1989, Shipers and Harlan 1989, Kozak, Harlan, et al. 1989, Kozak, Chu and Harlan, et al. 1989, Kozak, Chu and Mattingly, et al. 1990). LLW facilities are generally shallow land burial sites, such as trenches or boreholes, as opposed to deep geologic disposal. The methodology contained models and computer codes for source-term release, groundwater flow and transport, air transport, surface-water transport, food chain, and dosimetry. The methodology was put together in a modular structure, in which the codes were loosely grouped, a structure that greatly increases the flexibility of the methodology to handle a wide variety of disposal options and environmental conditions, but at the cost of increased user interaction to provide coupling between the codes. Following the same PA methodology described previously, a LLW PA was demonstrated using a simple conceptual model involving land burial of ^{14}C in a shallow trench. The methodology provided NRC with a tool for performing confirmatory analyses to evaluate whether a licensee's analyses and assumptions were reasonable and to compare calculated estimates of performance against the performance objectives in 10 CFR Part 61, Licensing Requirements for Land Disposal of Radioactive Waste.

The generic LLW PA considered dose to individuals from off-site releases under normal conditions, as well as on-site doses to inadvertent intruders. The models included:

- Groundwater flow
- Source term
- Groundwater transport
- Surface water transport
- Air transport
- Pathways and dosimetry.

The PA produced a series of dose histories for each radionuclide of importance. The contribution of each individual radionuclide to the dose was summed to produce the total estimated dose, which could then be compared to 10 CFR 61.41. Significant supporting work was conducted to support this effort including an identification of potential exposure pathways (Shipers 1989), assessment of relative significance of migration and exposure pathways (Shipers and Harlan 1989), selection and integration of models (Kozak, Harlan, et al. 1989), identification and recommendation of computer codes (Kozak, Chu and Harlan, et al. 1989), and computer code implementation and assessment (Kozak, Chu and Mattingly, et al. 1990). After that generic PA was conducted in 1989–1990, an evaluation of modeling approaches (Kozak, Olague and Rao, et al. 1993) and an assessment of validation needs (Kozak and Olague 1995) served to identify potential improvements in the LLW PA models.

The LLW PA demonstrated that the PA methodology was a valuable tool in evaluating LLW facility license applications.

3.6 Significance of NRC Licensing Projects in the Historical Development of the PA Methodology

SNL's support of the NRC was essential to the advancement of the PA methodology as well as the development of the radioactive waste disposal regulations.

The risk methodology described by Cranwell, Campbell, and Helton et al. (1987) and documented in the numerous reports produced by SNL in their program of support to NRC constitutes the first comprehensive and detailed description of a versatile PA methodology for assessment of very long-term risks and quantification of uncertainties. This methodology advanced and brought together (1) techniques for selecting and screening scenarios, (2) models for use in simulating the physical processes and estimating the consequences associated with the occurrence of these scenarios, (3) probabilistic and statistical techniques for use in risk estimates and sensitivity and uncertainty analyses, and (4) a procedure for utilizing these models and techniques to assess compliance with regulatory standards. It captured the lessons learned from the ongoing SDP PA work and formalized the PA systems analysis approach into a methodology suitable for general application.

Under contracts to the NRC, SNL applied the LHS method first to nuclear reactor risk assessments and then to radioactive waste disposal PAs. LHS approach was used at SNL for performance assessments of hypothetical disposal systems has been adopted for use for all SNL probabilistic PAs of radioactive waste disposal, including the WIPP and YMP PA models and the GCD and INL HLW PAs, as well. LHS has been adopted for use in probabilistic and statistical analyses in a broad spectrum of fields beyond nuclear reactor and radioactive waste disposal risk analyses including transportation of hazardous cargoes, epidemiology, aeronautical science, semiconductor circuit design simulation, and financial risk assessment and valuation.

SNL PA studies for NRC also helped establish the regulatory basis for NRC regulations and EPA environmental standards for radioactive waste disposal by demonstrating the PA methodology as an effective tool for demonstrating and measuring compliance. It also provided effective feedback to the regulatory standards, helping to illustrate the efficacy of the criteria in achieving the intended goals, (i.e., protecting the environment and the health and safety of workers and the public).

The PA of the generic bedded salt repository was the first full exercise of the PA methodology for a geologic repository for radioactive waste. It helped advance the science of FEPs and scenario development and screening and successfully demonstrated that the results could be compared to the then draft EPA standard 40 CFR Part 191 and helped inform that regulation. The structure of the PA calculation made it possible to identify the individual vectors that caused the EPA limit to be violated.

The PA of the generic basalt repository successfully demonstrated PA methodology in a different geologic medium with different conceptual models and different computer codes, effectively incorporating uncertainty/sensitivity into the analysis. It also produced a set of fully documented codes for saturated fractured media, potentially applicable to repositories other geologic media, including tuff. The PA of the generic tuff repository further expanded the PA methodology, demonstrating its capability to incorporate new or modified physical models.

As a result of this work, SNL successfully demonstrated that the PA methodology was independent of geologic media and could be used effectively to examine compliance with the regulatory standards and requirements in 40 CFR Part 191 and 10 CFR Part 60.

4. DOE WASTE ISOLATION PILOT PLANT (1975—)

A 1957 report from the National Academy of Sciences Committee on Waste Disposal (1957) observed that "most promising method of disposal of high level waste at the present time seems to be in salt deposits." From that initial conclusion, the national effort to address radioactive focused on geologic disposal in salt. Initially, the search for a suitable site and the scientific investigations of radioactive waste disposal issues was led by Oak Ridge National Laboratory. From 1961 through the early 1970s, Oak Ridge National Laboratory conducted radioactive waste disposal experiments—most notably Project Salt Vault in an abandoned salt mine near Lyons, Kansas, from 1963 to 1967. In 1970, the AEC selected the Lyons, Kansas, site to be its first demonstration repository. In 1971, previously unknown drill holes from mineral exploration as well as solution mining were discovered near the proposed Lyons repository site, which led quickly to that site being abandoned.

With support from local leaders of Carlsbad, New Mexico, Oak Ridge National Laboratory examined a potentially suitable site in the Permian Basin and Delaware Sub-Basin in southeastern New Mexico. This potential site was identified in 1973.

In January 1975, SNL became the lead laboratory for further site characterization and for development of a conceptual repository design, and an EIS. By the end of that year, based on potentially disqualifying evidence found by exploratory drilling at well ERDA-6 on the first proposed site, SNL recommended relocating the potential repository site 11 km southeast of the first location, as shown in Figure 24 (Rechard 1999b, Figure 3-1), 42 km east of Carlsbad, and nearer the center of the Delaware Basin, where the geology was more predictable (Rechard 1999b). A timeline of the program, shown against the backdrop of other contemporary developments in PA is shown in Figure 25.

The initial conceptual design of 1977 (SNL 1977) anticipated disposal of not only TRU wastes[5] but also HLW and SNF. The conceptual design included two levels, one for contact-handled TRU waste and the other designated for remote-handled TRU waste, HLW, and an experimental area. In 1979, under the Department of Energy National Security and Military Applications of Nuclear Energy Authorization Act of 1980 (Public Law 96-164), Congress authorized the Waste Isolation Pilot Plant (WIPP) as a research and development facility to demonstrate the safe management, storage, and disposal of defense-related TRU waste, excluding HLW and SNF from consideration. Detailed design work for the WIPP, contracted to Bechtel National, led to a final repository design reconfigured for the 1980 EIS as a single-level facility. Figure 26 (Rechard 1999b, Figure 5-1) shows the 1977 conceptual design in comparison with the final construction plans. The room layout was essentially unchanged, but room dimensions were reduced and spacing between the rooms was expanded. Figure 27 presents an artist's conceptualization of the final design for the WIPP repository. Exploratory-phase construction

[5] "Transuranic waste" is defined by 40 CFR 191 partly physically—i.e., as waste containing more than 100 nanocuries of alpha-emitting isotopes of elements heavier than uranium and half-lives greater than 20 years per gram of waste—and partly by regulatory categories and exclusions, in excepting "(1) high-level radioactive wastes; (2) wastes that the [DOE] has determined, with the concurrence of the [EPA], do not need the degree of isolation required by this part; or (3) wastes that the [NRC] has approved for disposal on a case-by-case basis in accordance with 10 CFR Part 61."

began in 1981, placing the site 655 m (2,150 ft) underground within a geologically stable salt formation known as the Salado. After signing the Consultation and Cooperation Agreement with the State of New Mexico in 1983, full-scale construction of the WIPP began and continued over the next five years while SNL conducted in situ experiments to further characterize the local disposal system.

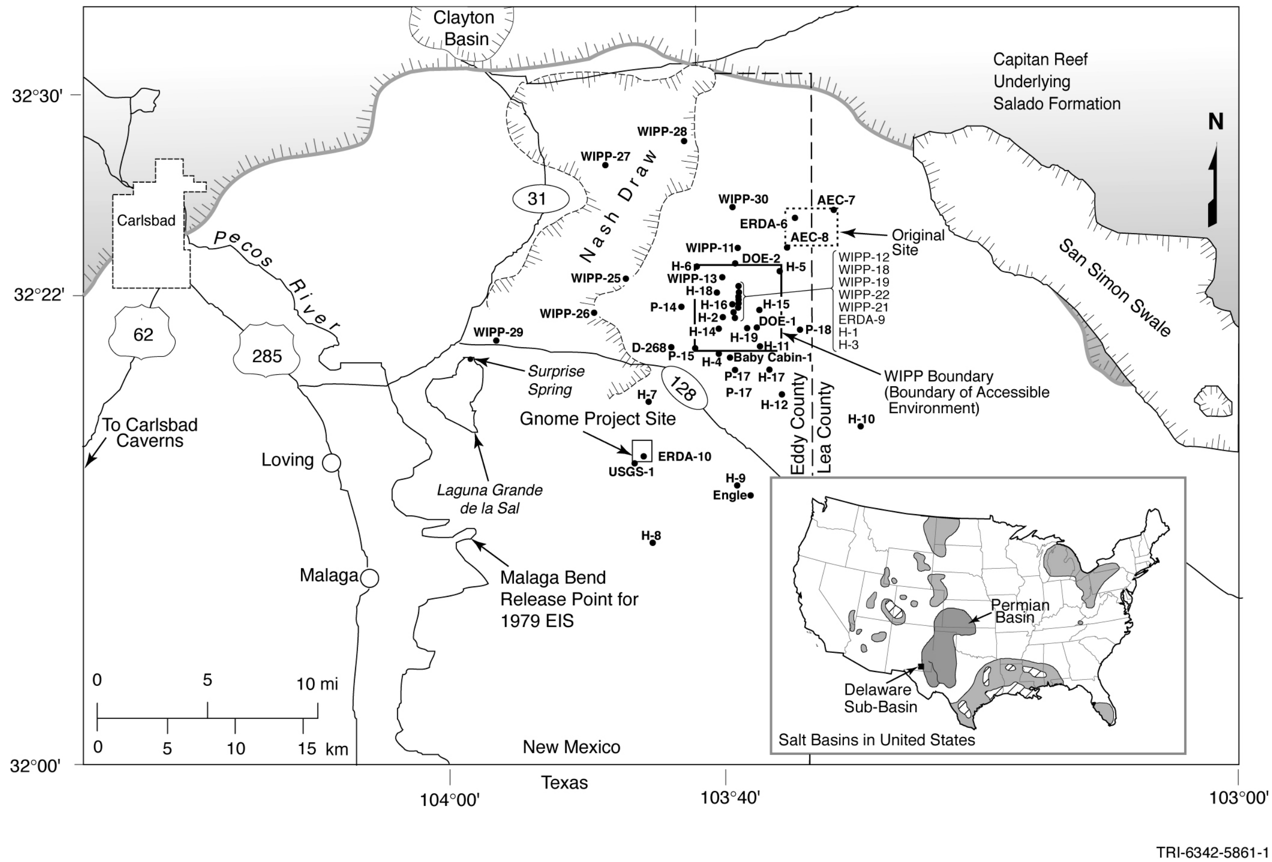


Figure 24. Location of WIPP, showing physical setting, exploratory boreholes, and the original site proposed by Oak Ridge National Laboratory

In 1986, DOE asked SNL to conduct the analyses needed to show compliance of the WIPP with EPA's environmental radiation protection standards, 40 CFR Part 191. Site characterization studies shifted toward the data needs for the demonstration PA conducted in 1989 (Marietta, Bertram-Howery, et al. 1989) and the three full PAs conducted in 1990 (Bertram-Howery, Marietta and Rechard, et al. 1990), 1991 (WIPP Performance Assessment Division 1991–1992), and 1992 (WIPP Performance Assessment Department 1992–1993). Results of the 1992 WIPP PA led DOE to conclude that the site was suitable for the disposal of TRU waste, and DOE proceeded on a path to certification under the EPA regulations. To facilitate this process, SNL developed a performance-based decision-aiding tool called the Systems Prioritization Method (SPM) (Boak, Prindle, et al. 1997), using it beginning in 1994 to assist in programmatic prioritization as the WIPP project transitioned from science to compliance.

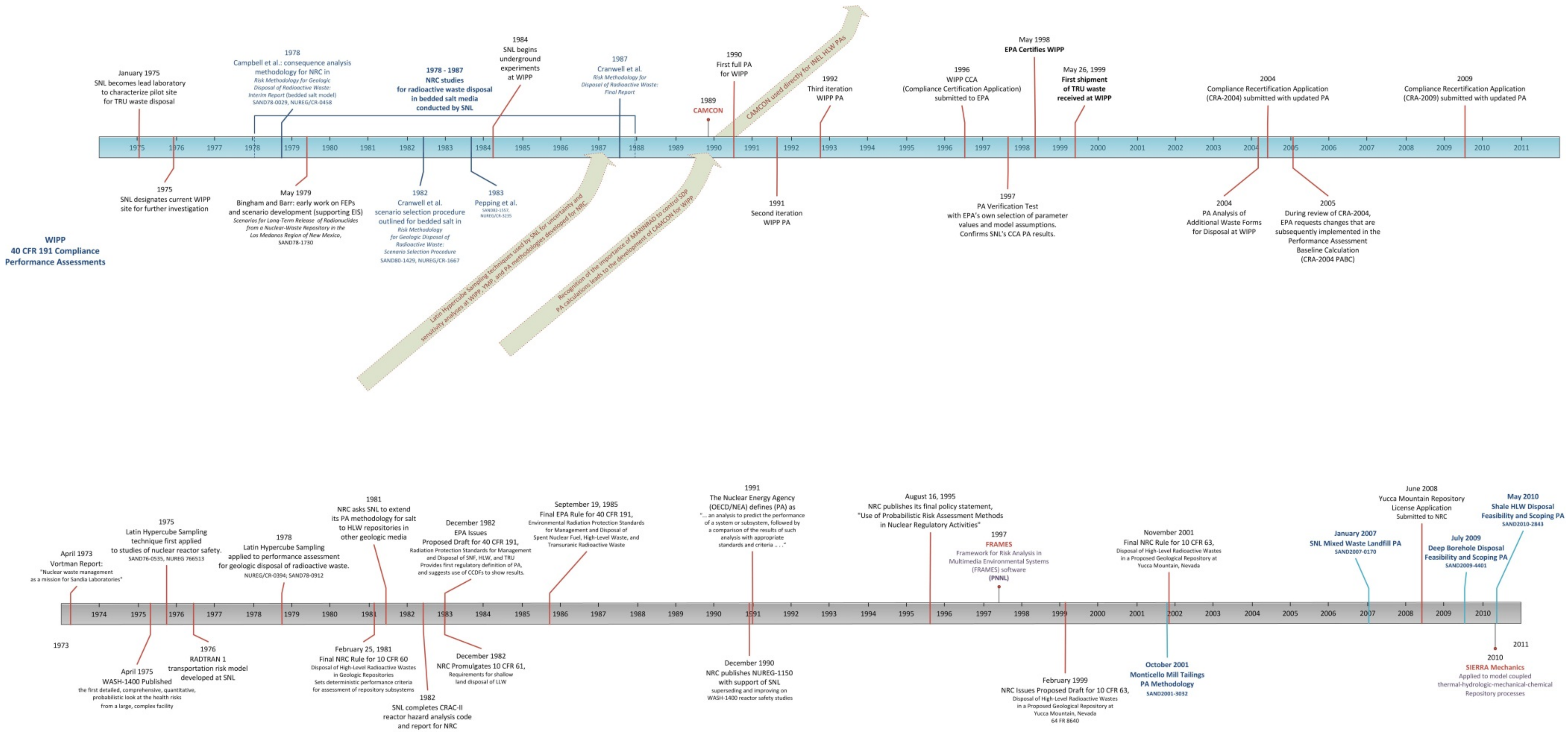
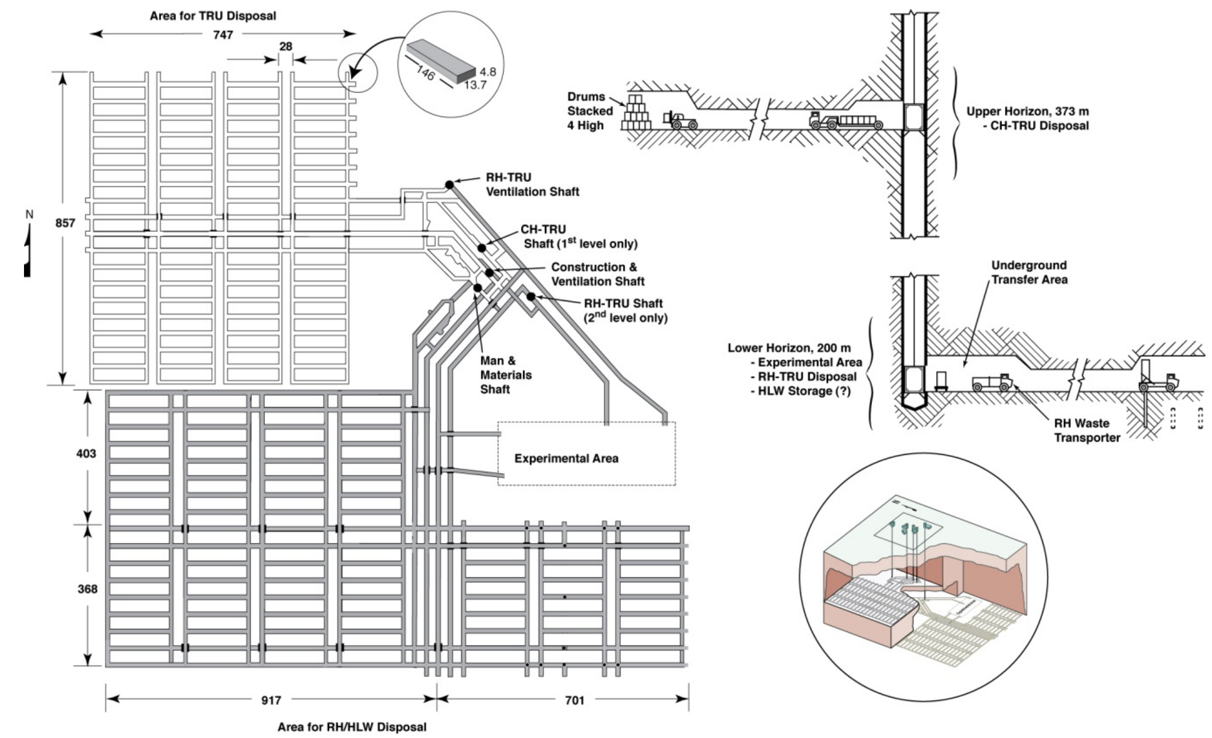
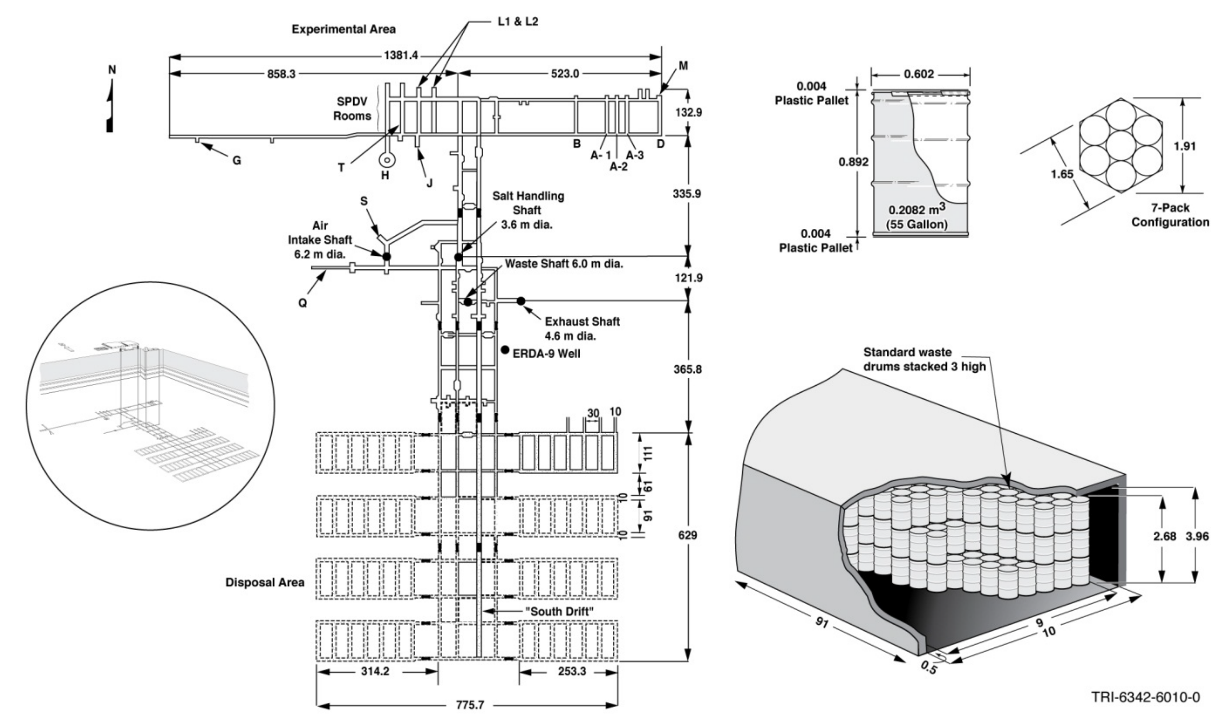


Figure 25. Waste Isolation Pilot Plant PA Timeline



(a) 1977 conceptual design



(b) Final construction plans for WIPP repository

Figure 26. WIPP repository design: (a) 1977 two-level conceptual design including emplacement areas for HLW and (b) final construction plans for a single-level repository to TRU waste only

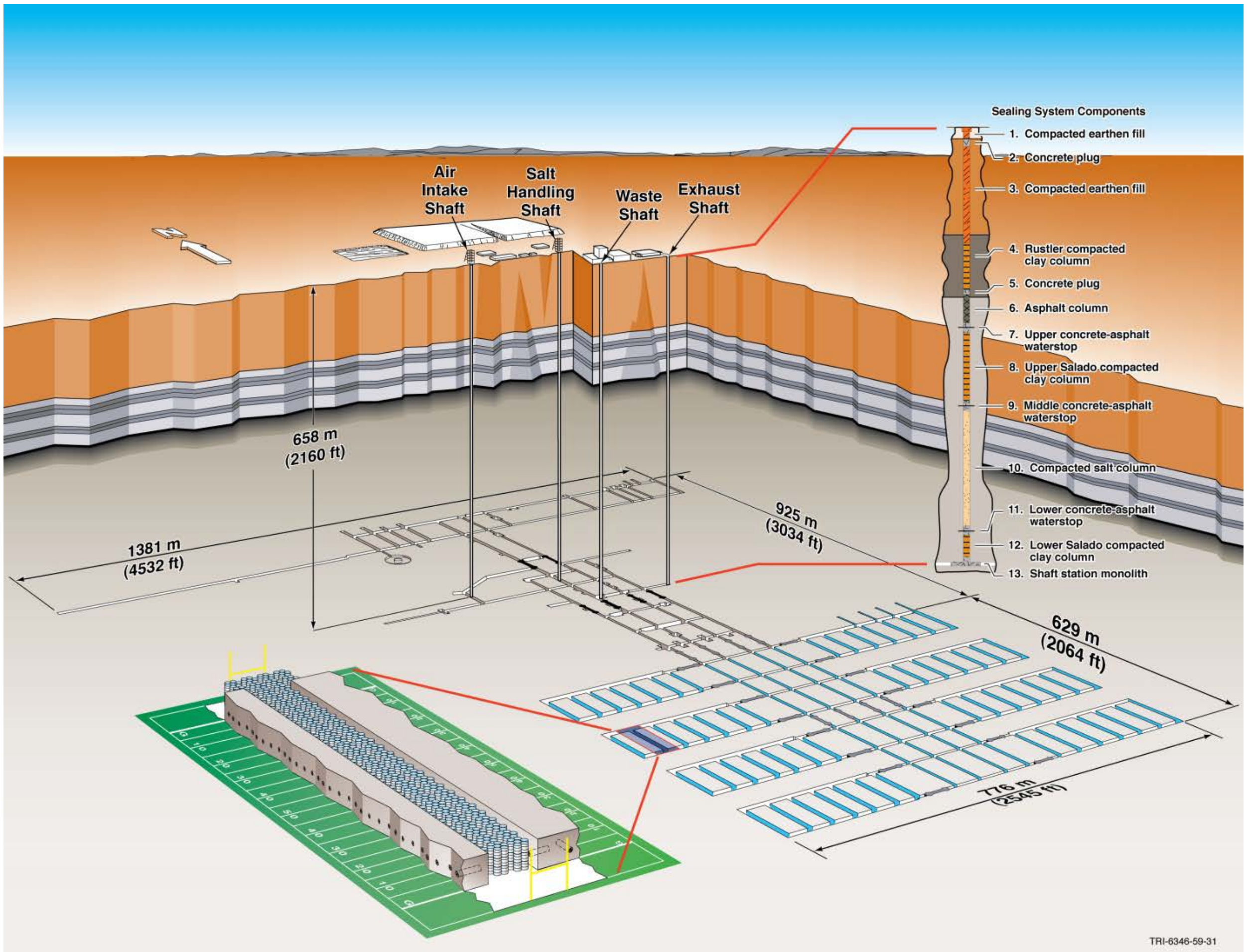


Figure 27. Conceptual illustration of the WIPP repository and shaft seals

The Compliance Certification Application (CCA) was submitted to the EPA in October 1996 (DOE 1996), and in 1999 WIPP became the first deep geologic repository certified in the U.S. to permanently dispose of TRU waste generated from the research and production of nuclear weapons, receiving its first waste shipment on March 26, 1999. In addition to the 1996 PA, which was included in the CCA, SNL subsequently conducted two additional PAs as part of the recertification applications for the site in 2004 (DOE 2004) and 2009 (DOE 2009).

4.1 Development and Description of CAMCON for the WIPP PAs

An executive program to control consequence calculations for radioactive waste disposal must meet several requirements, including built-in flexibility and built-in quality assurance. The executive program should be able to link several distinct physics model components with minimal analyst intervention; trace calculations so that they can be repeated; track parameter uncertainty using Monte Carlo techniques; and identify calculations to avoid misinterpretation. The controller should also provide easy examination of intermediate and final results, interpolation between modeling scales, and iteration between computer modules. Easy replacement of computer modules within the executive program was necessary for scenario screening, comparisons of alternatives, sensitivity analysis, and fine-tuning of the system for final compliance assessment.

The WIPP compliance assessment methodology was implemented using a modular system of computer codes controlled by a computerized executive package referred to as the "Compliance Assessment Methodology Controller"—CAMCON—developed by SNL to meet these requirements. The complex disposal system at the WIPP required that computer codes in the compliance assessment system be controlled by a computerized executive program. CAMCON was the controller for the system (Rechard 1989, Rechard, Iuzzolino, et al. 1989).

CAMCON modularized tasks so computer codes for a particular module were interchangeable. CAMCON contained translators that automatically translated output of one computer code into the appropriate input format needed for the next code. In this way, the executive controller performed a deterministic computation for each of a number of input vectors through the entire set of modules with little operator intervention (Rechard 1989), so a Monte Carlo simulation could be produced. Like packages used in other national programs, e.g., SYVAC and LISA, the CAMCON executive package was a Monte Carlo simulation controller.

The early version of the CAMCON system, outlined in Figure 28 (Rechard 1999b, Figure 7-2), consisted of six components (Rechard, Iuzzolino, et al. 1989, Rechard 1992, Rechard, Gilkey, et al. 1993, Rechard 1999b):

1. Code modules (or "grouping" of physics codes);
2. A directory' structure that facilitated configuration control;
3. A series of procedural files, CAMCONexec, that allowed an analyst to link the individual component codes and execute portions or all of a compliance assessment;
4. A set of libraries to interface with codes and users;
5. A series of help files containing instructions on use and history of updates; and
6. Two databases: CAMDAT (Compliance Assessment Methodology DATabase), a computational data base containing code outputs in .CDB files, and a secondary database of .SDB files containing parameter values. CAMDAT was the link between the computer modules.

By the version applied in the 1991 PA, CAMCON consisted of about 75 codes and FORTRAN object libraries and included approximately 293,000 lines of FORTRAN code written specifically for the WIPP Project and another 175,000 lines of code adapted from other applications (WIPP Performance Assessment Division 1991–1992, Vol. 1, p. 5-64).

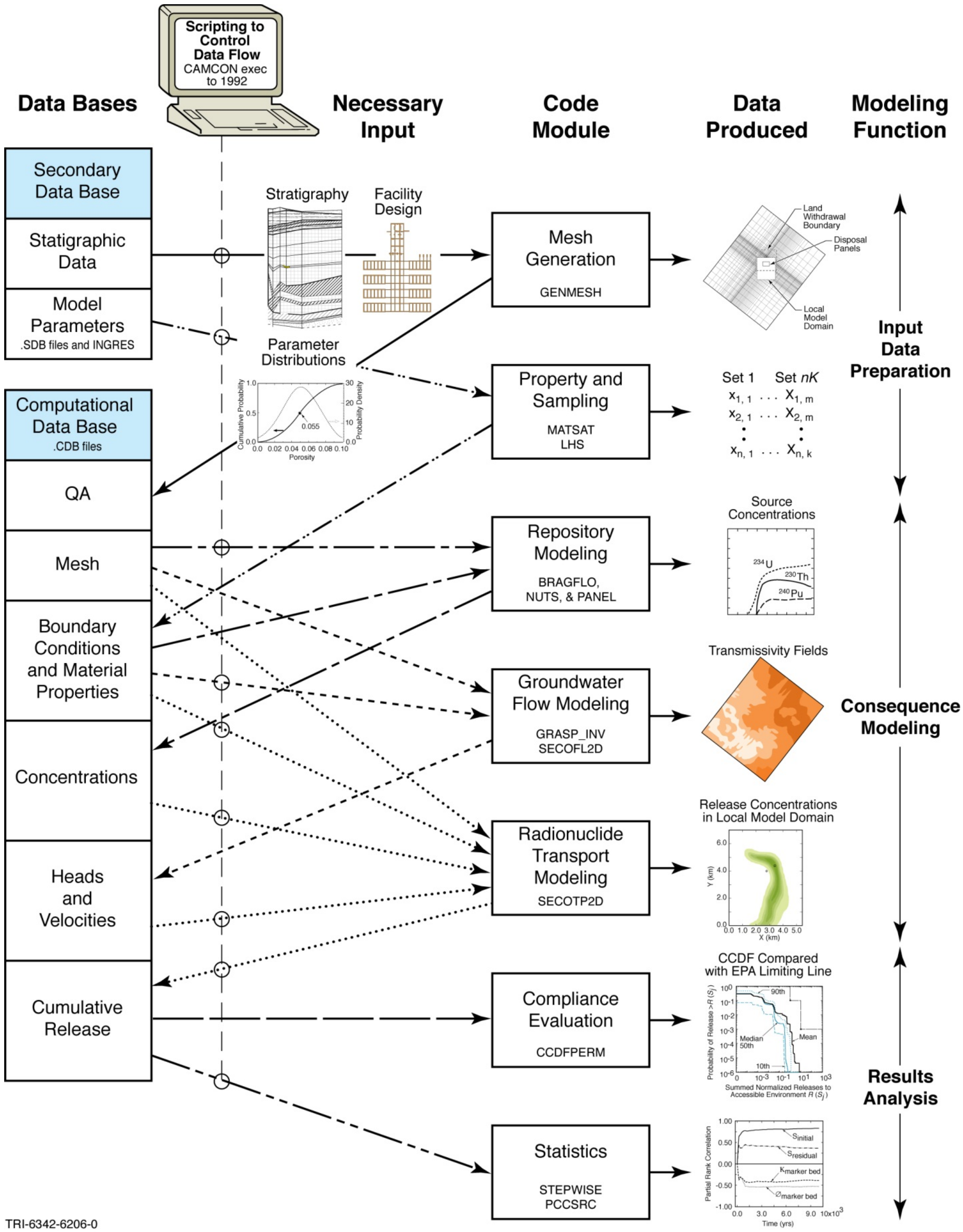


Figure 28. Schematic illustration of CAMCON system, 1991

The computational data base, CAMDAT, used a neutral file format that had evolved between 1980 and 1988 in the SNL Engineering Analysis Department (Taylor, Flanagan and Mills-Curran 1986, Mills-Curran, Gilkey and Flanagan 1988). A neutral file format allowed a series of codes to be linked by a “zigzag” connection, as illustrated in Figure 29 (Rechard 1989), rather than a serial connection. This format had the following advantages: (1) only one plotting program, which reads the computational data base, was needed to display any intermediate or final results from the many codes linked together; (2) codes were easily changed; (3) iterative calculations during calibration or convergence studies on multiple computational domains or grids were readily controlled; and (4) a controller that automates compliance assessment was easier to design because information was stored in one file format. Computational data base structure requires tracking codes that have added information to the data base to provide a trail for quality assurance by handling data from different types of codes (e.g., finite-difference, fluid-flow codes and finite-element structural-analysis codes). These features required that any program that adds information to a file also identify itself by writing a record in the file. This helped analysts to avoid misidentifying or misinterpreting a computer run when making repetitive calculations that produce similar results.

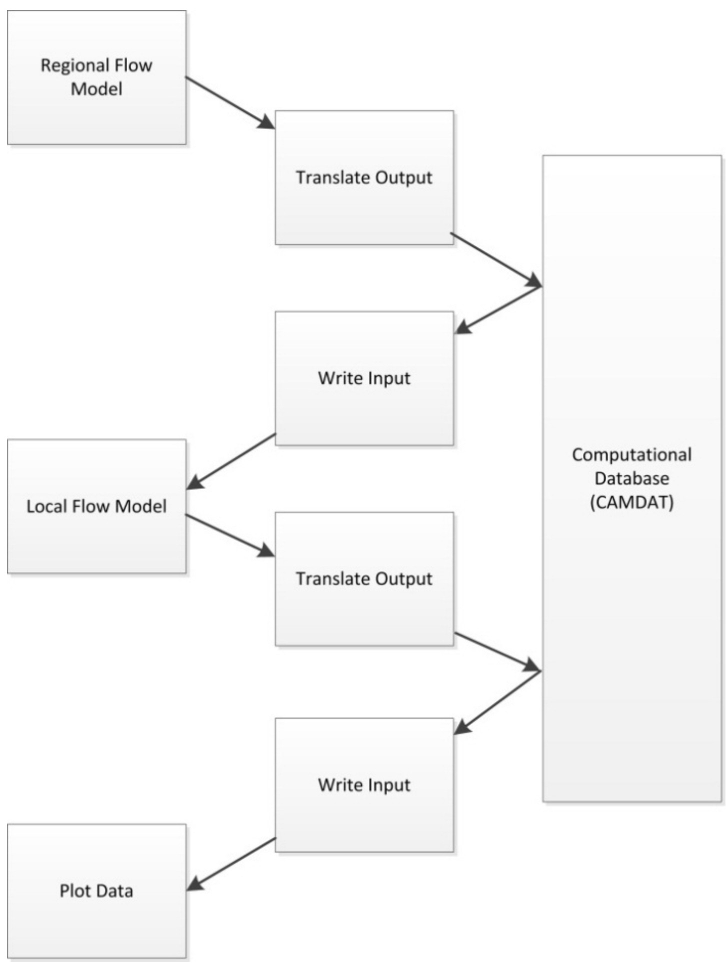


Figure 29. “Zigzag” coupling of models through the CAMDAT computational database using a neutral file format

The LHS technique (Iman and Shortencarier 1984) was used to sample the parameter PDFs, and repetitive deterministic calculations were performed using CAMCON to produce distributions of the consequence and, after including estimated scenario probabilities, CCDFs.

Code linkage and data flow through CAMDAT was controlled by CAMCON. Computer programs that made up the CAMCON system were major program modules, minor program modules, and translators. Major program modules refer to programs that represent major tasks of the consequence modeling. Minor program modules refer to programs such as interpolators that facilitate use of major program modules. Translator program modules refer to programs that translate data either into or out of the computational data base. Major program modules for consequence modeling, as shown in the schematic illustration in Figure 28 as well as the data flow illustration in Figure 30, are mesh generation, Monte Carlo sampling, regional hydrology, local hydrology, repository shaft source, transport hydrology, biological pathways, human dose, and human response. Minor program modules interpolate boundary conditions between models, track particles though simulated flow fields, generate various diagnostics and plot results. Translators communicate between codes and the secondary and computational data bases. As illustrated in Figure 30 (Rechard 1989), the algorithm for controlling a simple analysis using CAMCON on the WIPP PA was complex, but the advantages of rapid problem set up and execution with built-in quality assurance made a logical data flow and execution program necessary (Bertram-Howery, Marietta and Anderson, et al. 1989, IV-74 to IV-76).

The WIPP PA component model codes all used the first principles of physics (i.e., they contained and solved the relevant equations of, for example, transport, solubility, chemistry, and other processes, and they solved those equations for each iteration and stored the results for later review). This allowed the PA analysts to exchange one individual code module with another very simply and directly (see Figure 30), without having to rewrite much of the master code system. As the WIPP program matured, each of the physics codes were streamlined by using fixed values for the unimportant individual parameters and the unimportant subroutines, rather than using sampled values. This iterative process of streamlining component models of the PA allowed probabilistic analyses to be run in increasingly shorter times.

For the 1996 PA supporting the CCA (DOE 1996), the calculational system was conceptually the same, although details were changed. By the 1996 PA, the costs of the stringent quality assurance procedures required the selection of one code for each major component of the consequence model. Those codes specifically developed for the PA task were selected definitively, so flexibly interchangeable code modules were not necessary. Second, software specifically designed for configuration management was used rather than an ad hoc directory structure, and a disinterested third party specialist built the batch scripts for run management and control instead of using batch scripts built by PA analysts through CAMCONexec (as indicated in Figure 28). As described by Froehlich, Williamson, and Ogden (2000), efficiency of computer use increased as a result of these changes, although the driving force behind them was quality assurance: i.e., to provide the EPA auditors with objective evidence that the PA process was truly traceable, understandable, and repeatable by others.

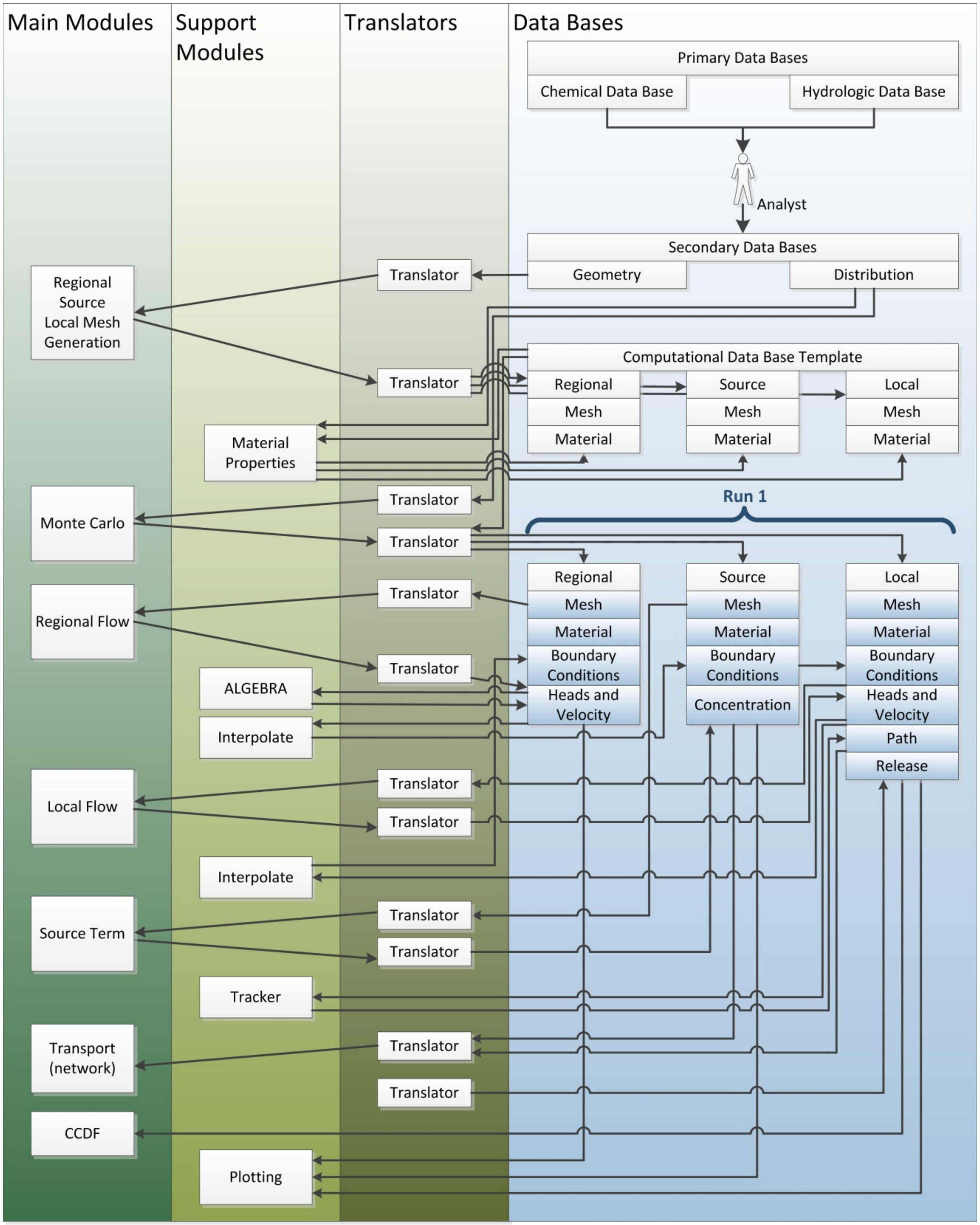


Figure 30. Algorithm for logical data flow during WIPP compliance assessment

4.2 Early Iterative PAs

A PA capability for the WIPP was developed through a sequence of PAs, with individual PAs carried out in 1989 (Marietta, Bertram-Howery, et al. 1989), 1990 (Bertram-Howery, Marietta and Rechard, et al. 1990), 1991 (WIPP Performance Assessment Department 1992–1993), and 1992 (WIPP Performance Assessment Division 1991–1992). In general, these PAs tended to follow a progression from simple and exploratory to complex and focused as the sophistication of the models and the analysis strategy for the use of these models increased and as the regulatory requirements that would be placed on the WIPP became better defined. Results presented in these PAs were recognized as preliminary and not suitable for final comparison with 40 CFR Part 191, Subpart B. Though, in each successive PA, portions of the modeling system remained incomplete, and the level of confidence in the performance estimates was not sufficient for a defensible compliance evaluation, the results were valuable in providing interim guidance to the WIPP Project as it prepared for its final compliance evaluation. (More importantly, 40 CFR Part 191, Subpart B, was remanded by a U.S. Appeals Court in 1987 (*NRDC v. US EPA*, 1987), but it was still used as the performance measure in the PAs, treating the vacated portion of 40 CFR Part 191 as if it were still effective until a new Subpart B is promulgated.) The iterative and evolutionary nature of the PA process for the WIPP contributed to the development of a final PA (i.e., the 1996 PA for the CCA) that was focused on regulatory issues of importance and was well-understood, computationally practicable, and free of serious errors.

The Iterative Approach—In 1989, the WIPP PA analysts adopted the idea of conducting an iterative sequence of PAs, conducting an initial PA with simple or incomplete models and preliminary data, followed by other PAs with better data and more detailed models (Rechard 1989). The iterative PA approach had been used in the SDP (Bishop and Hollister 1973, Klett 1997b) as well as on the 1975 Reactor Safety Study (NRC 1975) and its 1990 update as NUREG-1150 (NRC 1990). The value of repeating the PA process was that engineers and scientists gained an understanding about the disposal system and how best to model it, replacing weak links in the simulation chain as improved models and data became available (Rechard 1999b).

As described by Rechard (1999b), multiple PA iterations achieved other benefits. For example, a long, multiyear project could be divided into annual tasks, as was done with the SDP, with more easily agreed-upon goals and schedules, and allowing annual peer reviews that generated feedback that not only provided insights on the models and engineering analysis but also facilitated communication about controversial issues and fostered interactions among members of the multidisciplinary teams. In addition, later iterations based on more advanced models or newly collected data could sometimes answer critical questions posed in earlier iterations, which served to frame questions and focus investigations. For example, the question of whether a single-porosity or dual-porosity model better simulated radionuclide transport in the brine aquifer above the WIPP repository resulted in the design of a field test and a new well, H-19, to address this specific question in 1994. Finally, in combination with sensitivity analysis, iterative PAs allowed project managers, PA analysts, and experimentalists to decide how best to allocate resources for supplementary data collection and whether models should be elaborated upon or simplified in later iterations. Consequently, Sandia conducted four preliminary PAs from 1989 through 1992, with each building upon the others.

4.2.1 Demonstration WIPP PA (1989)

The 1989 demonstration WIPP PA (Marietta, Bertram-Howery, et al. 1989) demonstrated the SNL probabilistic approach and implemented a prototype of CAMCON as the calculational system controller. The results indicated that there would be no releases without a human intrusion event—a conclusion that would continue to be shown through all subsequent PAs incorporating additional site data, new events and processes, and increasingly sophisticated modeling. The calculations included features such as the pressurized brine pocket beneath the repository and gas generation caused by bacterial action, corrosion, and radiolysis that could compromise the effectiveness of the Salado Formation as a barrier. The calculations also included events such as inadvertent drilling into the repository and potash mining above the repository, and processes such as climate change that could affect brine flow in the overlying aquifer. These basic scenarios were studied in the full PAs of 1990, 1991, and 1992 PAs.

For the 1989 demonstration WIPP PA, Hunter (1989) reviewed previous work identifying events and processes that could affect the integrity of a generic disposal system (Burkholder 1980, IAEA 1983, Cranwell, Guzowski, et al. 1990) as well as specific locations (Claiborne 1974, Bingham and Barr 1979) and screened 24 events and processes, retaining eight broadly defined FEPs[6] to be included in the PA analysis. The WIPP FEPs built upon the preliminary list developed for the 1979 EIS (Bingham and Barr 1979) using new subsurface data gathered by SNL in the intervening years. The comprehensive FEPs list was based on input from nine international programs and the scenario selection guidance developed by Cranwell, Guzowski, et al. (1990), as well as WIPP-specific FEPs that included fluid intrusion from secondary and tertiary recovery at oil wells and potash mining above the repository, causing collapse in the Culebra aquifer and creating higher transmissivity zones above repository. This identification and screening was preliminary and only intended to demonstrate the process to be used in later PAs, but significant effort went into the development of the FEPs list, and the list of relevant FEPs remained fairly constant following the 1989 PA. This list was used to allow management to scope the size and length of the research and development phase of the program (i.e., the research program was complete when all of the FEPs were addressed). All FEPs were either excluded or included in the calculations. When it was formally updated with a complete re-evaluation of the FEPs list for the 1996 CCA PA, the results were fundamentally consistent.

The 1989 demonstration PA included 191 parameters, 27 of which were sampled in the stochastic analysis. Radionuclide solubility and exploratory drilling (time of intrusion, and borehole permeability) were identified during the sensitivity analyses to have the most significant effect on repository performance. CCDFs were compared to the containment requirements of the 40 CFR 191.13 for demonstration purposes only; they were not considered credible enough to judge the probability of compliance of the WIPP repository system. The preliminary, reference design CCDF based primarily on the reference conceptual models and data and exceeded the limits in the EPA standard. However, results of varying just two room parameters (hydraulic conductivity and porosity) indicated that results would not exceed EPA limits if engineered alternatives could achieve these modified parameter distributions. Figure 31

[6] By later terminology, the broad “events and processes” in this early analysis would be termed “scenarios” or even “scenario classes.”

(Marietta, Bertram-Howery, et al. 1989, Figure 4-17) shows the demonstration CCDFs, showing the reference design exceeding EPA regulatory limits and the performance improvement that could result from modifying the room and the waste. These CCDFs were constructed from 50 simulations per scenario for seven scenarios.

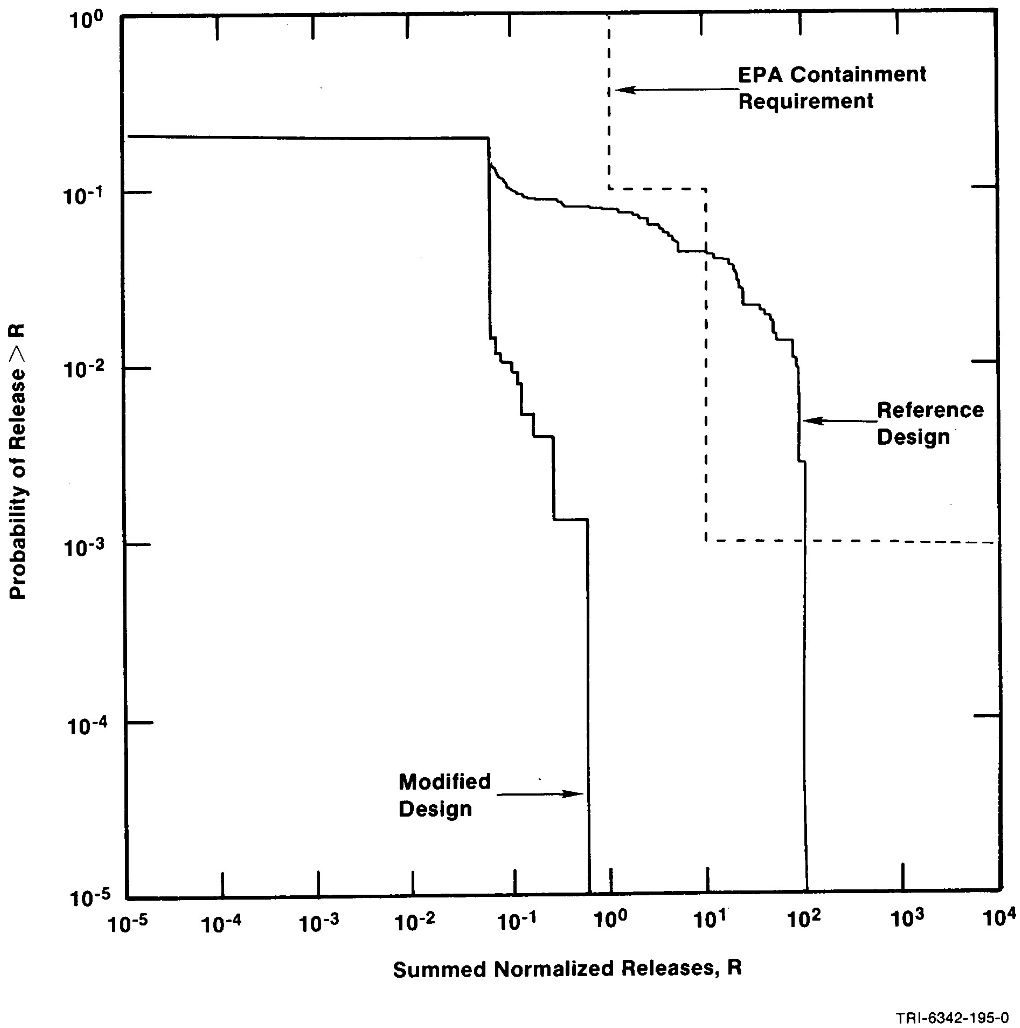


Figure 31. 1989 WIPP Demonstration PA results, showing the effect of a modified design

4.2.2 First Iteration WIPP PA (1990)

Between 1988 and 1990 SNL developed the CAMCON system to link together the detailed process models in the WIPP PA (Rechard 1989, Rechard, Iuzzolino, et al. 1989). A prototype of CAMCON was implemented in the 1989 demonstration PA (Marietta, Bertram-Howery, et al. 1989), and the full version was utilized in the first full PA conducted in 1990 (Bertram-Howery, Marietta and Rechard, et al. 1990). For the first full PA in 1990, 40 parameters were sampled, and the analysis included both scenario and parameter uncertainty. Direct releases at the surface from a drilling intrusion were identified at the most important release pathway.

The results of the 1990 WIPP PA were preliminary, contingent on assumed conceptual models and parameter value distributions. The 1990 WIPP PA included sensitivity analyses that addressed specific uncertainties in the modeling system, showing the degree to which some uncertainties in the conceptual models may affect predicted performance. Perhaps most importantly, the PA served as a full demonstration of the methodology used to assess performance. The reported CCDFs were statistical means of families of CCDFs. The modeling system was shown to be sensitive to changes in scenario probabilities, with reductions in the probability of intrusion significantly reducing predicted probabilistic cumulative releases. Comparison of clay-lined-fracture and dual-porosity transport models for the dominant water-bearing unit above the repository indicated a significant increase in radionuclide retardation and a consequent reduction in predicted releases with the dual-porosity model. Simulations of a variable number of intrusions showed that, for the selected probability model, multiple intrusions would not increase the largest cumulative releases. Simulations of a hypothetical waste modification suggested that for modifications to be effective, waste permeability must be reduced more than four orders of magnitude below the estimated unmodified value to restrict brine flow to an intruding borehole. Simulations of gas generation and the effects gas will have on brine flow and radionuclide transport were not sufficiently advanced to be incorporated in the CCDF curves for the 1990 WIPP PA, but preliminary results of one-dimensional simulations were included. Preliminary analyses for the individual protection requirements of 40 CFR Part 191 indicated that no releases will occur; therefore, dose predictions are not likely to be required for undisturbed performance (Bertram-Howery, Marietta and Rechard, et al. 1990).

4.2.3 Second Iteration WIPP PA (1991)

The main differences between the 1991 PA and previous PAs were refinement of summary scenarios into computational scenarios, the use of the Poisson assumption for calculating scenario probabilities, and the inclusion of climate variation, dual-porosity transport, and waste-generated gas effects in consequence modeling. A total of approximately 300 parameters were used in consequence modeling to specify physical, chemical, and hydrologic properties of the rock formations (geologic barriers) and of the seals, backfill, and waste form (engineered barriers), as well as parameters for climate variability and for drilling events. For the 1991 PA, 45 of these parameters were imprecisely known and were therefore selected to be sampled for use in consequence modeling for the Monte Carlo simulations of performance. For each, a range and distribution were assigned. The 1991 PA added sampled parameters related to two-phase flow and gas generation, and parameters related to dual porosity (both chemical and physical retardation) in the Culebra. In addition, a set of conditional simulations for transmissivity in the Culebra was developed in the 1991 PA, replacing a simple zonal approach used in 1990, a preliminary analysis of potential effects of climatic variability on flow in the Culebra. A Latin hypercube of sample size 60 was generated from the set of uncertain parameters to incorporate uncertainty into the PA (WIPP Performance Assessment Division 1991–1992, Vol. 1, pp. 6-17 to 6-20).

4.2.4 Third Iteration WIPP PA (1992)

The 1992 PA calculations made improvements in several important portions of the modeling system. Specific major improvements in the modeling system for 1992 included the inclusion of the effects of salt creep in the modeling of disposal-room behavior; the use of an advanced geostatistical procedure to account for spatial variability in the transmissivity of the Culebra Dolomite Member of the Rustler Formation; and the use of a new computational model for radionuclide transport in the Culebra that allows consideration of alternative conceptual models for dual-porosity and single-porosity transport. The 1992 PA was the first to apply the judgment elicited from expert panels to determine the probability of future inadvertent human intrusion into the WIPP.

Results of the 1992 preliminary comparison with the containment requirements of 40 CFR 191.13 were presented as mean CCDFs displaying estimated probabilistic releases of radionuclides to the accessible environment for 10,000 years. Results compare three conceptual models for radionuclide transport in the Culebra and two approaches to estimating the probability of inadvertent human intrusion into the repository by exploratory drilling.

The representation of system performance for the WIPP repository believed by the SNL PA analysts to be most realistic included intrusion probabilities based on judgment from expert elicitation and dual-porosity transport with chemical retardation. For intrusions occurring 1,000 years after decommissioning, the mean CCDF for this representation was more than one order of magnitude below the EPA limits. Using the same approach to intrusion probabilities used in the 1991 PA (i.e., basing the probability model on the maximum intrusion probability indicated in Appendix B of 40 CFR Part 191, without expert judgment taken into account) significantly increased the probability of releases, regardless of the model used for radionuclide transport.

Based on the premise that the major processes that will contribute to radionuclide releases had already been identified and included in the 1992 PA models for WIPP, it was believed that no additional studies would have a major impact on compliance with the EPA standard (i.e., to shift the location of the mean CCDF beyond the range displayed in the 1992 results). However, for purposes of confirmation, testing of alternative hypotheses, and reducing uncertainty in PA calculations, the 1992 WIPP PA identified several aspects of the modeling system and data base (e.g., conceptual models or distributions for important parameters that were insufficiently supported by experimental data) as requiring additional work before the PA can be considered defensible for a final comparison to the EPA standard. These needs included developing defensible values for radionuclide solubilities in repository brine; retardation factors for radionuclides in the Culebra; additional support for the dual-porosity model for transport in the Culebra; an improved model for the generation of gas as waste and containers degrade; and improvements in PA modeling. Conceptual and computational models were to be developed for pressure-dependent fracturing of the anhydrite interbeds above and below the repository. Spalling of waste into an intruding borehole as the repository depressurizes was identified to be investigated further and, if found important, to be included in PA modeling. The consequences of brine flow to the surface following borehole intrusion were to be modeled. Several aspects of groundwater flow in the Culebra, including the possible effects of subsidence related to potash mining, uncertainty resulting from the incomplete understanding of present recharge and vertical flow between units, and additional analyses of the effects of climatic change, were identified to be examined in a new three-dimensional model for regional groundwater flow. Finally, the PA suggested analyses to examine potential correlations between physical parameters than were assumed to be uncorrelated and how such correlations might affect estimated performance.

4.3 1994 Systems Prioritization Method

In March 1994, in the years intervening between the early PA iterations for WIPP and the 1996 PA for the CCA, SNL developed a performance-based decision-aiding tool for the DOE Carlsbad Area Office to assist in programmatic prioritization of technical work activities as the WIPP project transitioned from science to compliance (Helton, Anderson, et al. 1996). The goal of this tool, called the Systems Prioritization Method (SPM), was to reveal how potential activities (i.e., scientific investigations, engineering and design alternatives, and waste acceptance criteria), either alone or in combination, could contribute to a demonstration of compliance with regulatory performance requirements. SPM was designed by SNL to (1) identify programmatic options (activities) and their costs and durations, (2) analyze combinations of activities in terms of their predicted contribution to long-term performance of the WIPP disposal system, and (3) analyze cost, duration, and performance tradeoffs. In addition, it provided additional benefits to the WIPP program, including training and developing acceptance among individual field and laboratory scientists for the PA methodology as well as initiating a dialog with public stakeholders—including adversaries.

Estimates of predicted performance of disposal systems are determined by states of knowledge which change over time as a result of scientific investigations, changes in design and engineering, or modifications to waste acceptance criteria. The changed state of knowledge can result in reduced uncertainty or improved system performance, altering the position of the CCDF used to compare against regulatory performance criteria. The SPM applied expert judgment and implemented the SNL WIPP PA models to estimate how the disposal system might perform if activities were implemented, to calculate the activities' probability of demonstrating compliance.

As described by Boak et al. (1996), and as illustrated in the overview to the SNL PA methodology in Figure 9 (see Section 1.2.3), SPM can be outlined in eleven steps:

1. Define the performance objective (i.e., long-term performance requirements in 40 CFR 191.13(a) and 40 CFR 268.6);

2. Develop a technical baseline for SPM calculations;

3. Perform modeling of the baseline;

4. Determine whether the baseline is predicted to succeed or fail in meeting the performance objectives using a binary compliance indicator (if the baseline is predicted to comply, proceed to Step 11);

5. (If the baseline fails to meet performance objectives) Identify activities that, if implemented, could improve a predicted ability to meet the performance objectives, and elicit potential outcomes for those activities;

6. Evaluate the baseline combined with potential outcomes of activities (i.e., calculate the probability of demonstrating compliance);

7. Create a decision matrix containing the probability of demonstrating compliance, cost, and duration for all activities and perform decision analysis to develop final recommendations;

8. Make programmatic decisions about which activities to implement, if any;

9. Implement the activities;

10. Update the technical baseline with actual results after implementing the activities; and iterate the process from step 3 as necessary until the baseline is predicted to meet the performance objectives; and,

11. Perform final compliance calculations with approved data and models when the baseline is predicted to comply.

As shown in Section 1.2.3 in comparison to Figure 2 and also Figure 10, the SPM process (Figure 9) is very similar to the general SNL PA process, but it is distinct from PA compliance calculations in important ways. The SPM was a strategic planning approach, adopting performance measures (i.e., probability of demonstrating compliance as well as the cost and duration of activities) that are derived only in part from regulatory performance measures. Though the SPM may be conducted iteratively, it does not involve model validation or calibration activities for programmatic alternatives. The SPM applied PA codes at a level of abstraction sufficient to discriminate between programmatic options but insufficient for the rigor and detail required in a complete PA. Maintaining this distinction is important to keep probabilistic calculations tractable and in maintaining an efficient planning process.

The first iteration of SPM (SPM-1), completed in September 1994 (Helton, Anderson, et al. 1996), served as a prototype for the approach implemented in the second iteration (SPM-2) (Prindle, Mendenhall and Boak, et al. 1996, Prindle, Mendenhall and Beyeler, et al. 1996, Prindle, Boak, et al. 1996). SPM-2, completed in March 1995, was the basis for programmatic decision-making. WIPP project technical staff, stakeholders, and oversight groups contributed to establishing the SPM-2 baseline. Technical teams also defined proposed activities and were elicited on the predicted outcomes of those activities. Trained elicitors external to the WIPP project formally elicited the technical baseline and proposed scientific activities from the technical teams. The DOE Carlsbad Area Office and the Westinghouse Waste Isolation Division provided information regarding engineered alternatives, potential changes to waste acceptance criteria, and other programmatic guidance.

Potential outcomes were initially elicited for 37 scientific investigations, 18 engineered alternatives, and three waste acceptance criteria. These were screened to 26 discrete activities for the final SPM-2 analysis—21 investigations, three engineered alternatives, and two waste acceptance criteria. SPM-2 used existing WIPP performance assessment computer codes, with modifications required to model the baseline and activity sets, to calculate CCDFs of potential radionuclide releases. SPM-2 evaluated more than 600,000 possible activity sets. Activities that had no performance impact were removed from the decision matrix, reducing the number of activity sets in the decision matrix to roughly 46,700. The analysis of these 46,700 unique activity sets produced over 1.3 million CCDFs. A statistical regression analysis was conducted to determine the most favorable activity sets for meeting the DOE objectives for the WIPP. The results led to DOE's selection of eight activities (out of the 58 activities proposed for consideration) to be implemented to address key regulatory issues. In addition, analysis of the results also indicated that optimal programmatic options existed and that activities could be systematically cut or added if budgets changed. The analysis indicated that a demonstration of compliance could be anticipated within the DOE WIPP schedule. These eight activities were completed in support of and their outcome reflected in the successful 1996 Compliance Certification Application (Boak, Prindle, et al. 1997).

However, the WIPP SPM had significant shortcomings and limitations and has been subject to reasonable criticism. Notably, the SPM cost more in time and money than a general sensitivity analysis conducted as a normal part of the PA process, and the additional information it supplied mainly confirmed earlier sensitivity studies. In addition, some basic tenets of decision analysis, such as developing an explicit utility function, were not followed (Lee 1996). In its practical application, the WIPP SPM analysis was not probabilistic because the time needed to run a sufficient simulation would have been excessive; instead, only a deterministic simulation of each activity was run using an ad hoc combination of mean and median parameter values (Rechard 1999b).

The value—and cost—of the SPM approach may therefore best be seen in the inclusiveness of input to the process rather than in producing results differing from the normal sensitivity and uncertainty analysis process. The process involved external parties (supporters and critics alike) and internal parties with differing or competing points of view in framing the analysis and evaluating their alternative views and proposed activities. To take an internal example, when the WIPP PA program started, the probabilistic PA process was new and many WIPP researchers and scientists had not accepted use of PA sensitivity/uncertainty analyses as a tool to guide research, preferring instead traditional planning and budgeting approaches to research. Given that personal research interests might be put at risk, many scientists resisted the use of PA in identifying important (and unimportant) areas for further research. SPM helped institutionalize the use of PA in setting research objectives: individual scientists could no longer suggest that their data value or subroutine outcome was the most important one. Sampling over the total range of their data spread and in the context of overall system performance provided objective evidence of the relative importance of a single values and particular parameters. Similarly, oppositional views and other public concerns were reflected in the SPM calculations, which helped to resolve some disputes and, more importantly, demonstrate openness in consideration of criticism and concern from the public.

Thus, the SPM effort served to involve and inform a broad range of stakeholders by soliciting and using their input and then publishing SPM results in the form of a relational database on a CD-ROM that collected the PA results, data analysis and visualization tools, and other documentation. SPM built upon the power of both PA and decision analysis techniques, focusing on work to achieve compliance with long-term disposal system performance requirements and helping to eliminate concerns that program activities were not clearly and demonstrably focused on addressing regulatory and safety issues (Boak, Prindle, et al. 1997).

4.4 1996 Compliance Certification Application PA

For the 1996 PA in support of the CCA, a new FEPs identification and screening process was conducted. Hazard identification began with lists developed in the 1990s for international programs and relied heavily on the comprehensive list developed by Sweden in 1993 (Stenhouse, Shinta and Garner 1993). Reasons for omitting or retaining specific FEPs were fully documented. For example, the low probability and low consequence arguments for not considering criticality in or around the repository were formally documented in a 100-page report (Rechard, Stockman, et al. 1996). In addition, two human-initiated events were added to the initial list: (1) subsidence in the Culebra after potash had been mined above the repository, as mandated by the implementing regulation for the WIPP, 40 CFR Part 194, and (2) the potential for inadvertently injecting large volumes of water into the repository through anhydrite layers in the Salado because of failed casing (Stoelzel and O'Brien 1996, Stoelzel and Swift 1997). The latter event was based on experience in the Delaware Basin from drilling new oil wells in areas where water flooding had occurred to enhance oil recovery from deep oil reservoirs. Prior to 1996, the uncertainty about whether the most appropriate FEPs had been included for analysis had not been formally reviewed. Later, during the EPA's 1997 review of the CCA, the justifications for eliminating various FEPs were closely examined (Rechard 1999b).

The primary physics models, many having a number of submodels and codes, used in the 1996 CCA PA were the following:

- **BRAGFLO**, which calculated multiphase flow of gas and brine through a porous heterogeneous reservoir. It used finite difference procedures to solve system of nonlinear partial differential equations describing the conservation of gas and brine along with appropriate constraint equations, initial conditions, and boundary conditions.
- **BRAGFLO-DBR**, a special configuration of BRAGFLO model that was used to calculate dissolved radionuclide releases to the surface at the time of a drilling intrusion. It used initial value conditions obtained from calculations performed with BRAGFLO and CUTTINGS_S.
- **CUTTINGS_S**, which calculated quantity of radioactive material brought to the surface in cuttings, cavings, and spallings generated by a drilling intrusion. It used initial value conditions obtained from calculations performed with BRAGFLO.
- **GRASP-INV**, which generated transmissivity fields conditioned on measured transmissivity values and calibrated to steady-state and transient pressure data at well locations using an adjoint sensitivity and pilot-point technique.
- **NUTS**, which solved systems of partial differential equations for dissolved radionuclide transport in the vicinity of the repository, using brine volumes and flows calculated by BRAGFLO as input.
- **PANEL**, which calculated rate of discharge and cumulative discharge of radionuclides from a waste panel through an intruding borehole, using brine volumes and flows calculated by BRAGFLO as input.
- **SANTOS**, which determined quasistatic, large deformation, inelastic response of two-dimensional solids with finite element techniques to determine porosity of waste as a function of time and cumulative gas generation, which was an input to calculations performed with BRAGFLO.

- **SECOFL2D**, which calculated single-phase Darcy flow for groundwater flow in two dimensions based on a partial differential equation for hydraulic head, using transmissivity fields generated by GRASP-INV.
- **SECOTP2D**, which simulated transport of radionuclides in a fractured porous medium, solving two partial differential equations, one providing a two-dimensional representation for convective and diffusive radionuclide transport in fractures and the other providing a one-dimensional representation for diffusion of radionuclides into the rock matrix surrounding the fractures. It used the flow fields calculated by SECOFL2D.

The formal PA calculation for the 1996 CCA involved on the order of 50 codes altogether, represented by some 21 code sponsors. To ensure traceability and reproducibility, an integrated computational environment was established (Froehlich, Williamson and Ogden 2000). Automated scripting and file management software was used up front to eliminate as many manual steps as possible, also minimizing the repeated qualification of manual steps each time they were performed. In addition, all official runs were controlled by two Run Coordinators, who made all the decisions about utilization of CPU time and other system resources for the CCA, balancing resource utilization between automated runs and ongoing analysis tasks. The calculation took five months (March through July, 1996), and was conducted by only two people (the Run Coordinators). The CCA calculation ran on 40 DEC Alpha processors for 37,000 CPU hours (over 4.2 CPU-years), with over 225,000 files retained. The 59,000 batch jobs, each composed of runstreams invoking from 2 to 30 executables, translate to over 700,000 individual code executions. These practices ensured that compliance with regulations and quality assurance requirements could be readily demonstrated.

In the 1996 CCA PA (DOE 1996), 57 uncertain parameters were sampled using LHS, and 100 vectors were assembled (Rechard 1999b). The overall exposure pathway model for the 1996 PA was run 100 times with LHS samples. Random sampling of the occurrence of possible future events generated the possible futures that yield the CCDF. The 100 LHS vectors were then replicated three times (using new random numbers) to demonstrate the stability of the results. Many of the phenomenological models were run many more times than 300 because of the various pathways, scenarios, and times of intrusion used in the analysis.

The PA results showed that radionuclide release to the accessible environment boundary is negligible for the undisturbed scenario and has no impact on compliance. For the disturbed scenario, there are four potential release mechanisms:

1. Cuttings and cavings (releases via material brought to the surface removed directly by the drill bit or drill fluid)

2. Spallings (releases of material pushed to the surface by gas pressure in the repository)

3. Direct brine release at the surface during drilling (where contaminated brine under pressure at the repository flows up the intrusion borehole)

4. Groundwater releases following groundwater transport after the drilling event.

Cuttings and cavings are the most significant contributors to the mean CCDF, while spallings make a small contribution and direct brine releases are less important. The most significant parameters are microbial degradation of cellulose which was mitigated by the introduction of magnesium carbonate, shear strength of waste, corrosion rate for steel, waste particle diameter, initial value for halite permeability, borehole permeability, increase in brine saturation of waste due to capillary force, and anhydrite permeability. The CCA results demonstrated with greater than 95% confidence that the overall mean CCDF, which is presented in Figure 32 (DOE 1996, Figure 1-1), is in compliance with the containment requirements contained in 40 CFR 191.13. Figure 33 (Rechard 1999b, Figure 9-1) compares the WIPP PA results from the first three full PA iterations in 1990, 1991, and 1992 with the 1996 CCA PA results.

During the review of the CCA, the EPA requested an additional Performance Assessment Verification Test (SNL 1997), which revised selected CCA inputs to the PA. The Performance Assessment Verification Test analysis ran the full suite of WIPP PA codes and confirmed the conclusions of the CCA analysis that the repository design met the regulations.

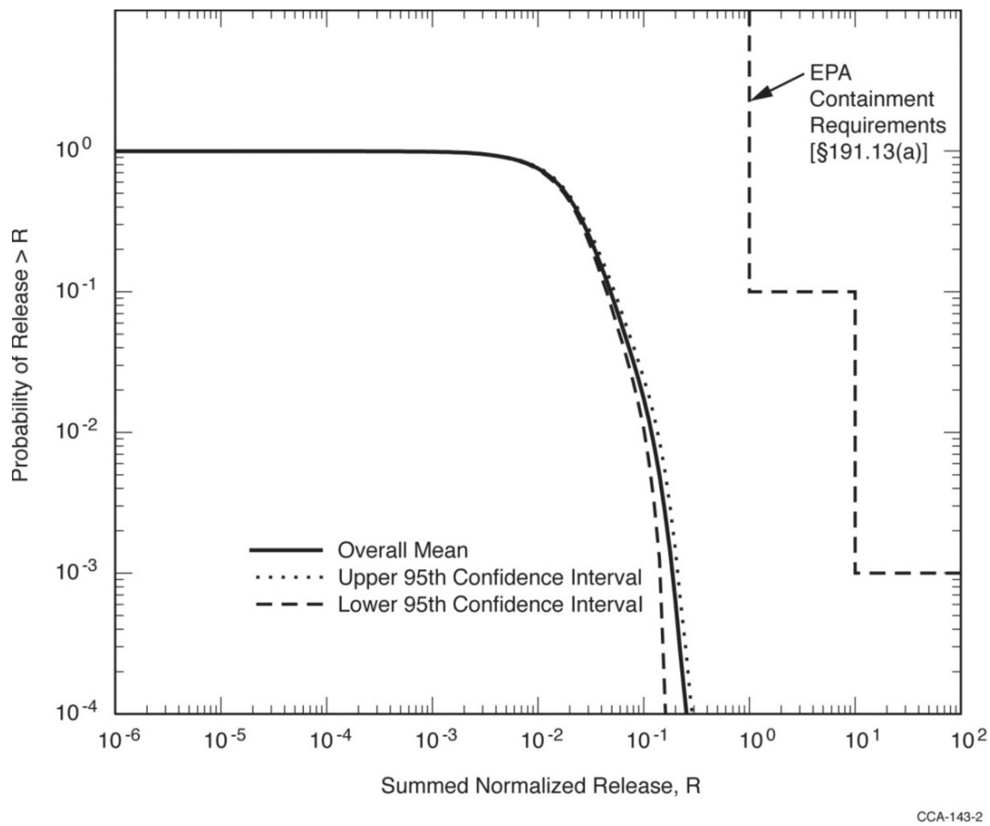
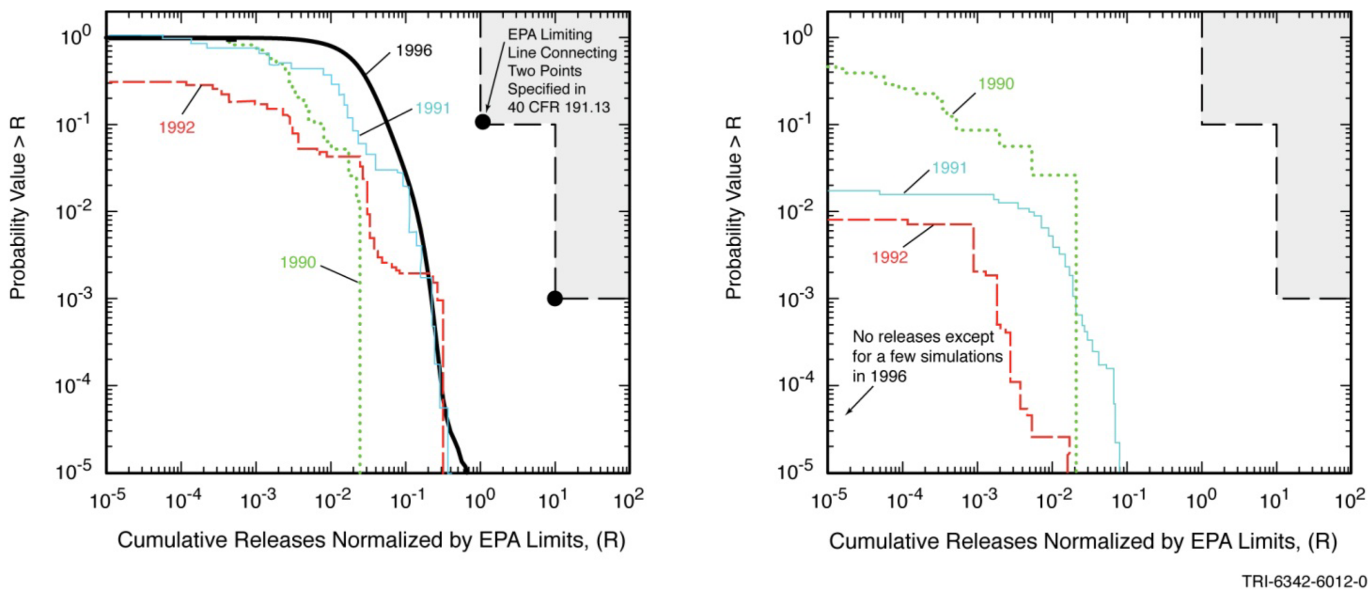


Figure 32. 1996 WIPP CCA results



(a) Direct Releases to the Surface during Drilling

(b) Releases via Groundwater Pathway through Culebra

Figure 33. Comparison of WIPP PA results 1990–1996

4.5 PA Analyses for Advanced Mixed Waste Treatment Project (Idaho) Supercompacted Waste and Other Waste Forms

When WIPP was originally certified in 1997, the planned waste containers were not physically strong (in comparison to the geologic forces that would be exerted over time). These waste containers, it was assumed, would degrade relatively rapidly in the WIPP repository environment would compress and mix somewhat under the force of halite creep during room closure, eventually forming a waste mass that, on the average, can be considered homogeneous. These original assumptions, along with the assumption of random placement, supported treating the waste as a homogeneous, well-mixed material in the CCA PA.

In December 2003, the DOE requested that the EPA approve for emplacement at WIPP the supercompacted wastes processed by the Advanced Mixed Waste Treatment Project (AMWTP) at the Idaho National Engineering and Environmental Laboratory and pipe overpack waste from Rocky Flats. The AMWTP is designed to retrieve, characterize, repackage, and compact 55-gallon drums of contact-handled, mixed transuranic debris waste, and place three to five of the compacted drums (now flattened into “pucks” of with final volumes between 15 to 35 gallons) into 100-gallon drums for shipping and disposal at WIPP. Pipe overpacks from Rocky Flats are stainless steel cylinders considerably more rigid than the standard waste containers received at WIPP.

When compared to standard (uncompressed) waste, the supercompacted waste from the AMWTP is expected to have stronger structural properties, higher concentrations of gas-generating material (cellulosic, plastic, and rubber materials), and lower radioactivity. The pucks will be compressed by a greater pressure than they would be subjected to underground, so they will not compress any further during room closure, unlike the standard waste. The pucks are expected to remain rigid, and unlikely to mix with other waste materials.

Hansen, Brush et al. (2004) conducted a PA to assess the impact of supercompacted waste on repository performance. In the first step of the Advanced Mixed Waste (AMW) PA analysis, the WIPP FEPs baseline was examined to determine whether the FEPs included (i.e., screened in) in the baseline WIPP PA would account for the pipe overpack and AMWTP waste. The analysis confirmed baseline screening decisions (for both included and excluded FEPs) were still valid. Thus, no new FEPs were added to the AMW PA to accommodate the pipe overpack and the AMWTP waste in the inventory. The FEPs analysis identified several models, parameters, or numerical implementation of models that, while their screening decisions would be unchanged, needed further investigation for specific effects of the inclusion of supercompacted and pipe overpack wastes. Those models and parameters were creep closure of waste-filled regions; chemical conditions in the repository assumed for calculation of actinide solubilities; gas generation models; parameters representing hydrological and mechanical properties of the waste (permeability, shear strength, and tensile strength); waste heterogeneity in direct release models; and mechanisms used in the model for spallings (blowout, stuck pipe, and gas erosion.)

The AMW PA considered four different panel loading schemes:

1. Standard Waste Model. The standard waste model represents a room filled with a homogeneous mix of waste in 55-gallon drums, identical to the assumptions for the CCA and Performance Assessment Verification Test. The standard model represents a bounding case of high initial porosity and structurally compliant waste packages.

2. Combined Waste Model. This model assumes that stiff and structurally compliant wastes are mixed within a room. Supercompacted waste is used for the stiff waste, and standard waste is used for the compliant waste. A mix of two-thirds supercompacted waste and one-third standard waste (by volume) was selected for this model.

3. Supercompacted Waste Model. This model assumes that all waste is structurally similar to supercompacted waste. This model reflects a bounding case where the initial porosity is low and the waste packages are stiff.

4. Pipe Overpack Model. This model assumes all waste is structurally similar to pipe overpacks. This model represents a bounding case where initial porosity is high and the waste packages are stiff.

For the AMW PA, 30% of the vectors used the first panel loading scheme, 30% used the second, 30% used the third scheme, and 10% used the fourth scheme. Additionally, to simulate non-uniform loading of cellulosic, plastic, and rubber materials within the repository, a sampled parameter was introduced for the AMWTP analysis to represent the fraction of a single panel's volume that is filled with AMWTP waste (supercompacted and not) with variation between 0.2 and 1. The analyses of the possible changes to waste representation in process models to account for the mechanical and hydrologic properties of supercompacted and pipe overpack wastes and to represent heterogeneity in the waste materials concluded that most models and parameters would be appropriately represented by the baseline WIPP PA models. The waste form chemistry was consistent with baseline radionuclide solubility models and corrosion models. Other parameters were consistent with baseline parameters, bounded by conservative representations in the baseline models, or shown by sensitivity analyses to be insignificant to overall performance.

However, modeling of waste porosity implemented new uncertain parameters for selecting porosity surfaces for waste-filled regions that represent bounding porosity cases, and an uncertain parameter representing nonuniform cellulosic, plastic, and rubber material concentration in waste-filled regions was implemented in the gas generation models (Stein and Hansen 2004).

Analysis of results from the AMW PA showed that total normalized releases from the repository fall below the regulatory limits specified in 40 CFR Part 194, demonstrating compliance with the regulations regardless of how waste is represented in the calculations. Comparison of the results of the AMW PA and baseline PA showed essentially the same range of uncertainty in repository performance, indicating that explicit representation of supercompacted waste and pipe overpack waste in the WIPP PA would not result in significant changes to the range of estimated repository performance. In addition, a sensitivity analysis showed that repository pressure and total releases were quite insensitive to the uncertainty in cellulosic, plastic, and rubber material distribution. The sensitivity analysis showed that, among the four porosity surfaces considered, the application of the Pipe Overpack Model resulted in the highest porosity, but that the Combined Waste Model in the rest of repository had the greatest effect on repository pressures. Pressures were systematically lower in the AMW PA calculations than in the baseline WIPP PA and porosity was typically higher. However, mean total releases in the AMW PA were not significantly different than for the baseline PA. Thus, the sensitivity analysis concluded that cellulosic, plastic, and rubber materials could continue to be represented as homogeneously distributed, and that PA should continue to use the Standard Waste Model to represent waste porosity.

The sensitivity analysis considered the importance of the assumption of random waste placement and spatial correlation among waste streams in the calculation of direct releases. The analysis found that the mean and 90th percentile CCDFs for cuttings, cavings, and spallings releases are not significantly different when waste is placed randomly or when waste is placed as contiguous blocks comprising single waste streams. Furthermore, above a probability of 0.001, the practice of selecting three waste streams for cuttings and the repository average radioactivity for spallings results in greater releases and is thus conservative. Therefore, the AMW PA concluded that direct releases are relatively insensitive to uncertainty in the spatial arrangement of the waste and the baseline practice of assuming no spatial correlation of waste streams in vertical stacks is conservative.

The AMW PA concluded that repository performance with supercompacted AMWTP waste and pipe overpack waste included in the inventory would still comply with the regulations specified in 40 CFR Part 194. Moreover, explicit representation of the specific features of supercompacted waste, such as structural rigidity and high cellulosic, plastic, and rubber concentration, would not be warranted, since the AMW PA results demonstrated that overall performance would be insensitive to the effects of these specific features. Finally, the AMW PA showed that PA results are not significantly affected by the assumption of random waste placement and the representation of waste as a homogeneous material, and, as a result, when the recertification application was made later in 2004, the changes resulting from the new waste forms were reflected in terms of their materials composition including cellulosic, plastic, and rubber materials and radionuclide inventory, but it was not necessary to significantly modify the PA or place operational restrictions on repository loading and distribution of these waste forms.

4.6 Recertification PAs (2004, 2009)

The WIPP Land Withdrawal Act requires the DOE to provide the EPA with documentation of continued compliance with the EPA's disposal standards within five years of first waste receipt and every five years thereafter. The PA conducted for the first Compliance Recertification Application (CRA), known as the CRA-2004 PA, is documented in the DOE's CRA (DOE 2004). During review of the 2004 CRA, the EPA required several changes to the PA. These changes were subsequently implemented in a new PA, known as the CRA-2004 Performance Assessment Baseline Calculation (Leigh, et al. 2005). The PA that SNL conducted for DOE's second CRA in 2009 was known as the CRA-2009 PA, and it was documented in DOE's 2009 WIPP CRA (DOE 2009). In all of these PAs, the results continued to show that the estimated releases are well below release limits, as was shown in the initial 1996 CCA PA.

As part of the 2004 recertification effort, SNL assessed the impacts of new information on the original FEPs baseline to determine if changes to the original decisions are necessary. The FEPs baseline could be affected by new information from literature, experiments, observations from monitoring programs, or changes implemented by the DOE (e.g., moving the WIPP horizon to Clay G). The reassessment of FEPs resulted in the addition of two new FEPs to better represent solution mining and the deletion of four FEPs by combining the deleted FEPs into related FEPs. Seven screening decisions were also changed as a result of new information. However, only three FEPs previously screened out were screened into the CRA-2004 PA. The impact of organic ligands, represented in two FEPs, was screened in as a result of new information and was the only FEPs screening decision representing an impact to the PA. The other newly screened-in FEP, surface disruptions, was already implicitly included in PA through past site characterization data and ongoing monitoring data.

The CCDFs for the CRA-2004 were developed using the same methodology that was used for the CCA and the CCA Performance Assessment Verification Test. The only changes were in the values of some parameters and modeling assumptions, for example, changes in parameter values and probability distributions carried forward from the CCA Performance Assessment Verification Test. Many of these changes were related to inventory changes, but some were related to modeling assumption changes (Leigh, et al. 2005, Section 2.0). However, the basic process the DOE used to develop the parameter information and sample the parameters did not change from the CCA methodology.

There were changes to several of the parameters from the CRA-2004 PA for the CRA-2004 Performance Assessment Baseline Calculation (Leigh, et al. 2005). The CRA-2004 Performance Assessment Baseline Calculation sampled 56 parameters whose values were obtained through random sampling in the PA; there were three sampled parameters added and fifteen removed after the CRA-2004 PA. As in the previous two compliance PAs, a total of 300 CCDFs (100 for each of the three replicates) were constructed for total normalized releases. Normalized release results for 10,000 simulations of possible futures were used to calculate each of the 300 CCDF curves. In addition, the CCDFs were provided for individual pathways. Figure 34 presents the mean CCDFs for total normalized release and for the normalized releases resulting from cuttings and cavings, spallings, and direct brine release. The mean CCDF for subsurface releases resulting from groundwater transport is not shown because those releases were less than 10^{-6} EPA units, so the CCDF cannot be shown at the scale of this figure (DOE 2004).

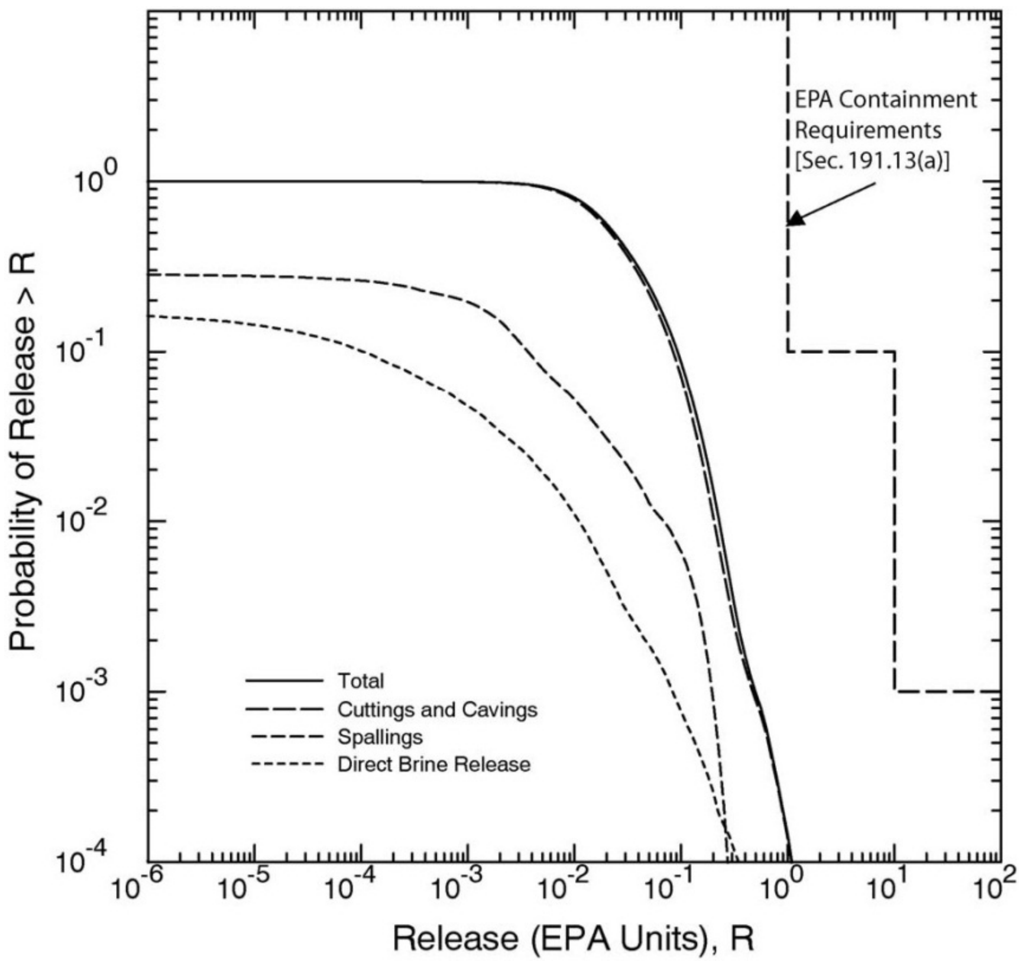


Figure 34. WIPP CRA-2004 PA, mean CCDFs for total normalized releases and specific release modes

Carrying forward determinations made in the CRA-2004 Performance Assessment Baseline Calculation for treatment of uncertain parameters, the CRA-2009 PA sampled the same 56 parameters.

Figure 35 shows the mean CCDFs for total normalized releases and for each of the specific release modes that are components of the total, along with the overall mean total releases for CRA-2009, CRA-2004, and the 1996 CCA. The contributions to total releases for each release pathway in the CRA-2009 PA are the same as those observed in the CRA-2004 Performance Assessment Baseline Calculation (Dunagan 2008). It is interesting to note that, consistent with the CRA-2004 Performance Assessment Baseline Calculation results, groundwater releases due to transport through the Culebra for occurred only in second replicate of the PA calculations. No transport releases larger than 10^{-6} EPA units occurred in the first and third replicates. Normalized transport releases for the CRA-2009 PA are qualitatively similar to the CRA-2004 Performance Assessment Baseline Calculation results, in that only the second replicate exhibited releases significantly larger than the numerical error inherent in the transport calculations. Overall, the mean releases for the second replicate of the 2004 and 2009 PA analyses were quite similar, and the numbers of vectors that had releases were identical, with only a slight increase in the CRA-2009 PA due to the increase in the drilling rate (Dunagan 2008). The similarity of each of the three replicates in the PA—and the consistency of results between the CRA-2009 PA and CRA-2004 PA, as well—demonstrate the stability of the WIPP PA results.

The overall mean CCDFs of the CCA, CRA-2004, and CRA-2009 PAs shown in the lower right of Figure 35 illustrate the wide margin of compliance of the predicted releases with respect to the release limits.

The PAs supporting both the 2004 and 2009 CRAs continued to show that, for most probabilities, cuttings and cavings are the most significant pathways for release of radioactive material to the land surface and that release by spallings and subsurface transport in the Salado or Culebra make essentially no contribution to total releases (Clayton, Dunagan, et al. 2008). For all WIPP compliance assessment PAs, the resulting CCDFs for total normalized releases have been within regulatory limits.

Cuttings, cavings, and direct brine releases account for the majority of the total releases estimated in the CRA-2009 PA, as they had in the CRA-2004 Performance Assessment Baseline Calculation. Sensitivity and uncertainty analyses showed that, in both the CRA-2009 PA and the CRA-2004 Performance Assessment Baseline Calculation, uncertainty in total normalized releases is largely due to uncertainty in waste shear strength (Kirchner 2008). The volumes of cuttings and cavings are primarily controlled by shear strength. The “solubility multiplier” applied to represent uncertainty in solubilities for all actinides in the +III oxidation state (Xiong, Nowak and Brush 2005) remained the second most dominant parameter contributing to variability in total releases in all replicates (Kirchner 2008). Solubility of actinides impacts their concentration in direct brine releases. The variability in total releases explained by the waste shear strength in the CRA-2009 PA dropped significantly from previous levels. In the CRA-2009 PA, waste shear strength only accounted for about 81% of the total variability in total releases, whereas in the CRA-2004 Performance Assessment Baseline Calculation it accounted for 88% of the variability (Kirchner 2008). This decrease is due to the increase in direct brine releases, which increases the contribution of direct brine releases to total releases (Clayton, Dunagan, et al. 2008).

4.7 PA Analyses for WIPP Disposal of Greater-Than-Class-C Low-Level Radioactive Waste (2008–2010)

In a set of analyses (SNL 2008a, 2008b, and 2010) separate from the existing WIPP mission, SNL supported DOE in its EIS analyses of alternatives for disposal of greater-than-Class-C (GTCC) low-level radioactive waste (LLW). Under current radioactive waste management laws and regulations, which limit the use of WIPP to the disposal of TRU waste generated from defense-related activities, disposal of GTCC wastes would not be permitted at WIPP, but it was considered among several alternative locations and disposal methods in DOE’s EIS (DOE 2011).

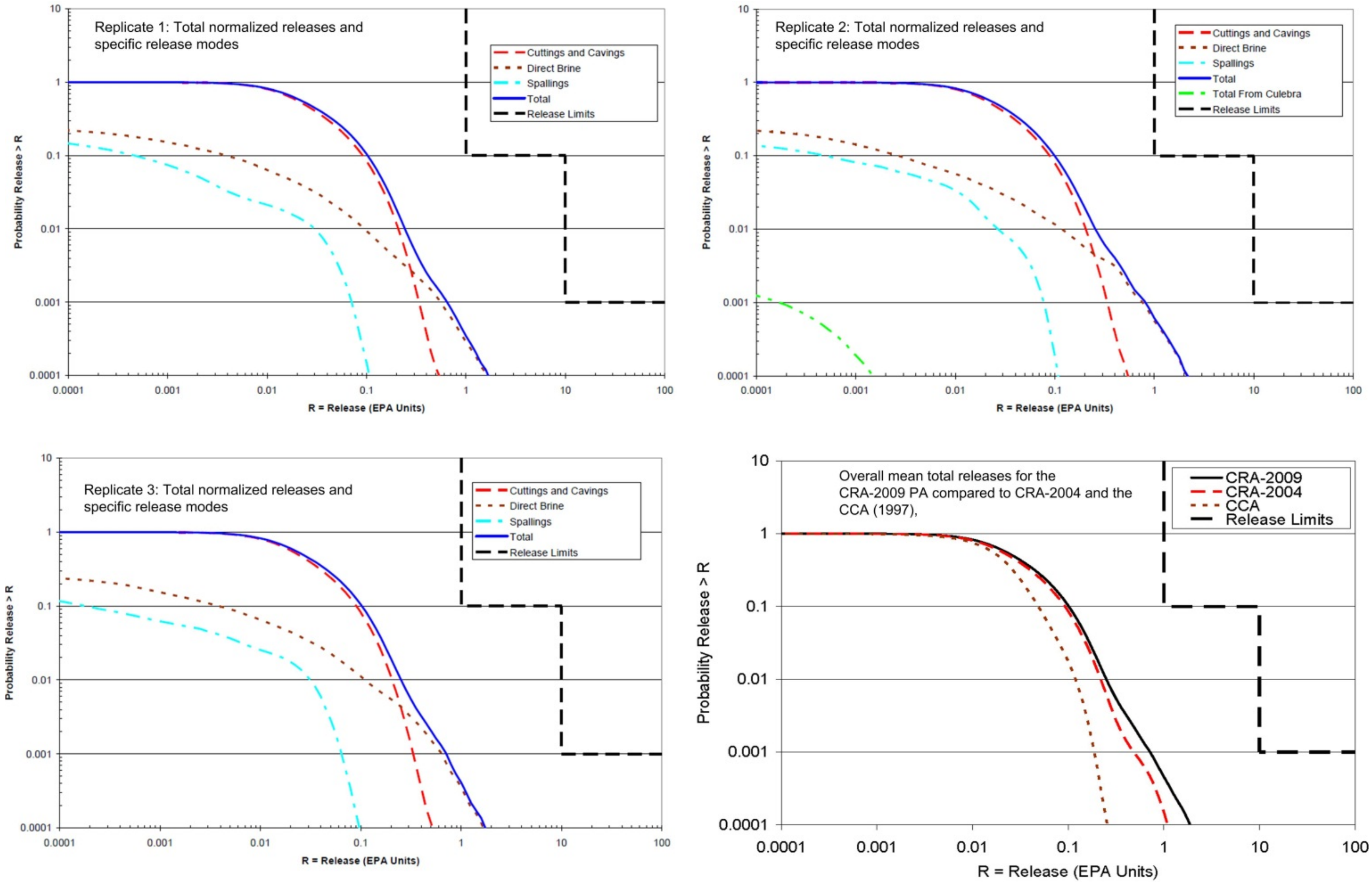


Figure 35. **WIPP CRA-2009 PA, mean CCDFs for total normalized releases and specific release modes presented separately for each of the three replicates and comparison of overall total releases for PAs for CRA-2009, CRA-2004, and the 1996 CCA**

Background—Greater-than-Class-C (GTCC) low-level radioactive waste (LLW) is LLW waste with radionuclide concentrations that exceed the NRC's limits for Class C LLW. These wastes are not generally acceptable for near-surface disposal; the disposal methods must be different and, in general, more stringent than those specified for Class C low-level-radioactive waste. NRC regulations require GTCC waste to be disposed of in a geologic repository, unless NRC approves an alternative disposal method. There is currently no repository approved for disposal of GTCC LLW.

As part of the responsibilities assigned to the DOE in the Low-Level Radioactive Waste Policy Amendments Act of 1985, the DOE has begun the EIS process for development of a disposal capability for GTCC low-level-radioactive waste and "GTCC-like" waste generated or owned by DOE and containing concentrations of radionuclides similar to GTCC LLRW. In DOE's draft EIS for disposal of GTCC LLW (DOE 2011), the GTCC and GTCC-like wastes are generally characterized as being in three groups:

1. Sealed sources (small quantities of highly radioactive materials enclosed in metal containers, with industrial uses including well logging and weld and pipeline inspection and medical uses including diagnosis and treatment of illnesses such as cancer and for sterilization of medical products). In the DOE's EIS analyses, this category was estimated to be 2,900 m^3 and have a total radionuclide activity of 2.0 million curies (2.0 MCi).

2. Activated metals that result from decommissioning of nuclear reactors, consisting of portions of the reactor assembly and other components near the nuclear fuel that were activated by neutrons during reactor operations, producing high concentrations of radionuclides. In the EIS analyses, the future inventory of activated metals was estimated to be 2,000 m^3 in volume and have a total radionuclide activity of 160 MCi.

3. Other wastes, most of which are transuranic wastes originating from nondefense activities and are therefore not currently authorized for disposal at the WIPP. In addition, this miscellaneous group includes contaminated equipment (e.g., glove boxes), debris, scrap metal, filters, resins, soil, and solidified sludges from domestic production of medical radioisotopes, radioisotope thermoelectric power systems used in space exploration and national security, and environmental cleanup of radioactively contaminated sites. In the EIS analyses, the future inventory of these other wastes was estimated to be 6,700 m^3 in volume and have a total radionuclide activity of 1.3 MCi.

Among the alternatives considered in the EIS is disposal of GTCC and GTCC-like wastes in the WIPP geologic repository. The other disposal locations evaluated included the Hanford Site in Washington; INL; Los Alamos National Laboratory; the Nevada National Security Site; the Savannah River Site; and two locations in southeastern New Mexico near WIPP. The disposal methods evaluated for these locations included intermediate-depth borehole disposal (similar to the Greater Confinement Borehole disposal approach described in Section 6), an enhanced near-surface trench, and an above-grade vault.

PA Analyses and Results—The PA done by SNL to analyze potential disposal of GTCC and GTCC-like waste at WIPP used the CRA-2004 Performance Assessment Baseline Calculation, including its FEPs and scenario development and models, as a baseline, and updated it to reflect the additional radionuclide inventory and activities and the room space required for the additional waste.

Of necessity, a number of simplifying assumptions were made in the analysis, but calculations were performed at a level of detail commensurate with the data required for EIS evaluations. In the WIPP CRA calculations, contact-handled wastes and remote-handled wastes are tracked separately for some analyses because release mechanisms are different for waste placed on the floor and the remote handled waste placed in the walls. But the EIS PA calculations assumed that GTCC LLW and GTCC-like waste would be emplaced using shielded containers using floor space in WIPP even though some of the waste is denoted as remote handled. Thus, the radionuclide activity of the GTCC LLW and DOE GTCC-like waste was tracked with the contact handled waste.

The radionuclide screening analysis, which screened out radionuclides with half-lives of less than 20 years and radionuclides that did not contribute at least 0.1% of the total activity resulted in the inclusion of 13 radionuclides: ^{14}C, ^{59}Ni, ^{63}Ni, ^{90}Sr, ^{137}Cs, ^{233}U, ^{234}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{241}Am, and ^{244}Cm.

The PA results demonstrated that the inventory of GTCC LLRW and GTCC-like wastes could be disposed of in WIPP in compliance with existing regulatory requirements (i.e., 40 CFR Part 191). The CRA-2004 WIPP repository has no significant groundwater releases and adding the GTCC LLW and DOE GTCC-like waste to the WIPP repository does not change that result: under undisturbed conditions, releases are essentially zero.

As in the baseline WIPP PA results for 40 CFR Part 191 compliance, only in the disturbed performance scenario, where the inadvertent human intrusion (i.e., drilling) penetrates the repository and causes releases, is there any significant probability of radionuclide release. These results are shown in Figure 36 (SNL 2010, Figure 2; DOE 2011, Figure 4.3.4-4) , which compares the WIPP Baseline performance (i.e., the CRA-2004 Performance Assessment Baseline Calculation results) with PA results adding GTCC and GTCC-like LLW from waste already in storage or expected to be generated from facilities already in operation ("Group 1") as well as the additional waste that may be generated from other proposed actions ("Group 2").

In the disturbed performance scenario, the incremental increases in the normalized releases to the inadvertent human intruder from adding the GTCC LLW and DOE GTCC-like waste to the WIPP repository are not substantial enough to jeopardize compliance with 40 CFR Part 191 release limits. As seen in Figure 36, at the 10% probability level, the mean total normalized release increased from 0.09 to 0.20, while at the 0.1% probability level, the mean total normalized release increased from 0.57 to 1.99, which are both well below 40 CFR Part 191 release limits. The increase is mainly due to the increase in the normalized radionuclide concentration for brine release, while the increase in the waste disposal area contributed as well (SNL 2010).

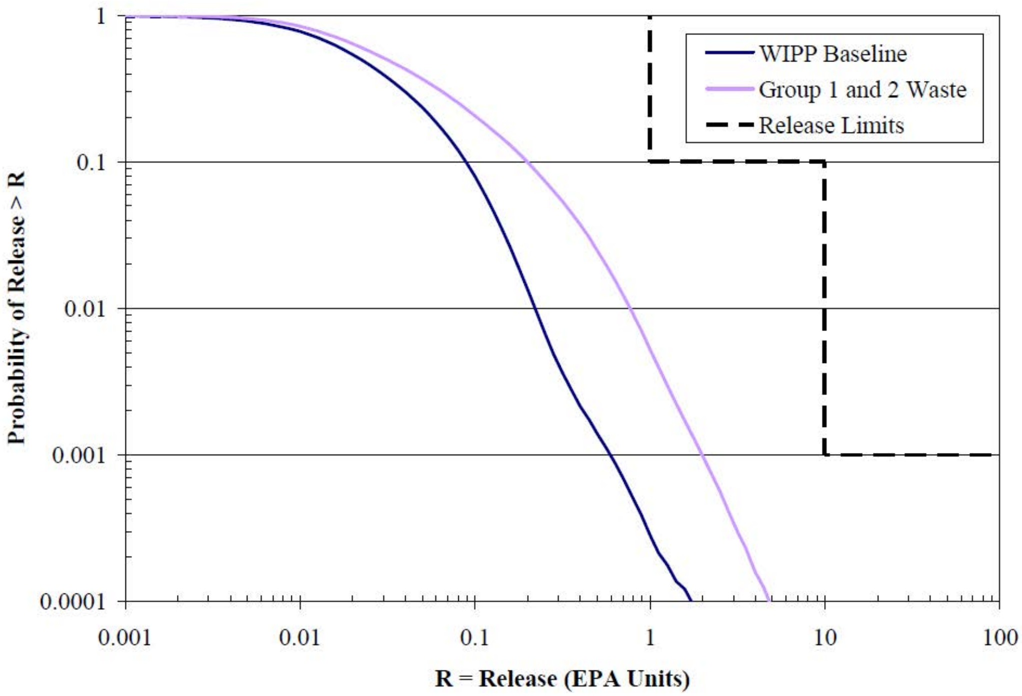


Figure 36. 2010 PA results for greater-than-Class-C LLW disposal EIS, for the WIPP disposal alternative, based on CRA-2004 Performance Assessment Baseline Calculation PA models and updated greater-than-Class-C LLW inventory

4.8 Significance of WIPP PA in the Historical Development of the PA Methodology

Beginning with the work of Bingham and Barr (1979) supporting the WIPP EIS (DOE 1980), SNL's WIPP PA studies developed the science of the FEP list and the FEP screening methodology. The significant effort that went into the development of the FEPs list for the 1989 demonstration PA, and the list of relevant FEPs remained fairly constant following the 1989 PA. When it was formally updated with a complete re-evaluation of the FEPs list for the 1996 CCA PA, the results were fundamentally consistent. The initial 1989 FEPs list was used by program management to scope the size and length of the research and development phase of the WIPP program and then, later, adjust planning as appropriate based on the iterative PA results. The importance of establishing a comprehensive FEPs list to serve as a stable foundation for the PA process and to inform programmatic planning and decision-making cannot be understated. For comparison, while WIPP had a rigorously developed FEPs list in its first preliminary PA iteration that helped to focus and prioritize research and to provide consistency in the iterative PA process, with minor adjustments to FEPs as necessary to account for new data and information from research and site characterization, the YMP PA process up until TSPA-SR tended to view FEPs and scenario development more as traditional hypothesis-testing, with scenarios being selected and exercised until challenged by PA results or new data inconsistent with the scenarios, which disrupted the research and development program and PA activities as well.

PAs for WIPP also advanced the science of conducting uncertainty and sensitivity calculations on both subsystem and system models to identify critical parameters for further study. Building on experience from the iterative SDP PAs, the SNL WIPP PA program further demonstrated the value of an iterative PA approach in supporting management of a large science and engineering program.

The implementation of the SPM assisted in the transition from site and laboratory investigations and technical analyses to a demonstration of regulatory compliance. SPM brought project scientists together and evaluated the effects of proposed technical activities on project budget, schedule, and compliance with U.S. EPA radioactive waste disposal regulations. The results of SPM were used to inform the experimental program to ensure that data and other information was focused on assessing the adequacy of the technical baseline for certification. As a result, new technical programs were initiated, some existing programs were refocused on reducing specific uncertainties, and other programs were cancelled when the uncertainties they addressed were determined to be acceptable without further data collection. SPM also served to inform stakeholders of the experimental program supporting the certification and to gain their confidence in the adequacy of the technical baseline.

Building on experience from SDP PAs, the WIPP PA program identified the need for a total coupled set of codes, allowing iterative deterministic and probabilistic calculations and developed and implemented a system of linked codes called the Compliance Assessment Methodology Controller (CAMCON). A prototype of CAMCON was implemented in the 1989 demonstration PA, and the full version was utilized in the first full PA conducted in 1990. The CAMCON system controller was later applied in the PAs for deep geologic disposal of DOE-owned HLW and SNF being stored at INL (see Section 7). The codes implemented via CAMCON were all using the first principles of physics (i.e., they contained and solved the relevant equations of, for example, transport, solubility, chemistry, and other processes, and they solved those equations for each iteration and stored the results for later review). This approach allowed the user to change an individual code very simply and directly, rather than having to rewrite much of the master code system. As the WIPP program matured each of the physics codes were streamlined by using fixed values for the unimportant individual parameters and the unimportant subroutines, rather than using sampled values. This iterative process allowed the PA analysts to run a set of probabilistic analyses in a very short time, and later, when CAMCON was applied in PAs for INL HLW, it allowed many sets of probabilistic analyses to be run for sensitivity analyses of INL wastes, with great efficiency and minimal cost.

WIPP PA also identified potential advantages in having two system models, one for detailed studies of the importance of parameters and one for the streamlined calculations needed for compliance applications, and it showed that control and transparency of the data inputs to the calculations is critically important.

The WIPP PAs, along with the field and laboratory testing and facility engineering, led to the first—and, to date, the only—successfully licensed deep geologic repository for radioactive waste in the U.S. WIPP has now been operating for 12 years.

5. YUCCA MOUNTAIN TOTAL SYSTEM PERFORMANCE ASSESSMENT (1984–2010)

Yucca Mountain, located approximately 161 km northwest of Las Vegas, Nevada, at the western boundary of the Nevada National Security Site, was proposed to be the nation's first repository for the disposal of military and civilian SNF and HLW. Figure 37 illustrates the final design concept for Yucca Mountain, after its evolution over more than 20 years guided in large part by iterative PA. On June 3, 2008, the DOE Office of Civilian Radioactive Waste Management submitted its license application (DOE 2008) to the NRC for authorization to construct the Yucca Mountain repository. A critical component of that license application was the Yucca Mountain TSPA led by Sandia National Laboratories. A timeline of the Yucca Mountain program, shown in the context of other contemporary developments in PA radioactive waste management, is shown in Figure 38.

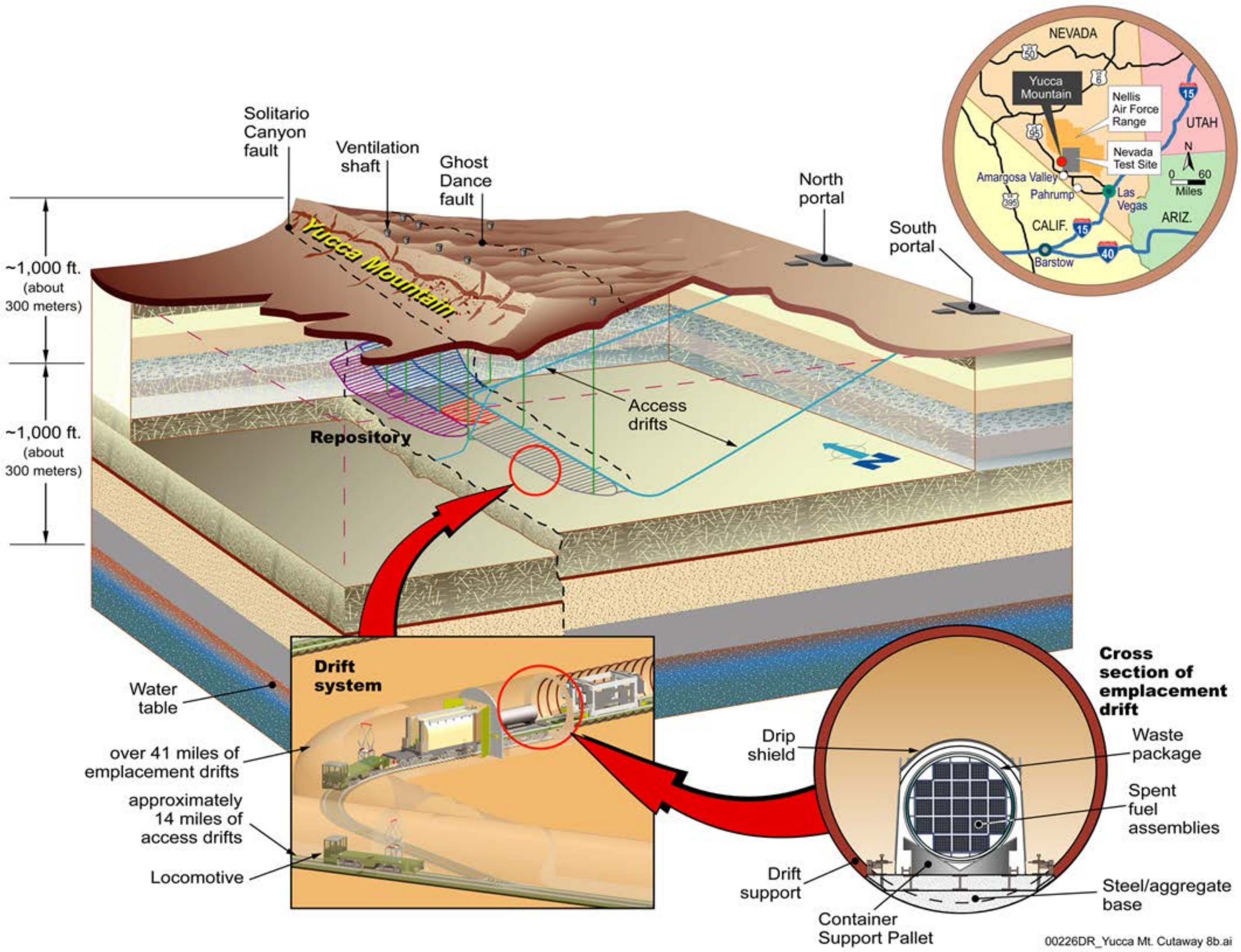


Figure 37. Schematic illustration of the Yucca Mountain repository, showing the mature design concept

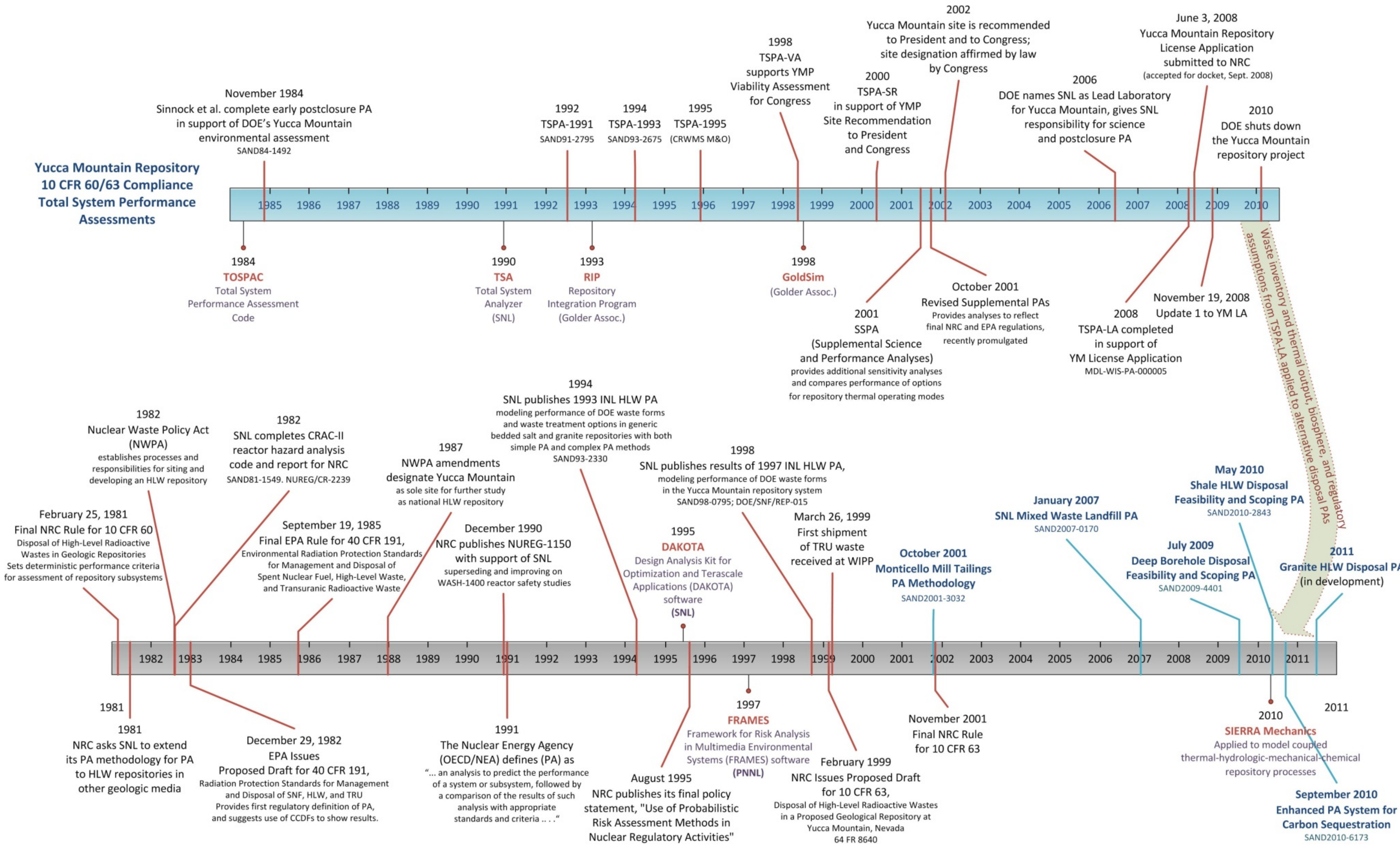


Figure 38. Yucca Mountain repository project timeline

In addition to the 2008 TSPA that the DOE Office of Civilian Radioactive Waste Management submitted to the NRC for licensing (TSPA-LA) (DOE 2008, SNL 2008c), SNL was involved in multiple iterations of the Yucca Mountain PA for nearly 30 years (Table 3), including the environmental assessment for Yucca Mountain (DOE 1986), the initial TSPA, also known as TSPA-1991 (Barnard, Wilson, et al. 1992), the second iteration TSPA, TSPA-1993 (Wilson, Gauthier, et al. 1994), TSPA-1995 (CRWMS M&O 1995), the viability assessment (DOE 1998), and the site recommendation (CRWMS M&O 2000, DOE 2002a, DOE 2002b). SNL lead only the earliest and latest Yucca Mountain PAs, but the development of each major PA is described here for completeness.

In 2010, the President decided that Yucca Mountain was no longer an option for the long-term management of nuclear waste in the U.S., and the DOE moved to withdraw the repository license application and shut down the Yucca Mountain Project. Had it continued, the NRC licensing requirements for the license application (including the TSPA) would have called for the TSPA to be iterated again, as appropriate, at key licensing milestones or at program changes such as application for operational licenses. As at WIPP, where performance assessment is periodically updated for EPA compliance recertification, TSPA would have continued iteratively at Yucca Mountain until repository closure.

5.1 Background

Performance assessment at Yucca Mountain began in the early 1980s, when Hunter, Barr, and Bingham (1982, 1983) conducted the earliest studies of potential scenarios applicable to long-term performance of a repository at Yucca Mountain had been carried out, and when Sinnock, Lin, and Brannen (1984) prepared a preliminary forecast of repository performance, using deterministic methods, in support of DOE's environmental assessment for Yucca Mountain, then one of five candidate sites being considered to host a repository.[7] These early studies were completed before the Topopah Spring Tuff unit had been selected by DOE to be the proposed repository horizon, after which Ross (1987) conducted a study of scenarios potentially applicable to a host location in the Topopah Spring Tuff that was subsequently expanded on in the section addressing total system performance assessment in the DOE's 1988 Yucca Mountain Site Characterization Plan (DOE 1988).

As with prior PAs, the goals for PAs at Yucca Mountain were iterative, each building on the conclusions of prior PAs. The Sinnock, Lin, and Brannen (1984) preliminary forecast supported DOE's environmental assessments of candidate repository sites, but also helped to identify areas of importance for site investigation, shaping the program planning after the site was chosen by Congress in 1987 as the sole site for further characterization. After Yucca Mountain was designated as the sole site for investigation, single-scenario deterministic analyses of the Performance Assessment Calculational Exercises for 1990, known as "PACE-90" (Barnard and Dockery 1991) contributed both to code development and to preliminary assessment of the total repository system, helping to establish goals for future Yucca Mountain total system PAs with

[7] Contemporaneous to the PA studies of postclosure performance, SNL also prepared a preliminary safety assessment of preclosure operations (Jackson, et al. 1984), based on the assessment methodology typical for nuclear power plants.

Table 3. History of DOE Yucca Mountain TSPAs

TSPA Iteration	Purpose and Summary of Key Results
Preliminary Yucca Mountain PA (Sinnock, Lin and Brannen 1984)	• Provided information for DOE's Yucca Mountain Environmental Assessment
TSPA-1991 (Barnard, Wilson, et al. 1992, Eslinger, et al. 1993)	• Demonstration of TSPA approach. • Included scenarios for human intrusion and volcanism • Identified importance of uncertainty in unsaturated zone flow paths.
TSPA-1993 (Wilson, Gauthier, et al. 1994)	• Improved unsaturated zone and saturated zone models. • Evaluated alternative models. • Included early models for thermal processes and engineered barrier system. • Identified importance of uncertainty in thermal hydrology, unsaturated zone flow, and corrosion of engineered materials.
TSPA-1995 (CRWMS M&O 1995)	• Incorporated new science and design; evaluated alternative models of flow about package. • Identified importance of process models to waste package degradation, seepage, unsaturated zone and saturated zone transport.
1998 TSPA-VA (CRWMS M&O 1998, DOE 1998)	• Supported the 1998 Viability Assessment. • Models based on best current information. • Ranked importance of uncertainty in each of the major components for 10,000, 100,000 and 1 million years. • Emphasis on seepage, water chemistry, corrosion, and saturated zone.
2000 TSPA for Site Recommendation (TSPA-SR) (CRWMS M&O 2000)	• Modeling system used fully qualified inputs. • Conservative approach adapted for some components. • Importance of volcanism identified. • Conservative treatments of uncertainty complicated realistic understanding.
FY 2001 Supplemental Science and Performance Analyses (SSPA) (Bechtel SAIC Company 2001a, 2001b)	• Analyzed alternative thermal loading strategies • Included more realistic treatment of uncertainty. • Incorporated new information since TSPA-SR. • Confirmed potential suitability. • Confirmed importance of volcanism and engineered barrier system performance for 10,000 years. • Insights into engineered barrier system and natural system effects on peak dose.
2001 Revised supplemental TSPA models supporting Site Recommendation	• Updated supplemental TSPA-SR model for the final EPA regulations and Final EIS (Bechtel SAIC Company 2001c) and for the final NRC regulations (Bechtel SAIC Company 2001d) to include new information, and revised regulatory boundary.
2008 TSPA-LA (SNL 2008c, DOE 2008)	• Provided the TSPA to fulfill the requirements of NRC's 10 CFR Part 63, demonstrating that the EPA's 40 CFR Part 197 environmental standards are met by the Yucca Mountain repository, in support of a successful DOE license application to the NRC for construction authorization. • Assessed peak dose for a performance period extended to 1 million years • Added seismic disruptive event. • Models updated to current information.

complete scenario analyses and stochastic calculations. Those Yucca Mountain total system PAs had the following stepwise goals, some established by the structure and methodology of PA itself, others directed by law and regulation:

1. **TSPA-1991** (Barnard, Wilson, et al. 1992): Develop an approach to deriving "abstracted" or simplified representations of complex repository processes and, secondarily, demonstrate that complex combinations of distributions of data can be assembled to provide a reasonable overall estimate of total system performance.
2. **TSPA-1993** (Wilson, Gauthier, et al. 1994): Provide feedback concerning the relative importance of specific site-characterization and design information. Secondarily, advance development of more defensible TSPA models for future use in a demonstration of compliance by enhancing the realism and representativeness of the analyses, incorporating new information and designs that had become available since TSPA-1991, testing the sensitivity of the predicted performance against various conceptual model and parameter uncertainties, and evaluating alternate measures of postclosure performance.
3. **TSPA-1995** (CRWMS M&O 1995): Focusing on the system components identified by TSPA-1993 as most significant to repository performance, (1) utilize what were believed to be still more representative conceptual models that built upon the assumptions employed in TSPA-1993, (2) incorporate more recent design information than was available for TSPA-1993, (3) utilize the most recent site information and models, and (4) evaluate an engineered barrier system release performance measure, as well as other alternative measures of total system performance, including cumulative radionuclide releases, peak concentrations, or doses.
4. **TSPA-VA**, 1998 (CRWMS M&O 1998, DOE 1998): Provide the TSPA required by the Energy and Water Development Appropriations Act of 1997 as part of the Viability Assessment of the Yucca Mountain site, and continue iterative application of the PA process in quantifying the significance of key system components to prioritize program science and engineering efforts.
5. **TSPA-SR**, 2000 (CRWMS M&O 2000), and supplemental performance analyses, 2001 (Bechtel SAIC Company 2001a, 2001b, 2001c, 2001d): Provide a TSPA required by DOE's site suitability regulation, 10 CFR Part 963, demonstrating a preliminary assessment of compliance with NRC's licensing criteria in 10 CFR Part 63 and EPA's environmental standards in 40 CFR Part 197.
6. **TSPA-LA**, 2008 (SNL 2008c, DOE 2008): Provide the TSPA to fulfill the requirements of NRC's 10 CFR Part 63, demonstrating that the EPA's 40 CFR Part 197 environmental standards are met by the Yucca Mountain repository, in support of a successful DOE license application to the NRC for construction authorization for the repository.

In addition to the comprehensive TSPAs performed and documented for the proposed repository system at Yucca Mountain, many subsystem analyses were performed. As more information about the site and design components of the potential repository system became available, these TSPA analyses evolved into progressively more complex representations of the system. The representation of some of the elements of the total system analysis included in each of the iterations has remained fundamentally the same, although the models and parameters have been revised and refined each time. However, as the collective scientific understanding of the FEPs specific to Yucca Mountain has progressed, the representation of certain process areas has been significantly updated. For example, in earlier assessments of Yucca Mountain, groundwater was believed to flow downward, almost exclusively in the rock matrix. This understanding resulted in extremely slow flow through the entire system, and thus to very low cumulative releases during the PA performance period. Sensitivity studies at higher fluxes suggested it to be an important parameter, which focused program attention on gathering better data on unsaturated zone flow. Having determined in the mid-1990s that flow in fact occurred in fractures within the rock as well as the matrix, the understanding of unsaturated-zone groundwater flow evolved and prompted changed in the TSPA models and the representation of the proposed repository system.

As the repository design evolved in response to experimental data and an improved understanding of site characteristics, the Yucca Mountain TSPA evolved to reflect those changes. For example, the repository design concept changed from a small waste package, vertically emplaced in the drift floor, as envisioned originally in the DOE's 1988 Site Characterization Plan (DOE 1988), to a significantly larger waste package emplaced horizontally in the drift. For the TSPA, this change required additional models and analyses (or improvements to existing models) of drift seepage and structural stability with time, the thermal and chemical environment in the drift, performance of alternative container material, and repository performance effects of backfill, emplacement pallets, and invert materials. TSPA-1993 (Wilson, Gauthier, et al. 1994) analyzed both emplacement approaches, evaluating the performance of each to help support DOE's design decision. Other notable changes in the evolution of the Yucca Mountain repository design include changes to the use of concrete drift liners, backfill materials, and drip shields, and a number of adjustments to the repository thermal loading approach to optimize performance of the waste package. All have all required accommodation by the TSPA models, and the iterative PA process helped to analyze design options and support decision decisions.

Changes in regulatory guidelines have also resulted in changes in the Yucca Mountain TSPA modeling activities. For instance, early TSPA analyses compared repository against the guidelines in 10 CFR Part 60 and 10 CFR Part 960, where the performance criteria considered cumulative releases to the environment. In the late 1990s, the site-specific regulations promulgated for Yucca Mountain in response to the Energy Policy Act of 1992 established a dose standard and a groundwater protection standard in place of release limits. The conversion to a dose standard resulted in increased significance of the saturated zone flow. When the performance standard only addressed cumulative release, details of saturated zone flow (e.g., dispersion, dilution) were relatively unimportant. But under a dose standard, radionuclide transport through the saturated zone is very important. A summary of the various features in each of the YMP TSPAs, expanded from a table developed during the TSPA-VA (CRWMS M&O 1998, Table 1-2), is provided in Table 4 to help show the evolution of the YMP TSPA models. In general, that evolution shows an increase in sophistication in the model representation, but in some cases reveals simplifications applied because of changes in understanding of the importance of a parameter or model (or changes in understanding of the parameter or system itself). This follows the evolution typical of iterative PA modeling and site characterization, where some models may increase in sophistication as detailed understanding of process increases as a result of site characterization while—simultaneously—the complexity of other models may tend to decrease as the unimportant components and data are removed from or simplified in the calculation as a result of sensitivity analyses.

Table 4. Summary of Yucca Mountain PA evolution from 1984 to 2008; features of TSPA

	Sinnock et al. 1984	PACE-90	TSPA-1991	TSPA-1993	TSPA-1995	TSPA-VA (1998)	TSPA-SR (2000-2001)	TSPA-LA
Infiltration	Up to 20 mm/yr	Min: 0.01 mm/yr Max: 0.5 mm/yr	0–39 mm/yr	dry: 0.5 mm/yr mean wet: 10 mm/yr mean	low: 0.01-0.05 mm/yr high: 0.5-2.0 mm/yr	dry: 7.7 mm/yr mean LTA: 42 mm/yr mean SP: 110 mm/yr mean	Present-day: 4.6 mm/yr mean Monsoonal: 12.4 mm/yr mean Glacial-transition: 18 mm/yr mean	Four scenarios (10^{th}, 30^{th}, 50^{th}, 90^{th} percentile) for each climate state.
Number of radionuclides	17	4	10	43 (direct) 8 (aqueous)	39	9	26	32 (including 2 included only as predecessors (sources) of ^{241}Am Subset of 12 used in barrier analyses
Time period(s) of evaluation	up to 100,000 years	up to 100,000 years	up to 100,000 years	up to 1 million years	up to 1 million years	up to 1 million years	10,000 years and 1 million years	10,000 years and 1 million years
Waste forms	CSNF	CSNF	CSNF	CSNF and HLW	CSNF and HLW	CSNF, HLW, DOE SNF (including naval SNF)	CSNF, HLW, DOE SNF (including naval SNF)	CSNF, HLW, DOE SNF (including naval SNF)
Distance to accessible environment	5 km	n/a	5 km	5 km	5 km 30 km	20 km	20 km (18 km considered in supplemental TSPA model for FEIS)	18 km
Saturated zone	Yes	n/a	Single composite medium	Multiple layers	Single composite medium	Effective continuum, 1-D, six stream tubes	3-D, effective continuum flow model with 3-D particle tracking transport modeling	3-D, single continuum flow model 3-D particle tracking transport modeling with advection, dispersion, sorption, and matrix diffusion; includes anisotropy
Stratigraphic discretization in the unsaturated zone	n/a	19 layers	5 layers	10 layers	5 layers	28 layers	32 layers	32 layers
Unsaturated zone flow model	1-D, matrix	1-D, 2-D, five codes	1-D, 2-D; equivalent continuum model, and weeps	1-D, 2-D; equivalent continuum model and weeps	2-D; equivalent continuum model	3-D; dual-permeability continuum model	3-D; dual-permeability continuum model	3-D; dual-permeability continuum model
Release model	n/a	Two water contact modes	Simple failure distribution for WP	Simple failure distribution for WP	Three alternative conceptual models in engineered barrier system	Diffusive, advective release from dripping and no dripping zones	Diffusive, advective release from dripping and no dripping zones	Diffusive, advective release from dripping and no dripping zones
^{14}C gaseous release	No	No	2-D steady state	2-D transient	No	No	No	Yes (incorporated in biosphere model)
Thermal effects	No	No	No	Dryout zone	Dryout zone	Mountain- and drift-scale thermal-hydrologic effects	Mountain-scale thermal effects, and drift-scale thermal-hydrologic effects.	Mountain-scale thermal effects and drift-scale thermal-hydrologic effects on the in-drift temperature and humidity. Drift-wall condensation effects were added during the first 3,000 years.
Near-field geochemistry	No	No	No	No	No	Limited	Abstraction of water chemistry predicted by 2-D drift-scale coupled THC models.	Near-field chemistry model representing thermochemical evolution of percolating water contacting the host rock and seeping into drifts.
Disruptive events	No	No	• Volcanism, • Human intrusion	• Volcanism, • Human intrusion	No	• Volcanism (two eruptive scenarios), • Seismicity, • Human intrusion, • Nuclear criticality	• Volcanism (eruptive and intrusive scenarios); • Human intrusion; • Also, nuclear criticality and water table rise addressed, but screened out • Seismicity included in nominal scenario	• Volcanism (eruptive and intrusive scenarios); • Seismicity (ground motion and fault displacement scenarios); • Human intrusion; • Early waste package and drip shield failure considered outside nominal scenario; • Nuclear criticality addressed, but screened out
Fracture flow	No	in equivalent continuum model	equivalent continuum model, weeps	equivalent continuum model, weeps	yes	Dual-permeability model	Active fracture model within the dual-permeability approach	Active fracture model within the dual-permeability approach
Dose	No	No	No	Drinking water and irrigation	Drinking water	Three receptor scenarios (subsistence farmer, resident farmer, resident); all pathways, including volcanic	One receptor (i.e., reasonably maximally exposed individual) defined by regulation; all pathways, including volcanic	One receptor (i.e., reasonably maximally exposed individual) defined by regulation; all pathways, including volcanic

	Sinnock et al. 1984	PACE-90	TSPA-1991	TSPA-1993	TSPA-1995	TSPA-VA (1998)	TSPA-SR (2000-2001)	TSPA-LA
Climate change	through range of fluxes	No	through range of fluxes	100,000-year random periods	100,000-year random periods	Three climate cycles: Dry, Long-term average climate, Super-pluvial climate	Three climate states (within 10,000-year analyses) • Present-day (next 600 years) • Monsoon (600–2,000 years) • Glacial-transition (2,000 years–10,000) Beyond 10,000 years, the model extends the glacial-transition climate for base-case simulations; a sensitivity study included a revised long-term climate model	Three climate states (within 10,000-year analyses) • Present-day (next 600 years) • Monsoon (600–2,000 years) • Glacial-transition (2,000 years–10,000) Beyond 10,000 years, climate is represented by a probabilistic distribution for a long-term average climate, as specified by 10 CFR 63.342(c)(2)
Uncertainty	through range of analysis parameters	No	in pdfs and flow models	in pdfs and flow models	in pdfs and flow models	in pdfs and flow models	Represented in parameter distributions (for parameters used directly in TSPA) and flow fields or derived abstractions.	Model included aleatory seismic uncertainty, and epistemic seismic degradation affecting package integrity and radionuclide release. Other uncertainties were represented as before, in parameter distributions and flow models.

The preliminary TSPAs were conducted by SNL. While TSPA-1993 was being conducted independently by SNL for DOE, there were other parallel PAs conducted by DOE and other organizations, as described below. From TSPA-1995 through the TSPA-SR, YMP PA was conducted by the DOE's Yucca Mountain project management and operating contractor (with the subcontracted support of individual contributors from SNL and other national laboratories). These TSPAs are described here in less detail than the SNL TSPAs. In 2006, following years of study at YMP, SNL was selected by the DOE Office of Civilian Radioactive Waste Management to be the lead laboratory for the management and integration of all Yucca Mountain scientific programs. SNL was selected to increase technical credibility with the scientific community, as well as with the regulators and stakeholders, and to develop and defend the TSPA.

5.2 Yucca Mountain PAs

At the conceptual level, the YMP PA is identical to the WIPP PA. The mathematical framework is much the same; however, the difference is in the details, or process. The Yucca Mountain TSPA models are formulated as "abstractions" from more detailed process models. The abstraction is a simplified/idealized model that reproduces or bounds the essential elements of the more detailed process models. The inputs for an abstraction may be those that form a subset of those required for a process model, or the abstraction maybe a response function derived from intermediate results. However, the abstracted form must capture uncertainty and variability. The abstractions must also be tested against process models to ensure their validity. The success of the abstraction process depends heavily on the effective integration of information from all elements of the project, including site characterization, design, and PA. Abstractions were used because the probabilistic/stochastic nature of the Yucca Mountain TSPA analyses creates a great demand for computationally efficient models. The intent of the abstraction process is to retain key aspects of process models while producing results usable in multiple realization probabilistic models.

5.2.1 Preliminary Estimates on Yucca Mountain Site Performance

SNL's earliest PA work related to Yucca Mountain was during the early 1980s, as DOE was investigating multiple candidate repository sites. Sinnock, Lin, and Brannen (1984) developed a preliminary assessment of site performance against NRC's general technical criteria for geologic repositories, which had just recently been promulgated, the DOE's then-draft rule for 10 CFR Part 960, and the EPA's recently proposed environmental standards for 40 CFR Part 191. Its deterministic calculations covered time periods up to 1 million years.

Sinnock, Lin, and Brannen (1984) used TOSPAC to perform the calculations needed to provide a preliminary estimate of site performance. TOSPAC was developed at Sandia National Laboratories to support characterization of the Yucca Mountain site for the DOE. Initially, it was intended to model, as simply as reasonable, all the systems of a geologic waste repository for high-level radioactive waste. TOSPAC was designed to combine known one-dimensional hydrologic and contaminant transport models within a controlling shell, following the computational advances developed in the MARINRAD control code that compiled the process models into a system level model for the SDP. It was written in FORTRAN 77 for portability and to be readily modified if needed. Later, it was retained for use in modeling unsaturated zone flow, contaminant source term, and saturated zone flow and transport. In TSPA-1991 and

TSPA-1993, TOSPAC was configured to run within the larger shell of the TSA, or Total System Analyzer.

The Performance Assessment Calculational Exercises for 1990, known as "PACE-90" (Barnard and Dockery 1991) contributed both to code development and to preliminary assessment of the total repository system. The first phase produced results from deterministic calculations for a nominal flow scenario only. Elements of these calculations were performed by SNL, Los Alamos National Laboratory, Lawrence Berkeley National Laboratory, Lawrence Livermore National Laboratory, and Pacific Northwest Laboratory. The results indicated that, if the parameters and models used in the calculations were valid, radionuclide transport through Yucca Mountain via liquid pathways is extremely slow. None of the models estimated release of nuclides to the saturated zone in amounts great enough to be of regulatory concern. However, in addition to further work to validate the models, sensitivity studies for the problem set indicated a need for additional numerical and analytic analyses, including the effect of transient liquid pulses through the system; the ability of fractures, matrix rock heterogeneities, or lateral discontinuities to form fast flow paths or flow barriers; the ability of potential fast paths to transport a significant amount of radioactive material; and the rates of flow and transport in the saturated zone to the accessible environment. The second phase of PACE-90 involved development of disrupted configuration problems, including those initiated with the occurrence of basaltic igneous activity and human intrusion. For both igneous activity and human intrusion, problems were defined that were subsequently identified as scenarios within the appropriate event tree. However, no calculations were performed based on these scenarios.

Instead, the PACE-90 calculations and scenario analyses were used as a basis for the more comprehensive TSPA calculations, which, unlike the single-scenario deterministic analyses of PACE-90, would comprise a stochastic study where ranges in geologic, hydrologic, source term, and other parameters were incorporated.

5.2.2 TSPA-1991

The first in the "comprehensive" TSPA studies conducted for the YMP was TSPA-1991 (Barnard, Wilson, et al. 1992, Eslinger, et al. 1993), constructed using PACE-90 as a basis. Its objective was to develop a framework for probabilistic total-system calculations and it was the first set of stochastic analyses for YMP. The repository design concept analyzed in TSPA-1991 was for the vertically emplaced waste container in backfilled emplacement drifts, as shown in Figure 39. The analyses were run in one- and two-dimensions using distributions of hydrogeologic parameters based on site and analogue data. Two conceptual models for unsaturated zone flow were analyzed: composite porosity that represented fracture-matrix equilibrium, and the "weeps" model that assumed flow occurred exclusively in the fractures. The calculations also included disturbances to the nominal system caused by basaltic volcanism, human intrusion, and climate change. The radionuclide inventory was expanded to include those nuclides prevalent in the inventory (plutonium, uranium, and americium isotopes), those expected to be important to dose (^{79}Se, ^{126}Sn), and ^{14}C to represent the gaseous component. Radionuclide transport from the waste package also included some near-field interactions. The saturated zone was also included explicitly and modeled out to the accessible environment at a 5 km distance. A simple drinking water dose was calculated, in addition to the cumulative releases.

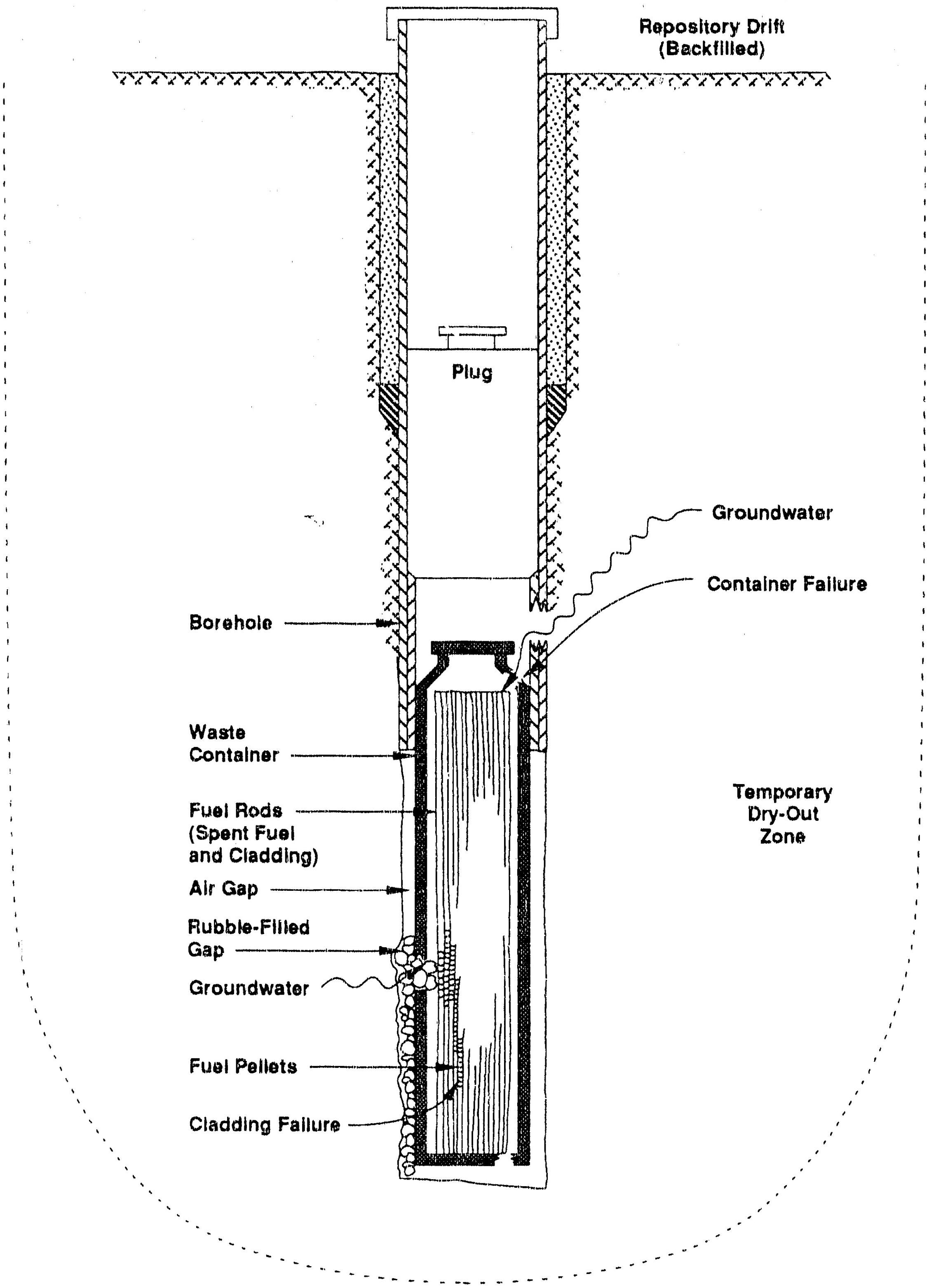


Figure 39. Conceptual illustration of a vertically emplaced waste container, showing some of the processes modeled in TSPA-1991

TSPA-1991 evaluated four scenarios: groundwater transport; gaseous transport, human intrusion, and basaltic volcanism (a fifth scenario addressing tectonic activity was addressed separately by Pacific Northwest Laboratory). At the time of TSPA-1991, the site-characterization process at Yucca Mountain was relatively immature; the FEPs occurring at the site were not well understood. It was recognized then that there may have been many important FEPs that had not yet been identified and that the FEPs could not be categorized definitively into expected processes and unexpected conditions or events.

No attempt was made to evaluate regulatory compliance, but qualitative comparisons were made with the EPA standard and NRC technical criteria. Results for TSPA-1991 (shown in Figure 40) suggested that aqueous releases for both the weeps and composite porosity models (top) and gaseous release using the weeps model (bottom) did not exceed the EPA's 40 CFR Part 191 cumulative release limits. Gaseous release, calculated with the composite porosity model, did exceed these limits. However, it was anticipated that a more realistic engineered barrier system model for waste packages and waste form failure (i.e., taking credit for gradual degradation of these engineered components) would reduce these releases to below the EPA limit. Releases due to human intrusion and to volcanism were also both well below these regulatory limits.

Several organizations contributed to the TSPA-1991. The problem definition was coordinated by SNL. SNL and PNL both performed PA calculations, although only SNL's work is discussed in this report. Lawrence Livermore National Laboratory and Lawrence Berkeley National Laboratory both contributed to the specification of the radionuclide source term by defining the waste-package failure modes and associated parameters. Los Alamos National Laboratory provided information on geologic events and features and the associated parameter distributions for the igneous-activity analysis. Los Alamos National Laboratory also provided information and parameter value distributions for the geochemical retardation modeled in the aqueous-flow analyses. The primary purpose of the SNL TSPA effort was to attempt to develop an ability to derive "abstracted" representations of the complex processes that contribute to the behavior of a repository system. Such abstractions are essential to the probabilistic modeling required for examining compliance with repository regulations.

The TSPA-1991 was a demonstration of the abstraction method and the use of the results in estimating the behavior of a total repository system. The TSPA-1991 demonstrated the ability to abstract complex models for use in a broader application. The CCDFs generated produced results sensitive to the processes at Yucca Mountain and consistent with work done using other models and techniques.

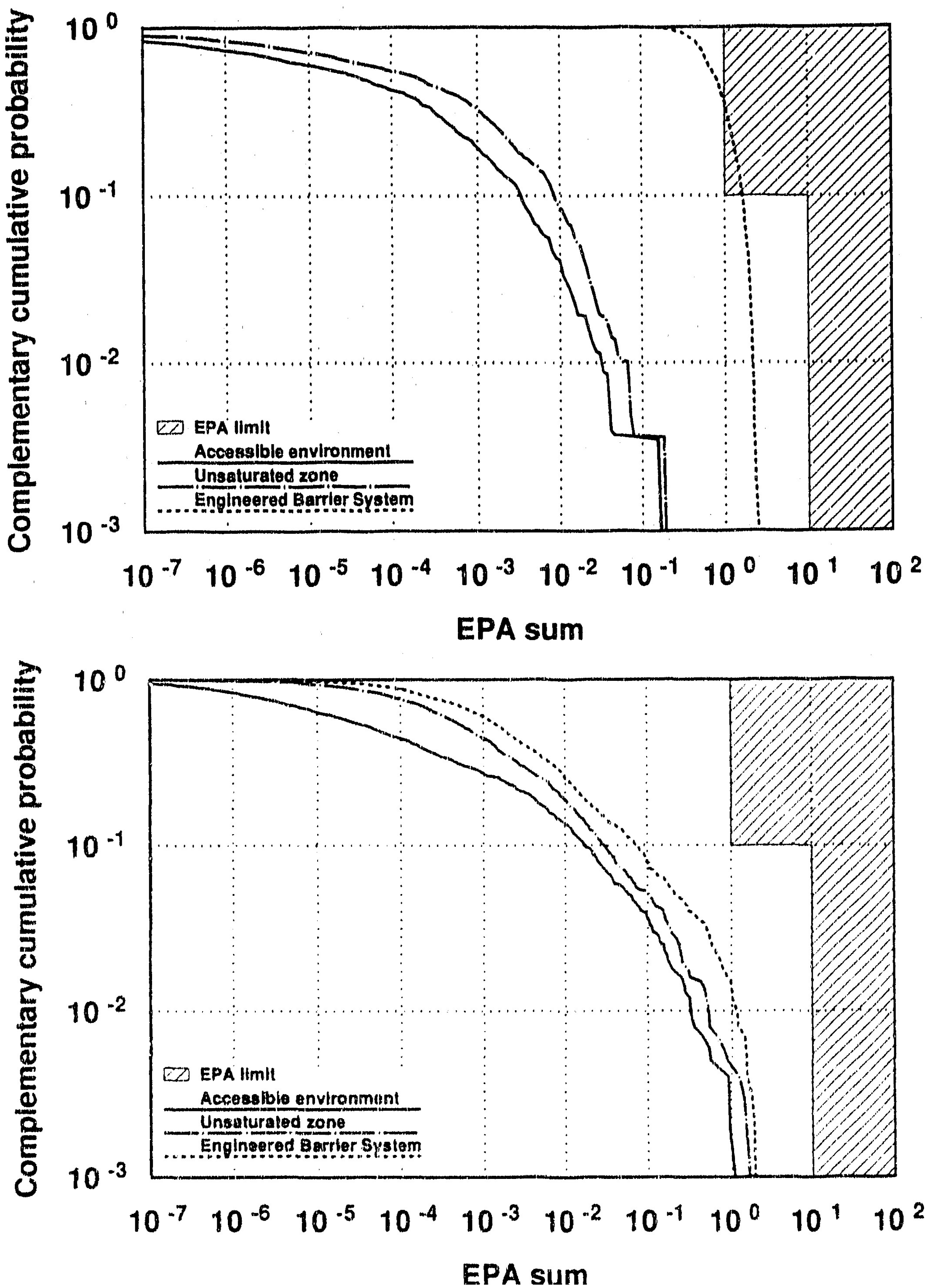


Figure 40. TSPA-1991: Conditional CCDFs for aqueous releases from the engineered barrier system, from the unsaturated zone, and to the accessible environment, calculated using the composite-porosity flow model (top) and weeps model of flow (bottom)

5.2.3 TSPA-1993

The primary purpose of TSPA-1993 (Wilson, Gauthier, et al. 1994) was to provide feedback concerning the relative importance of specific site-characterization and design information. Its secondary goal was to make progress in developing more defensible TSPA models for use in a demonstration of compliance.

There were a number of enhancements and revisions of the TSPA-1991 models and information including (Wilson, Gauthier, et al. 1994, 1-5, 1-6):

- A more geohydrologically representative model of the repository
- A three-dimensional geostatistically correlated stratigraphy
- An expanded hydrologic data set, explicit inclusion of wetter future climates
- Discrete modeling of individual stratigraphic units in the saturated zone
- Modification of retardation and sorption parameters
- Introduction of thermal dependence, spatial and temporal variation in fracture apertures (in the weeps model)
- Inclusion of waste package failure modes due to corrosion and dry oxidation
- Updated waste form dissolution and oxidation models
- Analysis of both the original vertical Site Characterization Plan containers and the horizontal multipurpose canisters, and inclusion of both spent fuel and vitrified waste.

The goals of TSPA-1993 were to (1) enhance the realism and representativeness of the analyses, (2) incorporate new information and designs that had become available since the completion of TSPA-1991, (3) test the sensitivity of the predicted performance against various conceptual model and parameter uncertainties, and (4) evaluate alternate measures of postclosure performance. The analyses, aimed at identifying the key assumptions and the sensitivity of the results to those assumptions, had eight major objectives:

1. Incorporate thermal dependency on individual processes and parameters
2. Evaluate the effects of alternate thermal loads
3. Evaluate the effects of alternate waste package designs
4. Evaluate alternate measures of total system performance
5. Incorporate new site and design information
6. Incorporate a more representative inventory, including high-level waste
7. Conduct sensitivity analyses to identify the key processes and parameters
8. Provide guidance to site characterization and design activities.

The human-intrusion analyses in TSPA-1993 consider a broad suite of 43 radionuclides. Nominal case and indirect volcanic effects consider 8 radionuclides, chosen for their transport characteristics (low retardation) or their potential contribution to individual dose. The waste form container designs had not been finalized, and two container types, shown in Figure 41, were analyzed in TSPA-1993: a smaller, vertically emplaced container proposed in the Site Characterization Plan conceptual design of 1988, and a larger, horizontally placed “in-drift container” approximating the multipurpose container waste package that was being considered as an option. Four combinations of containers and thermal loadings were examined in TSPA-1993.

A 57-kW/acre repository with vertically emplaced containers is the baseline analysis case for TSPA-1993, and most like the design described in the SCP and evaluated in TSPA-1991. In addition, the analyses considered a 114-kW/acre repository with vertically emplaced containers, a 57-kW/acre repository with horizontal containers, and a 114-kW/acre repository with in-drift containers.

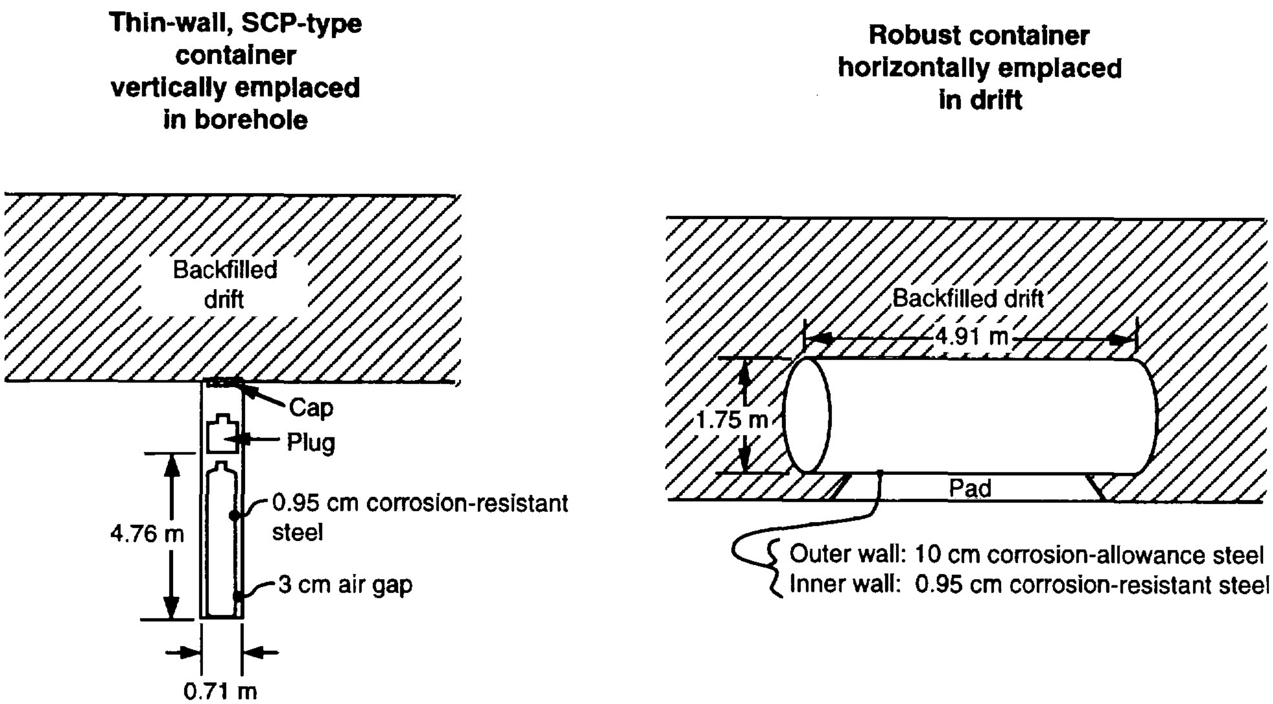


Figure 41. The two waste container types considered in TSPA-1993; the original vertically emplaced container concept (left), and an early concept for a horizontally emplaced waste package (right)

At the time of TSPA-1993, the National Academy of Sciences (NAS) was evaluating the appropriateness of a dose-based standard for Yucca Mountain (Section 1.3.1.1). Therefore, the performance measure was itself a primary issue requiring an assumption to be made in the PA. Another assumption arising from the lack of a standard was the time period of regulatory concern.

The SNL PA also concluded that regulatory change could lead to significant changes in program priorities for site characterization. A performance measure based on individual dose for the time period of regulatory concern, as in proposed 40 CFR Part 191 (58 FR 66398), would require additional characterization of the biosphere. A longer time period would lead to more emphasis on determining radionuclide release rates (Wilson, Gauthier, et al. 1994, ES-22). Therefore, assumptions were made about regulations, such as points of compliance for dose calculations, and CCDFs were produced for aqueous release as well as individual dose measures for time periods of 10,000, 100,000, and 1 million years.

For TSPA-1993, the nominal case consisted of a heat-generating repository that is subjected to climate-dependent groundwater flow. Two alternative conceptual models of groundwater flow in the unsaturated zone were considered. Waste containers within the repository degrade by a variety of mechanisms, but the most important mechanism is aqueous-induced corrosion. If and when containers fail, radionuclides are available for gaseous or aqueous transport to the accessible environment. For gaseous transport, radionuclides move upward through the unsaturated zone to the ground surface. For aqueous transport, radionuclides move downward through the unsaturated zone, then laterally through the saturated zone past the 5-km subsurface boundary. Radionuclides are tracked in terms of (1) cumulative releases to the accessible environment and (2) the dose an individual might receive by drinking contaminated water pumped from the saturated zone at the accessible environment. For TSPA-1993, two disturbed scenarios were investigated: (1) inadvertent human intrusion by exploratory drilling, and (2) volcanic activity that introduces corrosion-enhancing heat and volatiles into the repository. For human intrusion, radionuclides exhumed with the drill core and the drilling fluids contribute to releases. For such indirect volcanic effects, magmatic-induced corrosion of containers allows earlier releases of radionuclides that are transported in groundwater flowing as described in the nominal case. (Direct volcanic releases had been evaluated in TSPA-1991.)

Monte Carlo simulation of nominal releases was done with the total-system analyzer, or TSA (Wilson, Lauffer, et al. 1991, Wilson 1992), which was a shell written in the UNIX C-shell language for running multiple realizations of stand-alone programs. It was very flexible, taking only minutes to develop a TSA shell to run a particular sequence of programs, unless a new translation program was needed for converting the output of a program to the form needed for input to another program. The LHS program (Iman and Shortencarier 1984) was used to generate the realizations from the input PDFs. Different process models, including alternative conceptual models, could be run within the TSA as needed, depending on the analysis.

The analyses investigated sensitivities using both cumulative releases and dose results for up to 1 million years. Percolation flux was the single most sensitive parameter. Again, as in TSPA-1991, all releases were below the EPA's 1985 standard, except for gaseous ^{14}C releases. However, longer-term (greater than 100,000 years) peak doses for drinking water pathways were shown to be significantly above background. These analyses were being performed prior to the remanded EPA 1985 standard, 40 CFR Part 191, and were not designed to evaluate regulatory compliance with a dose or risk-based standard.

Figure 42 shows calculated CCDFs of 10,000-year normalized cumulative release using the composite-porosity (top) and weeps (bottom) unsaturated zone flow models for all modeled release mechanisms, with the base-case design, a 57-kW/acre repository with vertically emplaced containers, are shown. With the composite-porosity model, all repository configurations produce similar results (though the analyses showed that repository design does influence releases predicted by the weeps model). Gaseous releases are predicted to be the most significant, showing the potential to contribute to violation of a 40 CFR Part 191 standard when modeled by the composite porosity model. Releases caused by human intrusion and nominal-case aqueous releases were important, but did not violate the standard. Indirect releases caused by volcanism are both few and low; direct releases caused by volcanism are low primarily because their very low probability of occurrence in 10,000 years.

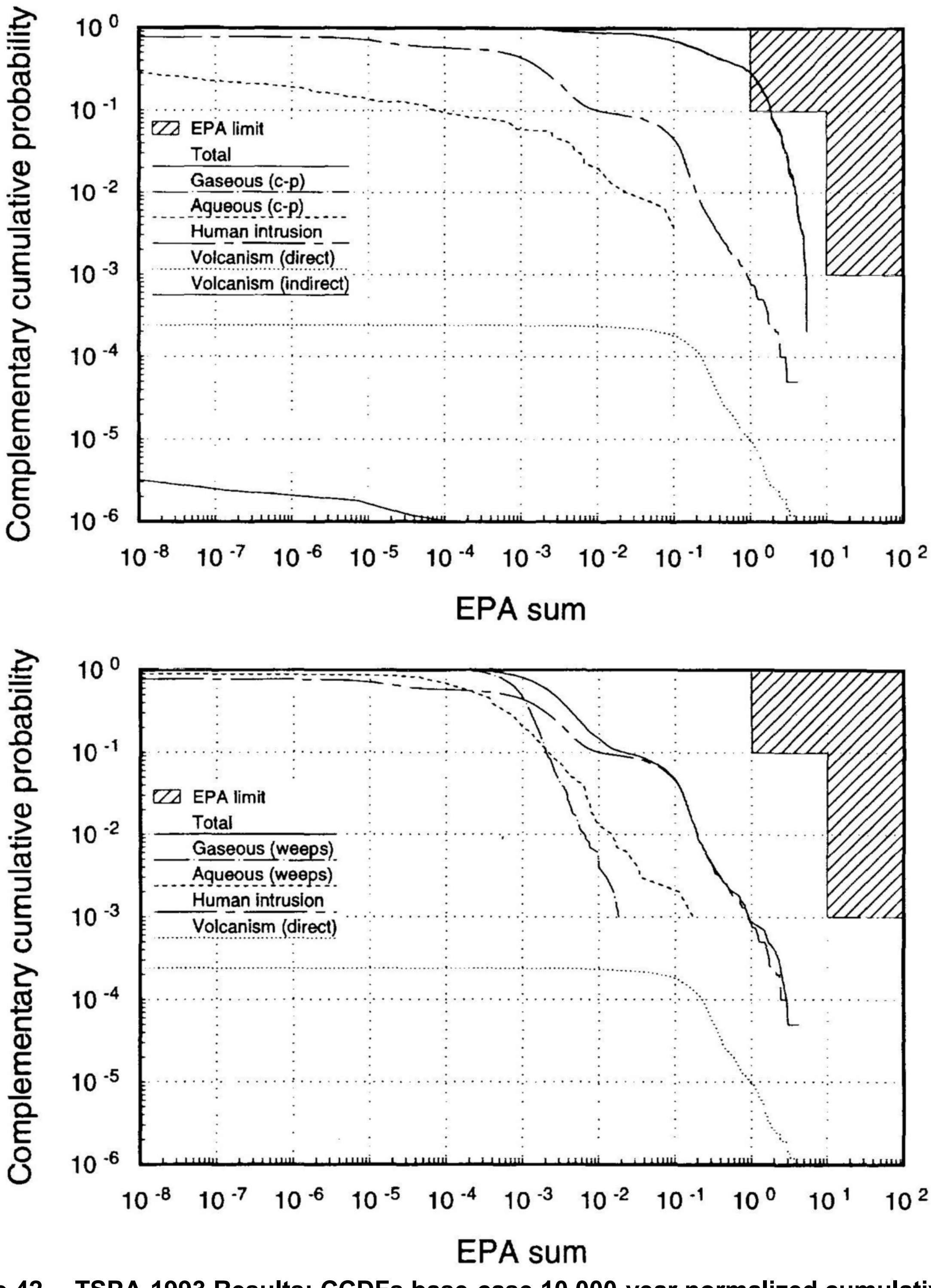


Figure 42. TSPA-1993 Results: CCDFs base-case 10,000-year normalized cumulative release predicted by the composite porosity (top) and weeps model (bottom), compared to the EPA 40 CFR Part 191 standard

Contemporaneous to SNL's TSPA-1993, other preliminary PAs for the Yucca Mountain site were being conducted. The Electric Power Research Institute (McGuire 1990, 1992), the NRC (Codell, et al. 1992), Golder Associates (Miller, Kossik and Cunnane 1992), Pacific Northwest Laboratory (Eslinger, et al. 1993), and DOE's Yucca Mountain management and operating contractor, the Civilian Radioactive Waste Management System Management & Operating Contractor (CRWMS M&O 1994), all produced preliminary TSPAs. The methods used by the different groups varied in degree of detail included in the models. EPRI, Golder, and the CRWMS M&O all used highly abstracted system models for their assessments, whereas Pacific Northwest Laboratory used detailed multidimensional models of flow and transport for its calculations. The NRC method represented an intermediate between these approaches, in terms of the level of detail and complexity maintained in the calculations versus the extent of abstractions used, as were SNL's application of the PA method in TSPA-1991 and TSPA-1993. The difference was reflected in the number of realizations used. Pacific Northwest Laboratory calculated flow and transport for a limited number of representative parameter values, whereas the others generally put more emphasis on probability distributions and exploring the sensitivity of the results to parameter variations (Wilson, Gauthier, et al. 1994).

The NRC used a method similar to that used for the Waste Isolation Pilot Plant site (WIPP Performance Assessment Department 1992–1993) in which the assessment is broken into separate calculations depending on the occurrence or nonoccurrence of disruptive events such as human intrusion. The method differs from that used for SNL's TSPA-1991 and TSPA-1993 in that each of the calculations is complete, including nominal releases as well as releases due to disruptive events. In TSPA-1991 and TSPA-1993, SNL calculated disruptive events to be independent from nominal conditions and from each other, as a means of simplifying the calculations. Pacific Northwest Laboratory used a method that combines some aspects of both methods. Golder and INTERA used a method that simulates the entire system at once, and, during each realization, there is some probability of a disruptive event taking place. "Importance sampling" is used to increase the number of realizations with low-probability disruptive events. The method used by EPRI was different from all the others in being based on logic-tree formalism. In the EPRI method, probability distributions were not defined for uncertain parameters, but rather a logic tree was defined, with branches representing a few discrete values of some of the uncertain parameters. Disruptive events were also represented with branches in the logic tree (Wilson, Gauthier, et al. 1994).

5.2.4 TSPA-1995

The work performed independently among participants of the Yucca Mountain project had been merged into a single TSPA effort lead by the Yucca Mountain management and operating contractor by the time the TSPA-1995 (CRWMS M&O 1995) was initiated (as described previously, SNL continued to participate directly in the conduct of YMP TSPAs between 1995 and 2005 but did lead the PA effort until 2006, during the development of the TSPA for the license application). Four specific goals were identified for the 1995 iteration of the Yucca Mountain TSPA:

1. Utilize what were believed to be more representative conceptual models that built upon the assumptions employed in TSPA-1993, in particular, for the treatment of the engineered barrier system, including the waste package, using reasonably

conservative representations of the relevant processes and parameters affecting total system performance.

2. Incorporate more recent design information than was available for TSPA-1993, evaluating a range of alternative conceptual models and parameters to explicitly address the uncertainty and variability in the understanding and the significance of that uncertainty on the predicted performance.
3. Utilize the most recent site information and models, acknowledging their uncertainty and variability, focusing the analyses on those components of the waste containment and isolation system that are most sensitive.
4. Evaluate the engineered barrier system release performance measure, as well as alternative measures of total system performance, using a range of possible measures of safety, including cumulative radionuclide releases, peak concentrations, or doses.

The focus of TSPA-1995 was on those components of the system that were determined in TSPA-1993 to be most significant in containing and isolating radioactive wastes from the biosphere. These were the engineered components of the system and the near-field environment in which the engineered components reside (CRWMS M&O 1995, p. 1-3).

The repository conceptual design had changed, with the original emplacement concept, with thin-walled containers vertically emplaced in the drifts being replaced by the horizontal waste package emplacement approach, which had been considered as an option in TSPA-1993. Specific changes to TSPA-1995 models included:

- Inclusion of a drift-scale thermal-hydrologic environment to derive relative humidity and temperature information adjacent to the waste package

- A more detailed waste-package-degradation model, including corrosion of both inner and outer waste-package layers and galvanic protection of the inner layer

- Calculation of releases both with and without backfill

- Modification of solubility and retardation values for radionuclide transport

- Consideration of two alternative ranges of percolation flux (high flux: 0.5 to 2.0 mm/yr; low flux: 0.01 to 0.05 mm/yr)

- Disruptive events and gaseous releases were not included in TSPA-1995. A simple climate change model was also incorporated into the analyses. The time period for the calculation of release and dose was up to 1 million years.

TSPA-1995 also reported percolation flux as the most important factor to performance of the proposed repository in a 10,000-year performance period. Many components of the engineered barrier system were also shown as key to performance, as indicated by sensitivity analyses of galvanic protection, varying thermal load and backfill configurations, and radionuclide transport through the engineered barrier system. Consistent with TSPA-1993, sensitivities were calculated in terms of system-level performance measures (e.g., dose and cumulative release). Though basic comparisons were made against 40 CFR Part 191 release limits, no attempt was made at evaluating regulatory compliance with a dose- or risk-based standard.

Because the EPA had not yet proposed an environmental standard for Yucca Mountain, the performance measure remained an issue for the overall program as well as the PA. In addition, the National Research Council had recently recommended a compliance assessment period on the order of 1 million years for Yucca Mountain. Therefore, five different measures of performance were evaluated in the 1995 TSPA (CRWMS M&O 1995). The first two considered subsystems: the waste package (substantially complete containment) and the engineered barrier system (the peak radionuclide release rate). The remaining three measures quantified total system performance: the cumulative radionuclide release reaching the accessible environment over 10,000 years; and the maximum radiation doses in both 10,000 and 1 million years to an individual located at the accessible environment boundary.

A number of combinations of repository design and natural system behavior examined in sensitivity analyses resulted in no releases at the accessible environment up to 10,000 years after repository closure. These sensitivity analyses resulting in no releases included: (1) low infiltration range (0.01 to 0.05 mm/yr), (2) cathodic protection of the waste package, (3) alternative thermal-hydrologic models with 80 MTU/acre thermal load with and without backfill and 24 MTU/acre thermal load with backfill, and (4) a matrix-flow-only (zero fracture flow) unsaturated zone model. Thus for these cases, there are no CCDFs or expected-value breakthrough curves to be shown.

Two cases that did result in releases to the accessible environment were the two thermal loads modeled with another alternative thermal-hydrologic model, 83 MTU/acre and 25 MTU/acre, with a gravel backfill, at the high infiltration range (0.5 to 2.0 mm/yr). Normalized total cumulative releases for these two thermal loads are shown in Figure 43, where the shaded area in the figure represents the release limits in EPA's 40 CFR Part 191. Although not specifically indicated in this figure, the radionuclides with greatest releases to the accessible environment during the 10,000-year time frame were nonsorbing radionuclides: ^{99}Tc, ^{14}C, ^{129}I, and ^{36}Cl.

TSPA-1995 predictions of dose and peak dose over a 1-million-year time frame were presented as various sensitivity analyses examining the effect of various natural system parameters and alternative models and various repository designs. These results were presented in a series of expected-value dose histories as well as CCDFs. An example dose history result from TSPA-1995, presenting a sensitivity analysis comparing different infiltration rates ("q_{inf}") is shown in Figure 44.

TSPA-1995 served to identify the highest priority in preparation for the next full iteration of TSPA to be conducting a detailed technical analysis of the robustness of the process models under development. Equally significant was ensuring that the developed and substantiated process models could be appropriately abstracted for use in the next TSPA (CRWMS M&O 1995, p. 10-26).

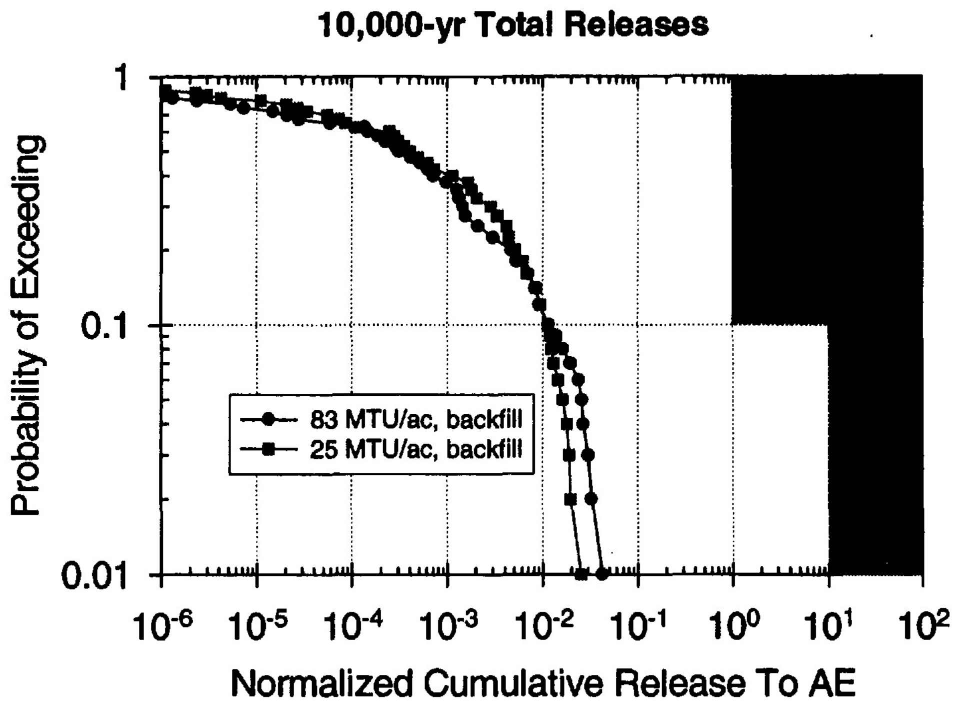


Figure 43. TSPA-1995 Results: CCDF of total normalized cumulative release: 10,000 years, 83 MTU/acre and 25 MTU/acre, backfill, high infiltration range, with EPA release limits from 40 CFR Part 191 in shaded area

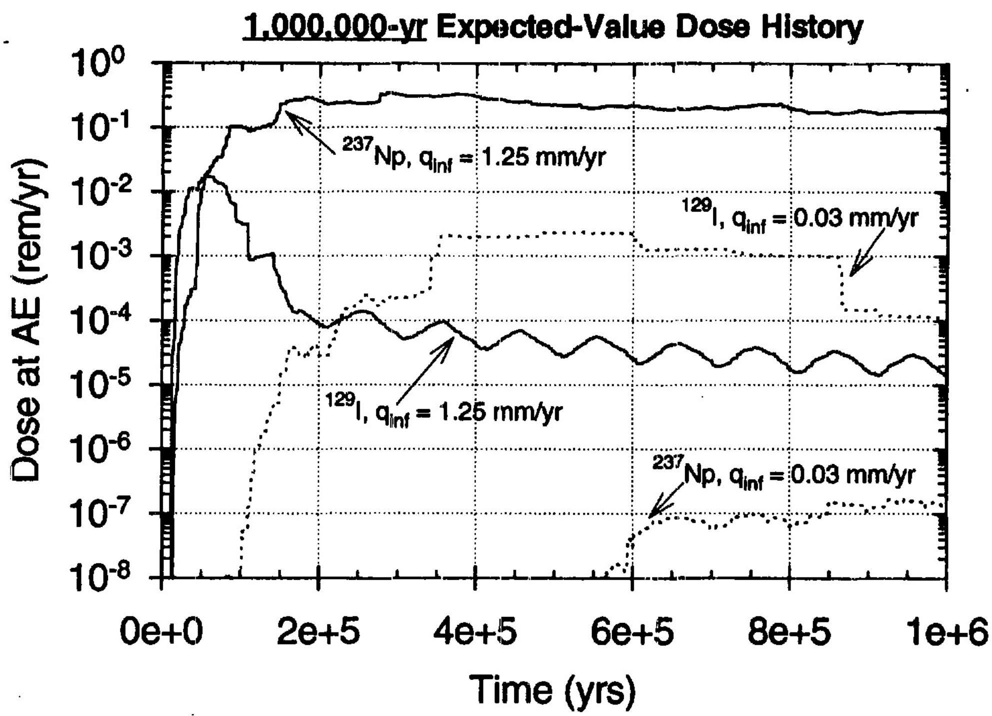


Figure 44. TSPA-1995 sensitivity analysis: 1-million-year expected-value dose history for ^{129}I and ^{237}Np: Infiltration Rate Comparison: "high" (1.25 mm/yr) versus "low" (0.03 mm/yr) infiltration

During this period, other organizations also performed TSPAs on Yucca Mountain, independently of the DOE's Yucca Mountain Project organization. They included TSPAs run by the NRC (Wescott, et al. 1995) and by the Electric Power Research Institute (Kessler and McGuire 1996). In addition, as discussed separately in Section 7, Sandia National Laboratories conducted a TSPA for the DOE Office of Environmental Management (Idaho), focused on evaluating the performance of the DOE-owned waste forms at INL on a hypothetical "Yucca Mountain-like" tuff site (Rechard 1995), which helped to guide the next iteration of design and PA. The suite of scenarios analyzed by each of these groups was essentially the same as those used in the YMP TSPAs. In each case, assumptions about conceptual models and parameter distributions differed somewhat from those used in the YMP TSPAs. However, the conclusions reached about the importance of particular processes and parameters (with the possible exception of basaltic volcanism in the case of the NRC analysis) with respect to performance of the repository system were very similar to those determined by the various YMP TSPA analyses.

5.2.5 TSPA for Viability Assessment (TSPA-VA) (1998)

The overall scope and objective of the TSPA for the Viability Assessment, known as the TSPA-VA, was outlined in the Energy and Water Development Appropriations Act for 1997, which required the DOE to:

> …provide to the President and to the Congress a viability assessment of the Yucca Mountain site. The viability assessment shall include [among other things] ... a total system performance assessment, based upon the design concept and the scientific data and analysis available by September 30, 1998, describing the probable behavior of the repository in the Yucca Mountain geological setting relative to the overall system performance standards…

A series of abstraction and testing activities were initiated to identify and construct appropriate numerical or analytical representations of components of the potential repository system to ensure the development of a valid, defensible TSPA. However, due to time and resource constraints, the model development was focused on only those issues expected to have the most influence on long-term performance. Site-specific EPA and NRC dose or risk-based standards were in development but not complete at the time, and the TSPA-VA analyses assumed a 10,000-year period, but also evaluated the consequences caused by the repository beyond that period, the analyses were extended to 100,000 and 1 million years in determining when the peak radionuclide doses or peak risk occurs. In addition, the TSPA-VA assumed the hypothetical receptor for exposure calculations to be located 20 km (12 mi) downgradient of the repository (DOE 1998, pp. 2-2 and 2-3). The TSPA-VA was the first effort by the DOE to incorporate an all-pathway biosphere model with three receptor scenarios for radiation dose assessment.

DOE also used the TSPA-VA to quantitatively define the significance of each of the key components in the repository safety strategy to assist in a systematic refocusing of the project resources. Though statutory goal of the TSPA was to address the probable behavior of the repository system, the available scientific information can also suggest alternative interpretations that may also be plausible. When propagated through a quantitative tool such as PA, these alternative interpretations can illustrate the significance of the uncertainty in the base case interpretation chosen to represent the probable behavior of the repository. The information about uncertainty helped DOE in defining work required either to minimize uncertainty or to modify the repository design to account for this uncertainty before making a Secretarial site recommendation and then submitting the license application for constructing the repository system. The quantitative performance analyses assist in identifying those areas where additional scientific and technical work are required to evaluate the site and to prepare a complete, cost effective, and timely license application.

The TSPA-VA also helped provide a vehicle for prelicensing discussions with NRC, identifying and resolving the key technical issues most important to repository performance. Especially in the context of a Congressionally mandated repository viability assessment, TSPA served an important role in evaluating the potential regulatory significance of these issues to establish a common basis for understanding the need for additional scientific and technical work (DOE 1998, p. 2-3).

The executive driver program that linked the component codes of the TSPA-VA was RIP V5.19.01, which was developed by Golder Associates for the DOE. RIP was a DOS-based predecessor to GoldSim. The RIP program was designed to conduct either single realization runs of the entire system or multirealization runs of the system, but was generally designed to handle simplified component models in order to conduct multiple realizations of the total system model (DOE 1998, p. 2-30). Therefore, for the TSPA-VA, the PA results were presented as a single-realization deterministic analysis, showing the outcome of sampling all uncertain input parameters in the TSPA-VA component models at the expected value of their ranges (i.e., at their mean or average value). The purpose was to illustrate how total system behavior (i.e., individual dose rate) is influenced by the various component or subsystem models and parameters. The overall system results for 10,000, 100,000, and 1 million years are shown in Figure 45. In addition, a probabilistic approach using a linked system of deterministic models to represent the repository and its associated geologic system, and a Monte Carlo technique to propagate parameter uncertainty through to the calculation of peak radiation dose rates at the specified location 20 km (12 miles) from the repository (DOE 1998, Section 4.3). These probabilistic analyses were used for uncertainty and sensitivity analyses. Those calculations used 100 realizations, presented in horsetail plots.

The results of the TSPA-VA were used, along with other considerations such as cost and schedule, to outline a detailed plan for preparation of a license application, including further site investigations and testing as well as further model development.

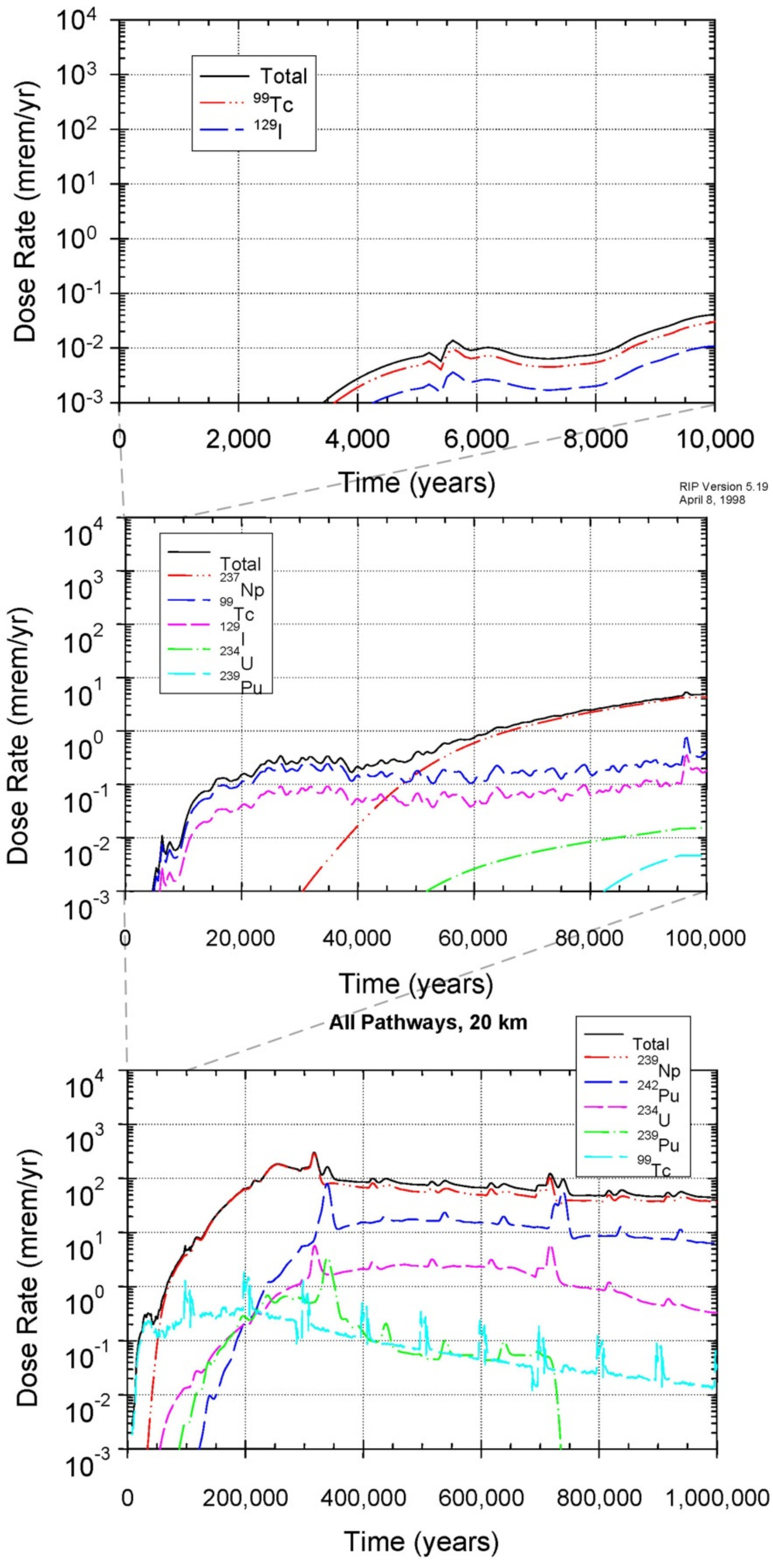


Figure 45. TSPA-VA result: "expected value" sampling presenting dose rate to an average individual withdrawing water from a well penetrating the maximum plume concentration in the saturated zone 20 km (12 miles) downgradient from the proposed Yucca Mountain repository

5.2.6 TSPA for Site Recommendation (TSPA-SR) and Supplemental PAs (2000-2001)

The scope of the TSPA-SR (CRWMS M&O 2000) was guided by technical requirements proposed by the NRC for 10 CFR Part 63 (64 FR 8640), and radiation protection standards proposed by the EPA for 40 CFR Part 197 (64 FR 46976). At the time, those proposed regulations specified a 10,000-year performance period, so the TSPA results quantify the performance of the potential repository for 10,000 years. However, to complement the results of the 10,000-year PA, the peak dose to the reasonably maximally exposed individual beyond the 10,000-year time period was calculated. TSPA results for 100,000 and 1 million years were calculated for the nominal case to gain insight into possible peak dose levels at times significantly beyond 10,000 years.

For the TSPA-SR, a rigorous and comprehensive FEP analysis was performed and a FEP database developed by Freeze, Brodsky, and Swift (2001). This was the first YMP FEP analysis to rigorously and comprehensively apply the scenario development process developed by Cranwell et al. (1990). The YMP FEP database was initially developed from a comprehensive list of FEPs from other international radioactive waste disposal programs and was supplemented with additional YMP-specific FEPs from project literature, technical workshops, and reviews. The sources identified above produced 1,646 specific FEPs. These FEPs, when combined with the 151 general FEP classifications, resulted in a FEP database that contained 1,797 entries.

To organize these FEPs and to help evaluate the completeness of the FEP list, a hierarchical classification structure was adopted within the YMP FEP database. The review of the YMP FEP database resulted in 111 classification entries (40 heading entries were reclassified as primary FEPs), 323 primary FEP entries representing a single process or event or a few closely related or coupled processes or events that can be addressed by a specific screening discussion, and 1,363 secondary FEPs that were determined to be redundant or better represented by a Yucca Mountain–specific primary FEP. Since the secondary FEPs were subsumed in the primary FEPs, only primary FEPs were screened.

Screening was done as specified by regulations, which provided that FEPs could be excluded from the TSPA (screened out) on the basis of low probability (having less than one chance in 10,000 of occurring over 10,000 years may be excluded) or consequence (i.e., if their exclusion would not significantly change the expected annual dose).

The TSPA-SR was the first TSPA to use GoldSim as the probabilistic shell for the TSPA component models. GoldSim was developed by Golder Associates as a successor to the RIP code developed for DOE for geologic repository PA and applied first for TSPA-1995. It was also used in the supplementary PA analyses for site recommendation and continued to be used through the TSPA-LA.

The supplemental science and performance analyses performed in 2001, in preparation for the 2002 site recommendation, were intended to quantify uncertainties, update process models based on recent scientific information, and analyze the effects of a lower-temperature repository operating mode on process model results. A supplemental TSPA model was prepared, incorporating information from those analyses and updated process models and reflecting the alternate thermal operating modes. In 2001, when the EPA's environmental standards for Yucca Mountain (40 CFR Part 197) and the NRC's Yucca Mountain licensing requirements (10 CFR Part 63) were finalized in June and November, respectively, that supplemental TSPA model was revised in two separate analyses. The supplemental TSPA (Bechtel SAIC Company 2001a, 2001b) and the revised supplemental TSPA models for the Final EIS (Bechtel SAIC Company 2001c) and for the final NRC regulations (Bechtel SAIC Company 2001d) all used the same TSPA approach and method (DOE 2002a). All of these models and analyses explicitly considered both disruptive events and alternative models that could result in unanticipated behavior. The TSPA-SR model and supplemental and revised supplemental TSPA models consisted of approximately 1,000 parameters, many of them uncertain or variable. The evaluations directly addressed uncertainty in both the understanding of the site and in future conditions, and they included numerical sensitivity analyses to test how the repository might perform if current or future conditions differ from those expected.

Results from the multiple realizations were graphically summarized in "horsetail plots" showing time versus annual dose (i.e., annual dose histories) for all realizations, 300 realizations for the nominal case and 5,000 realizations for the igneous-activity scenario. These results of the multiple-realization simulations were displayed along with statistical measures of the output. The mean (representing the arithmetic average of data points from each realization at each time step) was the performance measure established by the regulations. The median of the output along with 5th and 95th percentile of the output was also frequently plotted in graphical representations of the results.

Figure 46 presents the results in horsetail plots for the nominal case of the TSPA-SR model (top) and for the nominal case of the supplemental TSPA model (bottom) for the higher-temperature operating mode (DOE 2002a, Figure 4-179). The figure presents the results for each of the 300 realizations, as well as the mean, median, and 5th and 95th percentiles of the distribution of these simulations to examine the effects of uncertainty on the projected dose.

The supplemental and revised supplemental TSPA models forecast doses occurring before 10,000 years, whereas the TSPA-SR model forecast no dose in the first 10,000 years. The primary reason for this was the incorporation of nonmechanistic early waste package failures into the supplemental TSPA. Based on the low probability and the use of administrative controls to further reduce the probability of mechanisms that could lead to early failure, no mechanisms that could lead to early failure of waste packages were included in the TSPA-SR model. In reevaluating the potential of early failure mechanisms and their potential consequences, a more conservative approach resulted in the inclusion of improper heat treatment and subsequent possible failure of up to three waste packages in the supplemental TSPA analyses. The early waste package failure assumes failure of both the inner and outer Alloy 22 lids and the stainless steel inner lid. To ensure that the potential consequence of early waste package failures is treated conservatively, it was included in the nominal scenario, not as a sensitivity analysis, for the supplemental and revised supplemental TSPA model analyses. In the revised supplemental model, assuming nonmechanistic early failure of all waste packages would result in an annual dose during the first 10,000 years of less than 1 mrem/yr, or less than one-third of one percent of the average dose from natural background radiation (DOE 2002a, pp. 4-462 and 4-463).

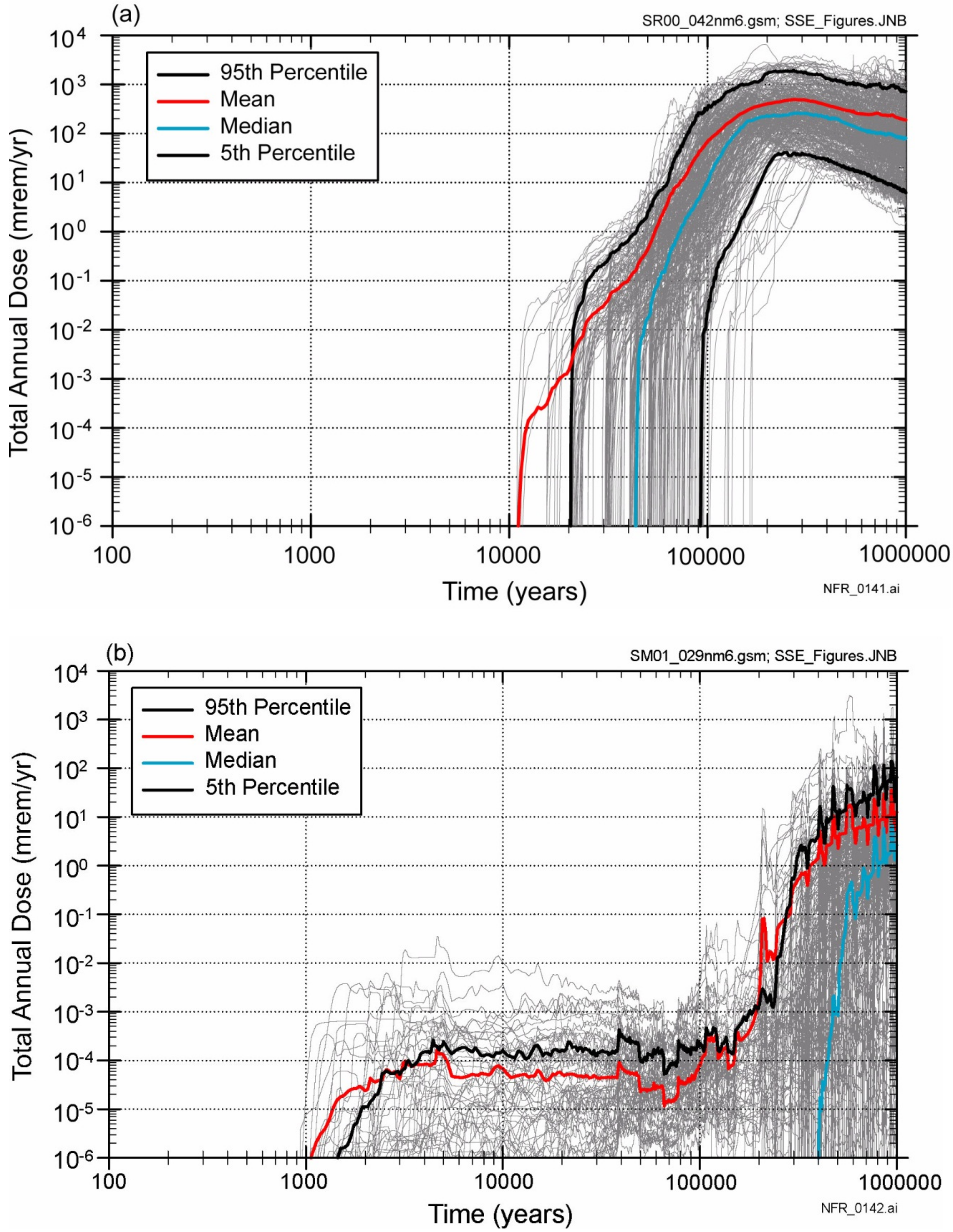


Figure 46. Nominal scenario results for the TSPA-SR model and supplemental TSPA model for the higher-temperature operating mode

Figure 47 (DOE 2002a, Figure 4-180) shows the results of the TSPA and supplemental models, presenting the mean annual dose for each for comparison. The "Rev. Suppl. Model, HTOM" results are for the model for the final EIS; the analyses for the final NRC regulations showed that these calculated doses would be reduced by approximately one-third using an annual water demand of 3,000 acre-ft, consistent with final NRC regulations.

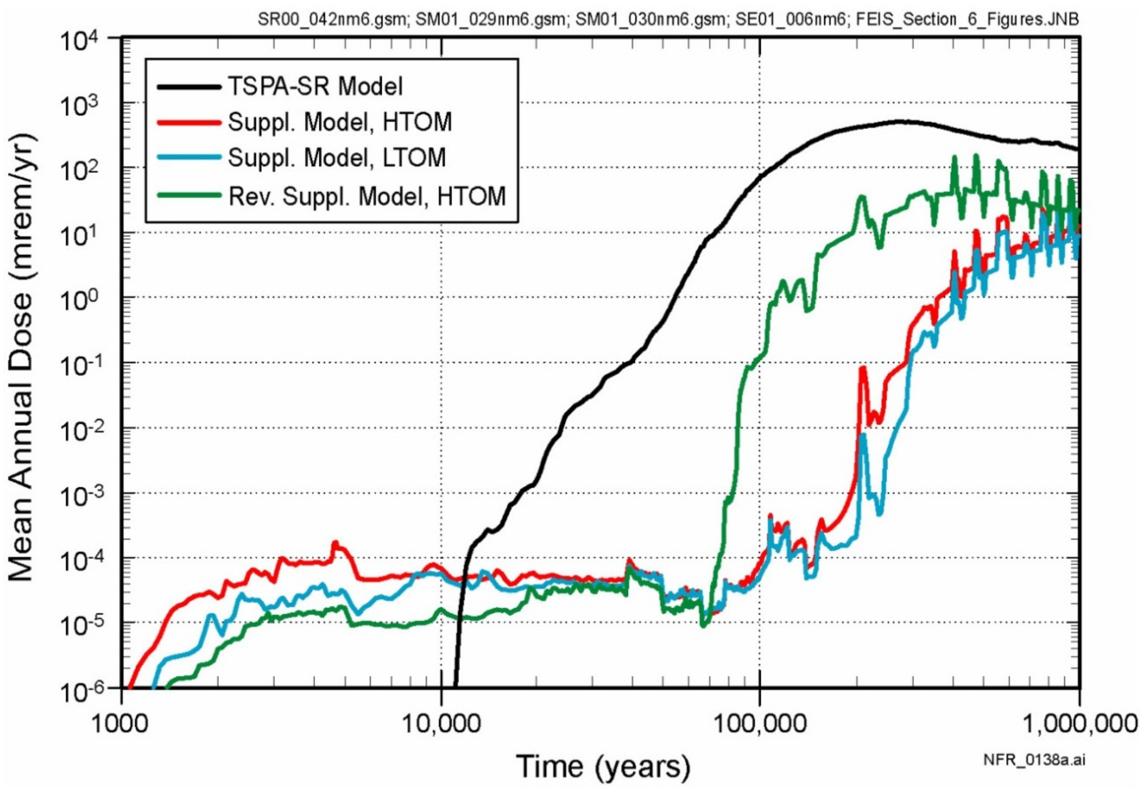


Figure 47. Mean annual dose results, nominal scenario for TSPA-SR, supplemental model (both higher temperature and lower temperature operating modes), and revised supplemental model for the EIS (higher temperature operating mode)

5.2.7 2008 TSPA for License Application (TSPA-LA)

The TSPA for the license application, known as the TSPA-LA (SNL 2008c), is built on the foundation of the earlier Yucca Mountain PAs, enhanced by updated analyses of the processes affecting Yucca Mountain and the design elements of the repository, including a comprehensive consideration of the FEPs that are relevant to repository system performance. The conceptual model examined in the TSPA-LA was fundamentally the same as previous TSPAs.

As before, the performance measures established by the regulations included dose limits of the individual protection standards, radionuclide concentration and dose limits of the groundwater protection standards, and simulations for a stylized human intrusion scenario, as well. However, the performance measures called for by the individual protection standard were changed significantly between TSPA-SR and TSPA-LA. Though prior TSPAs did include calculations out to 1 million years, the regulations then only required a 10,000-year compliance period. In the intervening years, the EPA standard was remanded and then re-promulgated to include a peak dose calculation within a 1-million-year compliance period for the individual protection standard. But, in general, as a system-level model that integrates numerous submodels representing each of the components of the natural and engineered barriers at Yucca Mountain, the TSPA-LA model remained very much the same, relying on abstractions of some of the major processes due to the complexity of those processes and the large number of system-level simulations required for the Monte Carlo uncertainty analysis.

For the 2008 TSPA-LA (SNL 2008c), a total of 374 FEPs were identified as relevant and 222 were excluded, leaving 152 FEPs in the analysis. The included FEPs were assembled into four discrete scenario classes that were analyzed probabilistically:

1. An early failure scenario class, in which one or more waste packages or overlying drip shields fails prematurely due to undetected manufacturing or material defects or to preemplacement operations including improper heat treatment,
2. An igneous disruption scenario class in which a volcanic event causes magma to intersect the emplacement region, with or without an accompanying eruption,
3. A seismic disruption scenario class, which comprises a seismic ground motion modeling case and a seismic fault displacement modeling case
4. A nominal scenario class in which none of these three types of events occurs.

Each event-based scenario class was subdivided into separate modeling cases to simulate the consequences of specific events. The total mean annual dose for 10,000 years was developed by summing the mean annual doses for each modeling case. The TSPA-LA results were well below the NRC's 10 CFR Part 63 and EPA's 40 CFR Part 197 rules for 10,000 years and after 10,000 years and within the period of geologic stability (1 million years). As shown in the plot at the top of Figure 48 (SNL 2008c, Figure 8.1-1[a]), for the period ending 10,000 years after disposal, the result obtained by adding together the mean annual dose curves for the four scenario classes indicates that the mean annual dose for the total repository system is approximately 0.24 mrem/yr. Even considering the conservative nature of the TSPA model and analyses, this mean annual dose was significantly less than the individual protection standard. This dose result was most significantly affected by the seismic ground motion modeling case and the igneous intrusion modeling case. Uncertainty analyses showed that the parameters that most affected the total uncertainty in the TSPA-LA model were factors that govern degradation of the waste packages, the occurrence of damage from seismic events, and the frequency with which igneous intrusions occur.

For the period after 10,000 years and within the period of geologic stability as prescribed by 10 CFR Part 63, the TSPA-LA projected a peak median annual dose of approximately 0.96 mrem/yr, as shown in Figure 48 (bottom) (SNL 2008c, Figure 8.1-2[a]). This dose result was also most significantly affected by the seismic ground motion modeling case and the igneous intrusion modeling case. The modeled dose was a fraction of naturally occurring background radiation and well below the standard of 350 mrem/yr. The TSPA-LA results indicated that the largest contributors to the estimated maximum mean annual dose came from the igneous intrusion and seismic ground motion scenario classes, considering the probability of occurrence of these events. The primary release mechanism late in the million-year period was nominal corrosion processes that lead to degradation and failure of the waste packages.

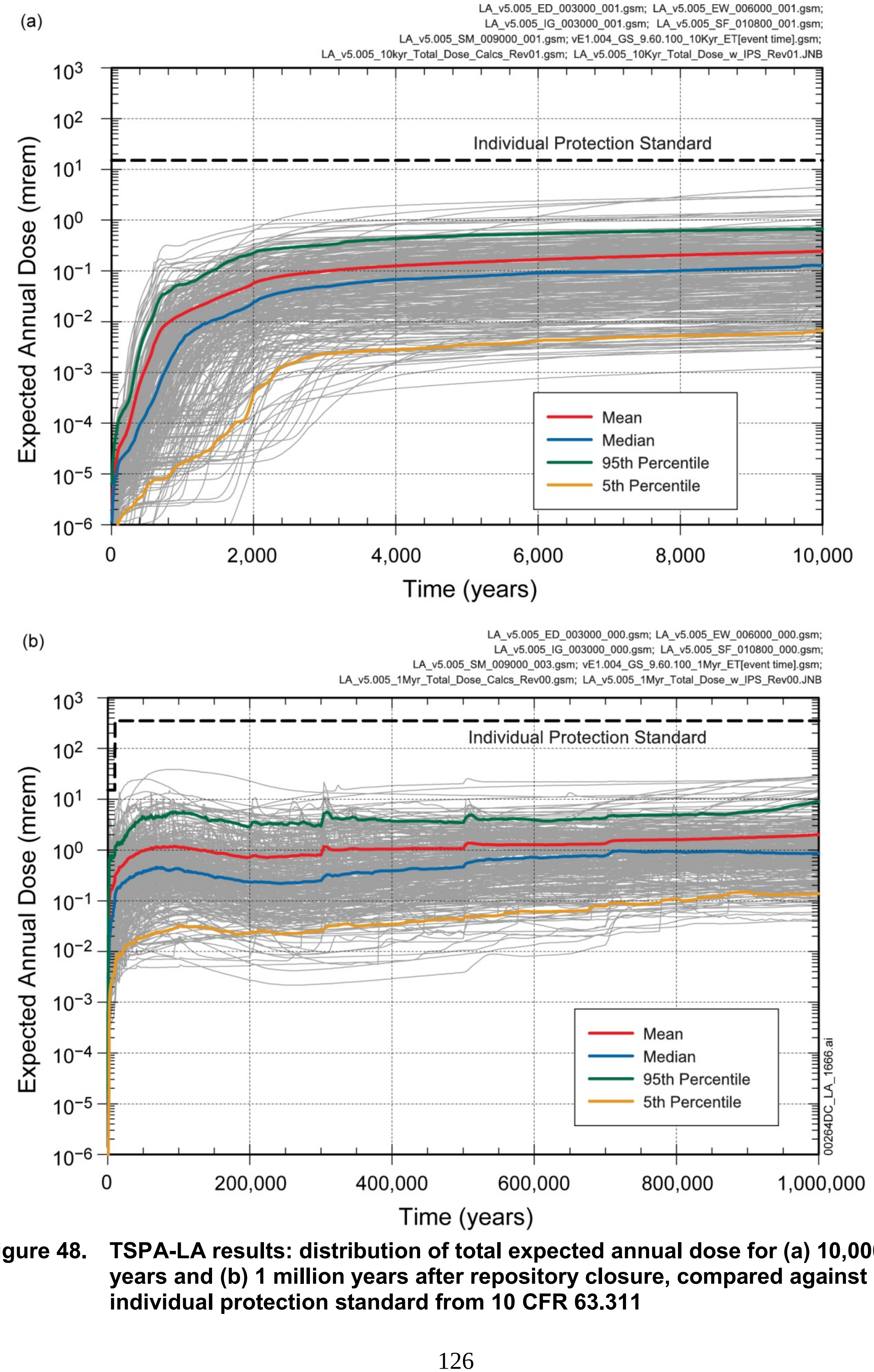


Figure 48. TSPA-LA results: distribution of total expected annual dose for (a) 10,000 years and (b) 1 million years after repository closure, compared against the individual protection standard from 10 CFR 63.311

In addition to the dose results for comparison to the individual protection requirements, demonstration of the performance of the repository also included compliance with the separate groundwater protection standards for specific radionuclide concentrations. The results of the calculations for groundwater protection standards are shown in Figure 49 (SNL 2008c, Figures 8.1-16[a], 8.1-9[a], and 8.1-11[a]). Radionuclide concentrations were calculated by summing the mass of radionuclides reaching the accessible environment in each year for all likely FEPs and dividing that sum by the representative volume of water to calculate the annual radionuclide concentrations. The groundwater protection standards require calculations of the predicted concentrations of combined ^{226}Ra and ^{228}Ra and gross alpha activity in a representative volume of 3,000 acre-ft of groundwater. The standards also require calculation of the annual dose to the whole body and organs from beta- and photon-emitting radionuclides resulting from drinking 2 L of water per day. The background level of the combined ^{226}Ra and ^{228}Ra concentration in groundwater is about 0.5 pCi/L. This measured background concentration must be added to the calculated concentration of combined ^{226}Ra and ^{228}Ra released from the repository for comparison with the postclosure groundwater protection standard for combined ^{226}Ra and ^{228}Ra. Because the calculated concentration of combined ^{226}Ra and ^{228}Ra released from the repository is less than 10^{-6} pCi/L, the total combined ^{226}Ra and ^{228}Ra concentration was reasonably approximated by the measured background level of about 0.5 pCi/L, well below the limit of 5 pCi/L.

In addition to the four scenario classes above, a separate human intrusion scenario was stylized based on the specifications in 10 CFR Part 63, rather than developed from FEPs analysis of probability and consequence. The human intrusion scenario assumes a single human intrusion as a result of exploratory drilling for groundwater. Based on analyses of drip shield and waste package integrity, the analysis indicated that the earliest that the drip shields and waste packages would degrade enough that a human intrusion could occur without recognition by drillers is approximately 200,000 years after disposal, resulting in dose results shown in Figure 50 (SNL 2008c, Figure 8.1-16[a]). The estimated annual dose resulting from the stylized human intrusion is approximately 0.01 mrem/yr, well below the regulatory limit of 350 mrem/yr.

5.3 Significance of Yucca Mountain Repository PA in the Historical Development of the PA Methodology

The Yucca Mountain repository system is the most complex yet modeled in a regulatory PA. Its natural system, including the geologic and hydrologic setting, and its engineered systems in their final form were more complex than considered in any prior PAs, as was the representation of coupled thermal effects on repository processes at Yucca Mountain. In addition, the PA also addressed seismic effects and the probability and consequences of volcanic intrusion and eruption.

Moreover, the regulatory performance measures applicable for Yucca Mountain (i.e., dose rates to a reasonably maximally exposed individual) result in a more complex PA than the regulatory performance measures applicable to WIPP (i.e., total radionuclide release). Dose limits tend to demand more detail in the total system simulations in order to reveal dose peaks over time, requiring the inclusion of more parameters in the model; probabilistic assessment of cumulative radionuclide release, in comparison, tends to have fewer important parameters.

In those aspects among others, the Yucca Mountain repository PAs help to demonstrate the versatility of the PA methodology.

Comparison with other PAs, notably the SDP, WIPP, and GCD programs, shows the very important finding that, all else being equal, the simpler the geologic formation the simpler, quicker, less costly, and more transparent the PA will be. While this finding has no bearing whatever on the relative safety of geologic repositories, it has important ramifications on the sociopolitical aspects of siting and licensing repositories.

Because of computational limitations of GoldSim, the YMP PAs tended to implement abstraction models based on lookup tables rather than simplified or streamlined physics models such as those implemented in, for example, the PAs for INL HLW, as described in Section 7. Though abstractions to lookup tables may tend to diminish transparency of the PA calculations and require additional technical justification in comparison to use of physics models, prelicensing interactions with the NRC as well as technical discussions with the U.S. Nuclear Waste Technical Review Board suggest that this abstraction approach can be successful.

Though the NRC's safety evaluation report for the TSPA-LA has not been published, the Yucca Mountain repository license application was considered by the NRC acceptable for docketing, which indicates that its PA documentation was complete enough to serve as a basis for their technical review.[8] The President and his Administration have decided that the Yucca Mountain repository is not a workable option for disposal of SNF and HLW, and based on this determination the DOE has moved withdraw the Yucca Mountain repository license application from NRC review. Nevertheless, the technical foundation of that license application—including the engineering design, and preclosure safety analyses as well as the postclosure PA described in this report—represents a significant technical accomplishment in management of SNF and HLW, and its PA is among the most important that have been conducted to date.

[8] In August 2011, the NRC published a Technical Evaluation Report (NRC 2011) for the postclosure performance evaluation contained in the YMP license application. While the Technical Evaluation did not include regulatory findings or conclusions regarding whether the PA satisfied the NRC regulation, its summary conclusions indicated that "the Total System Performance Assessments (TSPAs) used for the individual protection, human intrusion, and separate groundwater protection calculations are reasonable; and … the technical approach and results in DOE's TSPA, including the average annual dose values and the performance of the repository barriers, discussed in this TER, are reasonable."

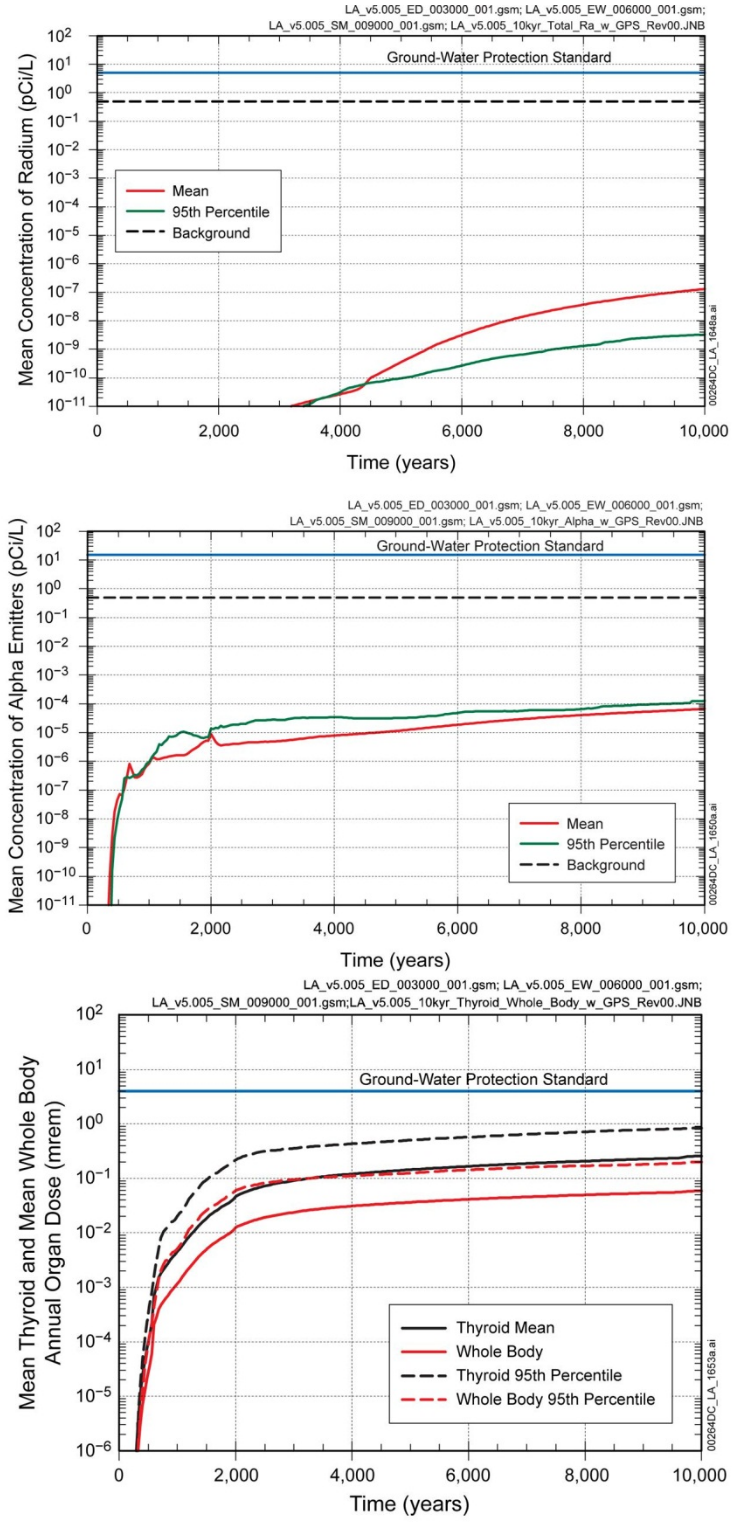


Figure 49. TSPA-LA results for groundwater protection for 10,000 years after repository closure: activity concentrations for total radium (^{226}Ra and ^{228}Ra), excluding natural background (top); activity concentration of gross alpha (including ^{226}Ra but excluding radon and uranium) (middle); and annual drinking water doses for combined beta and photon-emitting radionuclides (bottom)

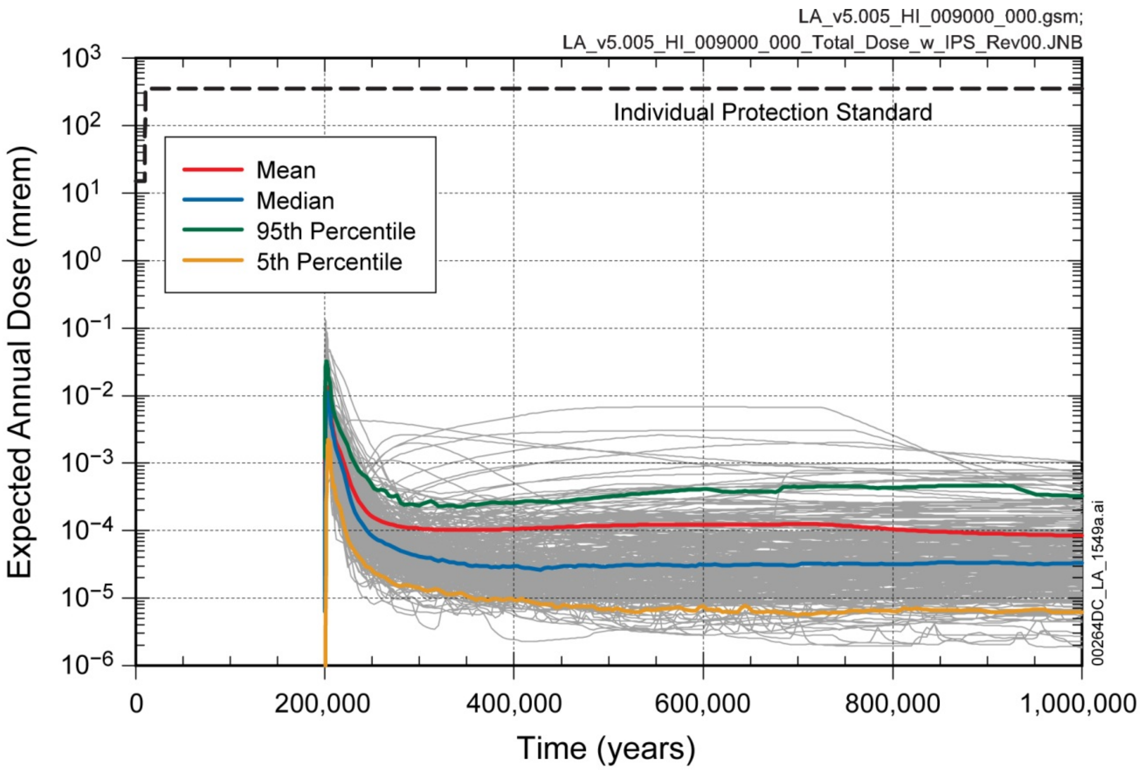


Figure 50. TSPA-LA results: distribution of expected annual dose for the human intrusion modeling case with drilling intrusion event at 200,000 years after repository closure

6. DOE GREATER CONFINEMENT DISPOSAL PAs (1989–2001)

6.1 Background

Greater confinement disposal (GCD) refers to the use of intermediate-depth boreholes for the disposal of radioactive waste. In 1981, DOE Nevada Office (DOE/NV) began investigating the use of GCD boreholes at Area 5 at the Nevada National Security Site (formerly, Nevada Test Site) for the disposal of LLW and certain high specific activity LLW. In 1984, approximately 1.11 million curies of high-specific activity LLW were placed in a GCD test borehole. Intermediate depth disposal operations were from 1984 through 1989; twelve additional GCD boreholes were constructed. Four of those boreholes are empty and the remaining eight boreholes contain LLW, hazardous (i.e., RCRA, Resource Conservation and Recovery Act) wastes, and TRU wastes. A timeline of the program, shown against the backdrop of other contemporary developments in PA is shown in Figure 51.

The GCD boreholes were developed in the alluvium of Frenchman Flat at the Radioactive Waste Management Site at the Nevada National Security Site (Figure 52) and consist of 120-foot (36.6-m) deep, 10-foot (3-m) diameter boreholes (Figure 53). Waste is placed in the bottom 50 feet (15.2 m) and the upper 70 feet (21.3 m) is backfilled with alluvium. There are no caps, sleeves, liners, or engineered barriers, and the bottom of each borehole is approximately 650 ft (198 m) above the water table.

In 1989, DOE asked SNL to evaluate whether the TRU waste disposed in GCD boreholes at the Nevada National Security Site would comply with the containment requirements of 40 CFR Part 191 Subpart B. SNL evaluated the long-term performance of the GCD boreholes through three iterations of the PA, which were completed in 1993 (Price, et al. 1993), 1994 (Baer, et al. 1994), and 2001 (Cochran, Beyeler, et al. 2001). The final GCD PA provided reasonable expectation that the cumulative release in 10,000 years would be within the maximum release specified in 40 CFR Part 191. These PAs were significant for their effective demonstration of the probabilistic, iterative PA methodology for the GCD concept. Additionally, the GCD boreholes at the Nevada National Security Site were the first successful completion and acceptance of a PA for TRU waste under DOE self- regulation and became only the second site, after WIPP, to meet the safety requirements of 40 CFR Part 191 for disposal of TRU waste.

In 1995, SNL was asked to evaluate the GCD boreholes at Nevada National Security Site for the disposal of vitrified Fernald byproduct material, and in 1997 SNL completed a preliminary PA (Cochran, Brown, et al. 1997) that identified several issues; however, a full PA has not been conducted to date.

6.2 First Iteration (1993)

In 1989, DOE tasked SNL to conduct a preliminary PA to determine the technical feasibility of the GCD disposal concept, examine the usefulness of existing data and information, and identify significant uncertainties in order to prioritize future site characterization activities. The PA, which was completed in 1993 by Price et al. (1993), consisted of (1) system description, (2) scenario development and screening, (3) process and pathway identification (4) consequence modeling, and (5) uncertainty and sensitivity analyses. Results of the sensitivity analyses were used iteratively to guide site characterization activities for the next PA iteration.

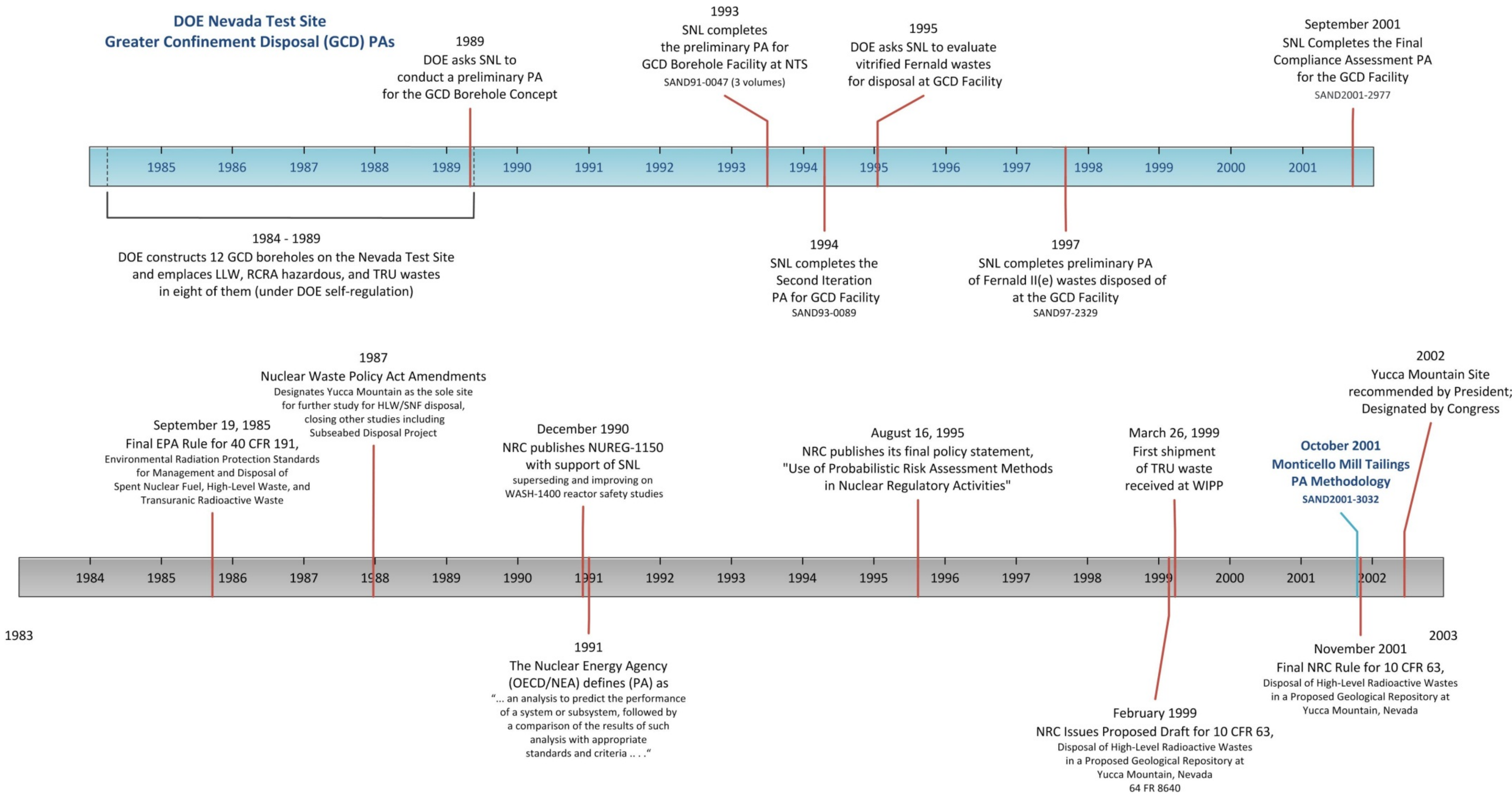


Figure 51. Greater Confinement Borehole PA timeline

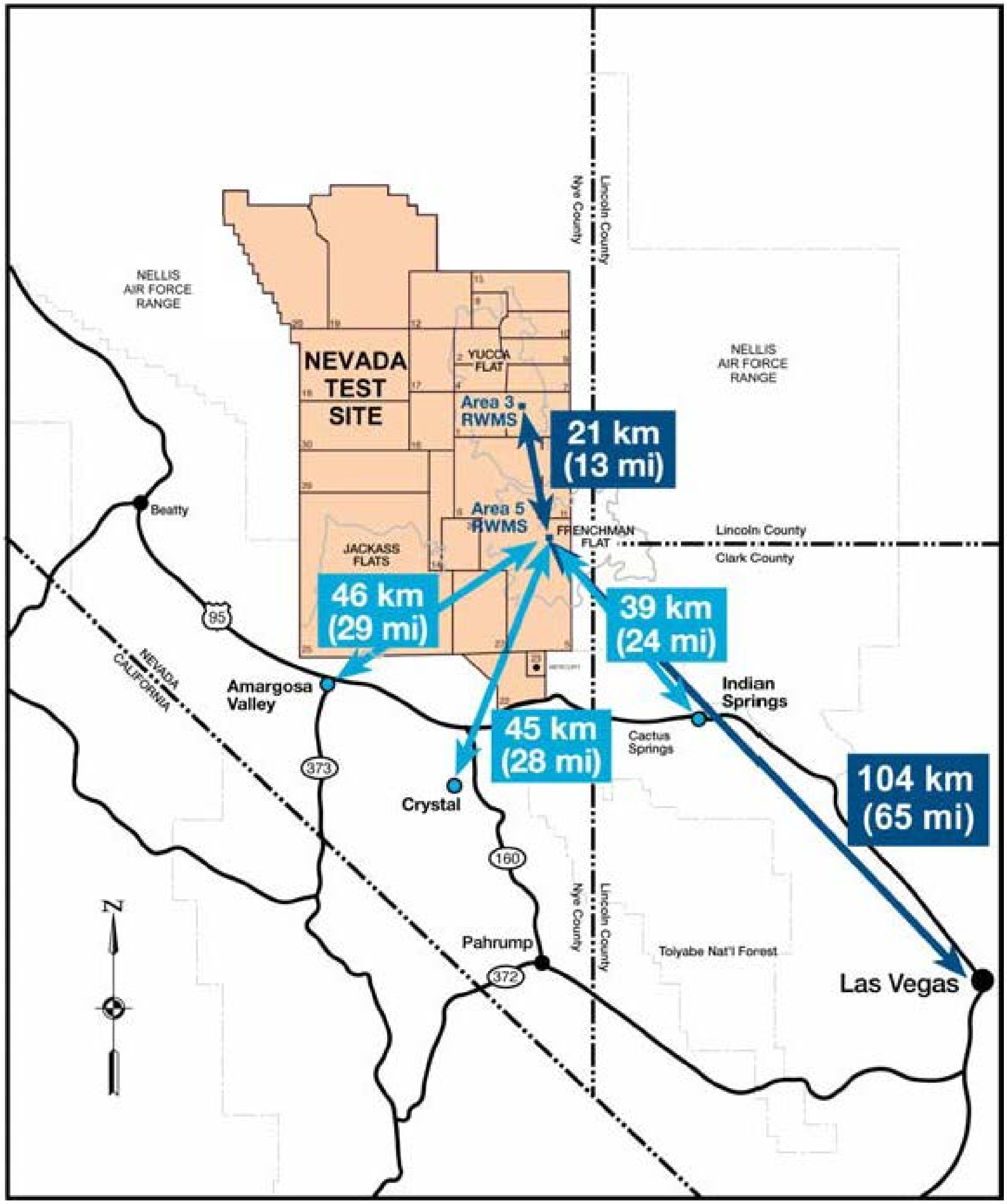


Figure 52. Location of Area 5 Radioactive Waste Management Site (RWMS) on the Nevada National Security Site (formerly Nevada Test Site) and GCD boreholes and distances to nearest populations

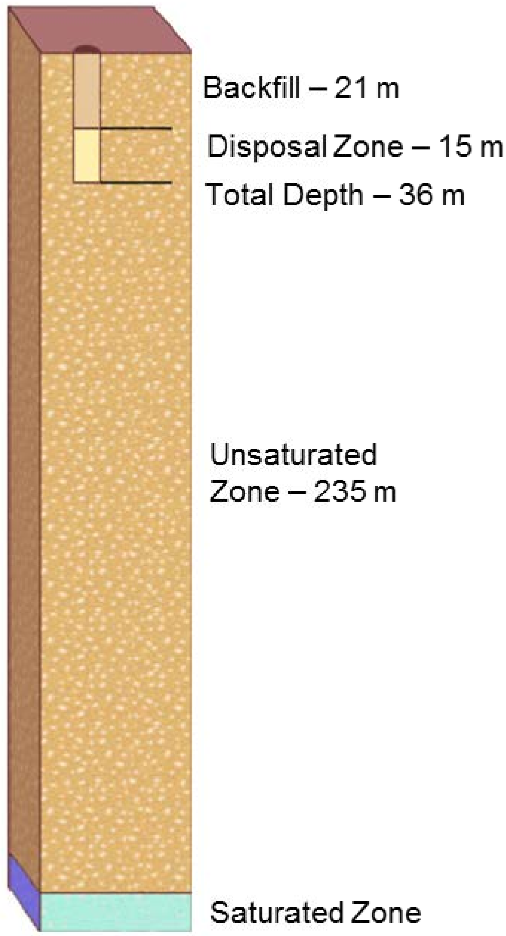


Figure 53. Schematic illustration of Greater Confinement Disposal boreholes

Forty-nine events and processes were identified that could affect the long-term performance of the GCD site. These were screened on the basis of physical reasonableness, probability of occurrence, and consequence. The final list included 12 events and processes. All potential combinations of these 12 events and process yielded 4,096 scenarios; therefore, to reduce the number of scenarios for evaluation, each scenario would consist of the occurrence of one of the 12 events or processes, rather than combinations of these 12 events. The three scenarios were then selected from the 12 based consequence and probability of occurrence relative to the others. The three scenarios selected were climate change, erosion, and human intrusion. In addition, a base case scenario was conducted. Each scenario was assumed to have a probability of one and the scenarios were compared to each other rather than summing probability-weighted scenarios to calculate a cumulative release.

Three performance measures derived from the requirements in 40 CFR Part 191 were evaluated:

- Individual protection requirements, specifying a maximum annual dose to any member of the public in 1,000 years;
- Groundwater protection requirements, specifying maximum concentrations of radionuclides in significant sources of water averaged over a given year for 1,000 years; and
- Containment requirements, specifying limits on cumulative release, in term of curies, of specific radionuclides to the accessible environment for 10,000 years.

Models for the scenarios were developed and three radionuclide transport processes were identified: convection, diffusion in the liquid phase, and diffusion in the vapor phase (for radon only). These transport processes resulted in nine different radionuclide migration pathways:

1. Flow of contaminated groundwater down through the unsaturated zone and laterally in the saturated zone to a point 5 km from the GCD site,
2. Diffusion of radionuclides in the liquid phase up to the ground surface (when recharge is negligible),
3. Diffusion of radionuclides in the liquid phase (when recharge is negligible) up to the roots of plants and transport by the roots to the portion of the plant that is above the ground surface,
4. Flow of contaminated groundwater down through the unsaturated zone to the saturated zone,
5. Diffusion of radionuclides in the liquid phase (when recharge is negligible) up to the roots of plants consumed by animals that are subsequently consumed by humans,
6. Vapor phase diffusion of radon up to the ground surface,
7. Vapor phase diffusion of radon up to the roots of plants, uptake of radon daughter products by the roots, consumption of plants by animals, and consumption of animal products by humans,
8. Vapor phase diffusion of radon down to the Valley Fill Aquifer, and
9. Vapor phase diffusion of radon downward and convection of daughter products down to the Valley Fill Aquifer.

For the consequence analysis, computer codes were selected and the analyses were automated to reduce the potential for human error and to provide traceability. Radionuclide transport was simulated using the computer code NEFTRAN, which considers 1-D convective transport for multiple radioactive decay chains with multiple members. Analytical solutions were used for 1-D liquid-phase diffusive transport to simulate the upward diffusion of radionuclides and for 1-D gas-phase diffusive transport of radon (a daughter product). The cumulative release for each radionuclide is normalized by the release limits listed in 40 CFR Part 191 which are based on the amount of waste disposed of. One of the considerations in doing the analyses was whether or not non-TRU waste (i.e., ^{90}Sr, ^{137}Cs, ^{226}Ra, and ^{227}Ac) should be included in the initial GCD inventory (isotopes of uranium were considered to be TRU waste, although by definition uranium is not transuranic). Results were presented for both TRU-only waste and TRU-plus-non-TRU wastes.

Parameter values were selected from the available data and information and occasionally expert judgment was used to identify parameter values where no data existed. Monte Carlo simulation was used to propagate the effect of parameter uncertainty through the models. The PDFs for uncertain parameters were sampled using LHS. For the base-case and the climate-change scenarios, 4,000 vectors were used, and 1,000 vectors were used for the human-intrusion scenario.

For the base case, 3,889 of the 4,000 realizations generated using LHS resulted in no radionuclide release to the accessible environment, implying that the probability of any one of those radionuclides reaching the accessible environment in 10,000 years was less than 3%. However, the value of the EPA sum corresponding to a probability of 0.001 (i.e., the value that causes the CCDF curve to go through the cross-hatched area) was 11.7, slightly more than the limit of 10.0. The highest EPA sum for the base case was about 460, corresponding to a cumulative release of about 23 curies. The isotopes ^{239}Pu and ^{240}Pu accounted for 98% of the released radionuclides when convection was the dominant mechanism and for 89% of the released radionuclides when liquid-phase diffusion was the dominant transport mechanism. Most of the TRU inventory (in terms of curies) consisted of plutonium. Liquid-phase diffusion to the ground surface was the dominant transport mechanism in only 46 of the 4,000 samples, and of those 46 samples, 22 resulted in nonzero cumulative releases. Furthermore, the five highest EPA sums were the result of liquid-phase diffusion (Price, et al. 1993).

For the climate change scenario, it was assumed that the climate would become cooler and wetter. The highest EPA sum was about 5,206, corresponding to a cumulative release of about 231 curies, of which 230 curies were ^{239}Pu and ^{240}Pu, as expected because the initial inventory (in terms of curies) consisted primarily of plutonium. Slightly more than half of the 4,000 samples for this scenario resulted in releases of radionuclides to the accessible environment in 10,000 years (i.e., nonzero EPA sums), which was significantly more than for the base case scenario. Liquid-phase diffusion was the dominant transport mechanism in only 10 of the 4,000 simulations, compared to 46 of 4,000 for the base case scenario, and all 10 of those samples had releases to the accessible environment. The highest EPA sum resulting from the liquid-phase diffusion pathway was 47, indicating that liquid-phase diffusion was not as important in the climate change scenario as it was in the base case.

For the erosion scenario, it was assumed that erosion would remove enough sediment over the next 10,000 years to expose the shallowest GCD wastes at a depth of 21.3 m (70 ft). Results of the erosion scenario consequence analyses were identical to those of the base case scenario.

For the human intrusion scenario, it was assumed that an 8-inch (20-cm) diameter hole is drilled through the center of one of the GCD boreholes at 100, 200, 300, 500, and 1,000 years after closure and radionuclides entrained in the drilling fluid are brought to the ground surface. It was further assumed that only one intrusion event would occur during the 10,000-year regulatory period, and each GCD borehole was assumed to have the same chance of being drilled into. Finally, waste was assumed to be uniformly mixed within a given borehole.

For the human intrusion scenario, none of the samples yielded a zero release. The highest EPA sum was about 20, resulting primarily from the release of 0.60 curies of ^{239}Pu and ^{240}Pu at an intrusion time of 100 years. The shape of the curve and the highest EPA sum did not change much for the other intrusion times (i.e., 200, 300, 500, and 1,000 years) because the TRU waste is very long-lived and does not change substantially over a few hundred years.

The preliminary PA concluded that the Nevada National Security Site GCD site would most likely comply with 40 CFR Part 191 for the base case (undisturbed) condition. The sensitivity analyses identified downward recharge rate as the most important parameter, followed by the sorption and solubility of the plutonium isotopes. The erosion scenario was concluded to be unimportant as the results did not change from the base case. The human intrusion scenario was nontrivial for the entire regulatory period (10,000 years), but since no accompanying probability was determined for this event, the effects of this event on compliance could not be addressed. A very conservative model of climate change with greatly increased groundwater recharge was used for the human intrusion scenario, and it was found that compliance was equivocal given the probability of a change in climate within the regulatory period.

6.3 Second Iteration (1994)

Building upon the results of the preliminary PA, a second iteration was completed in 1994 by Baer et al. (1994), incorporating site-specific data and information. The sensitivity analyses from the preliminary PA identified downward recharge rate as the most important parameter, suggesting the need to collect site-specific data on the groundwater recharge rate within Frenchman Flat. Therefore, for the second PA iteration, recharge rates were inferred from measurements of the concentrations of three natural environmental tracers within several boreholes drilled at Frenchman Flat. Based on these measurements, it was concluded that recharge was very small and would not allow transport of radionuclides to the water table within the regulatory timeframe. Incorporation of this new recharge data effectively eliminated recharge within the base case scenario and removed the entire downward advective pathway from further consideration in that scenario.

Because the downward advective pathway was eliminated, the remaining pathway, upward diffusion of contaminants in the liquid phase to the surface, coupled with diffusion into the root zone and adsorption and transport by vegetation, became more important. Therefore, for the second PA iteration a more refined plant transport model was developed and a simple model of erosion was incorporated because erosional processes have a larger effect on the total release in the new conceptual model. Probability distributions for uptake factors, surface biomass densities, rooting depths, and erosion depths were developed from available data to support these new models. In addition, site-specific measurements of near-surface moisture contents were analyzed and expressed in terms of a PDF.

The increased importance of the liquid diffusion pathway suggested that the tortuosity distribution used in the preliminary PA be reevaluated based upon existing correlations between moisture content and tortuosity. This parameter accounts for the slowed diffusion in a convoluted sample of porous media with respect to unrestricted liquid. The sensitivity analyses indicated that the tortuosity parameter was the most significant in terms of release; the largest releases were always associated with the smallest values of tortuosity.

At high EPA sums, it was found that deep plant roots could have a considerable impact on the site. The critical rooting depth is approximately 50 feet (15 m) at which point the CCDF starts becoming sensitive to rooting depths. Although these types of plant communities do no not currently exist at Nevada National Security Site that could change as a result of a change in climate.

The sensitivity analyses from the preliminary PA also identified sorption and solubility of the plutonium isotopes as significant parameters. Additional research found that, in general, plutonium isotopes in the GCD environment were less soluble and adsorbed more strongly to the surrounding media than originally estimated for the preliminary PA. Incorporation of the new plutonium data significantly changed the results of the second GCD PA iteration. The plutonium isotopes no longer dominated the release activity, being replaced by ^{230}Th, ^{234}U, ^{226}Ra, and ^{210}Pb as the dominant releases in terms of curies.

The second PA iteration considered the consequences and probability of an inadvertent human drilling event. A simple Poisson model that had been developed for the WIPP PA was used compute the probability of such an event. This analysis concluded that a human intrusion event would result in a release that would slightly violate the regulations. However, the probabilities were quite small. In addition, the analysis assumed a very conservative value for the drilling density in Frenchman Flat.

The new conceptual model and updated and additional parameter distributions were used in the uncertainty analysis. A total of 10,000 realizations of sampled probabilistic parameters were completed for the second PA iteration. Each set of parameter realizations was input into the models and an EPA sum was calculated. The sensitivity analyses indicated that the tortuosity parameter was most significant in terms of release. The PA results complied with the containment requirements, individual protection requirements, and groundwater protection requirements of 40 CFR Part 191. However, the results did not include the effects of climate change within the base case conceptual model.

6.4 Final PA Iteration (2001)

Building upon the results of the second iteration, the final GCD PA iteration was completed in 2001. FEPs screening began with comprehensive lists of over 760 processes and events and resulted in identifying a master list of 205 FEPs to begin the screening analysis. The screening analysis concluded with the inclusion of 28 individual FEPs into four significant scenarios that represented climate change, subsidence of the waste and overlying alluvial fill, and two scenarios involving inadvertent human intrusion. The final GCD PA iteration identified that operation and closure of the Area 5 Radioactive Waste Management Site would result in future landfill subsidence which could affect runoff, resulting in the downward movement of pore water and the formation of ephemeral wetlands. In addition, climate change could result in downward movement of pore water and the return of open piñon-juniper woodlands. Therefore, a detailed screening analysis was conducted as part of the final PA to determine the effects of subsidence and the eventual return to a glacial climate. Both subsidence and climate change would move moisture (and radionuclides) away from the land surface and deeper into the vadose zone (with a decrease in releases to the accessible environment). However, surface water would not reach the water table in 10,000 years.

The Final GCD PA used a relatively simple computational model which was implemented in Microsoft Visual Basic macros in a Microsoft Access database, referred to as “the Unnamed Code” (TUC). Using the conceptual model illustrated in Figure 54, the final PA code modeled a continuation of current conditions (with upward advection of the pore water), coupled with deeper-rooted, glacial-climate plant species, which overestimates the releases. This PA model was built from a mathematical expression for mass conservation that includes the operation of a number of transport processes, including dissolution, precipitation, reversible chemical sorption onto soil, advection, diffusion, dispersion, radioactive decay and ingrowth, plant uptake, and bioturbation. Two independent benchmarking exercises initiated by DOE-Nevada comparing the Unnamed Code against Bechtel-Nevada’s Area 5 Radioactive Waste Management Site Composite Analysis Model and against a GoldSim-based model developed by Neptune and Company showed that the Unnamed Code conservatively overestimated releases in comparison to the other codes. What differences were found, in conclusion, raised no major concerns and the results were “generally compatible” or showed “reasonably close output results” in comparison to the alternative codes (Cochran, Beyeler, et al. 2001).

The calculations were completed in two phases. The first calculated the movement and cumulative release of summed radionuclides over 10,000 years and produced a CCDF of the EPA Sum for assessment against the containment requirements of 40 CFR Part 191. The second calculated the cumulative radionuclide release over 1,000 years and translated those releases to doses for assessment against the requirements of the individual protection requirements of 40 CFR Part 191. A total of 5,000 realizations of sampled probabilistic parameters were completed, and the resulting CCDF was well within the limits specified in 40 CFR Part 191, as shown in Figure 55, which presents results for comparison against the containment requirement limits.

Probability distributions of doses were estimated for the individual protection requirements for two exposure conditions: an offsite resident farmer and an on-site homebuilder. Doses were estimated conservatively from the cumulative releases summed over 1,000 years and included doses from radon—the only U.S. PA known to include radon in a dose assessment. The maximum calculated whole-body dose value was 0.16 mrem, and the largest calculated dose to any organ was 4.5 mrem, both far below the limits specified in 40 CFR Part 191 of 25 mrem for whole-body dose and 75 mrem for critical organ dose. Though 40 CFR Part 191 sets the compliance period of 10,000 years, The Federal Review Team requested additional information to examine the robustness of the modeled disposal system. To provide this additional information, the simulation time was increased from 10,000 years to 20,000 years. The results showed that doubling the simulation time allows more time for upward advection to move radionuclides towards the land surface, where the waste is more likely to be removed into the accessible environment by plant uptake and bioturbation, but it demonstrated that the rate that radionuclides reach the accessible environment does not increase dramatically after 10,000 years.

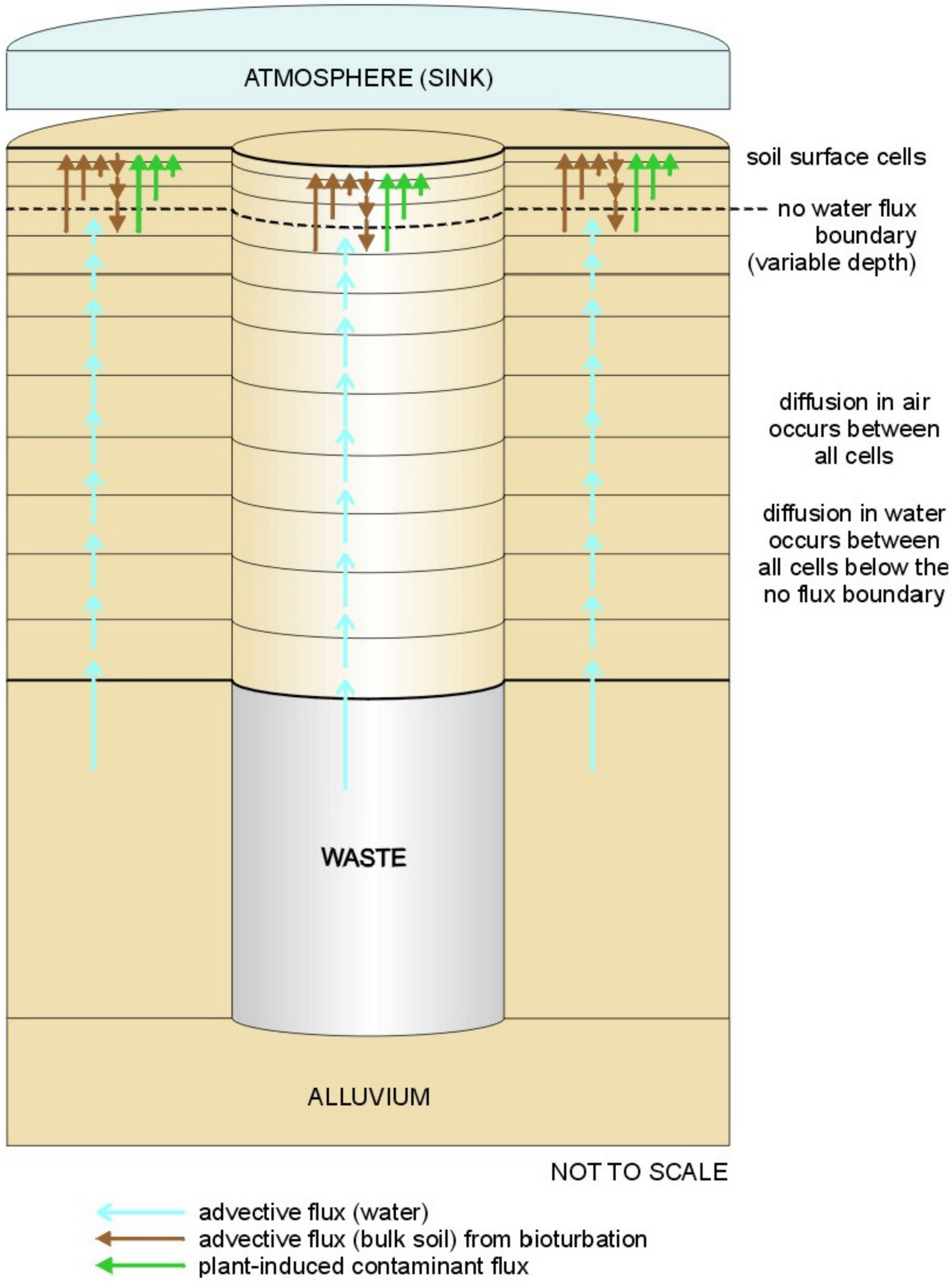


Figure 54. Conceptual model of a GCD borehole, with waste disposal zone configuration and hydrologic and biotic processes affecting the system

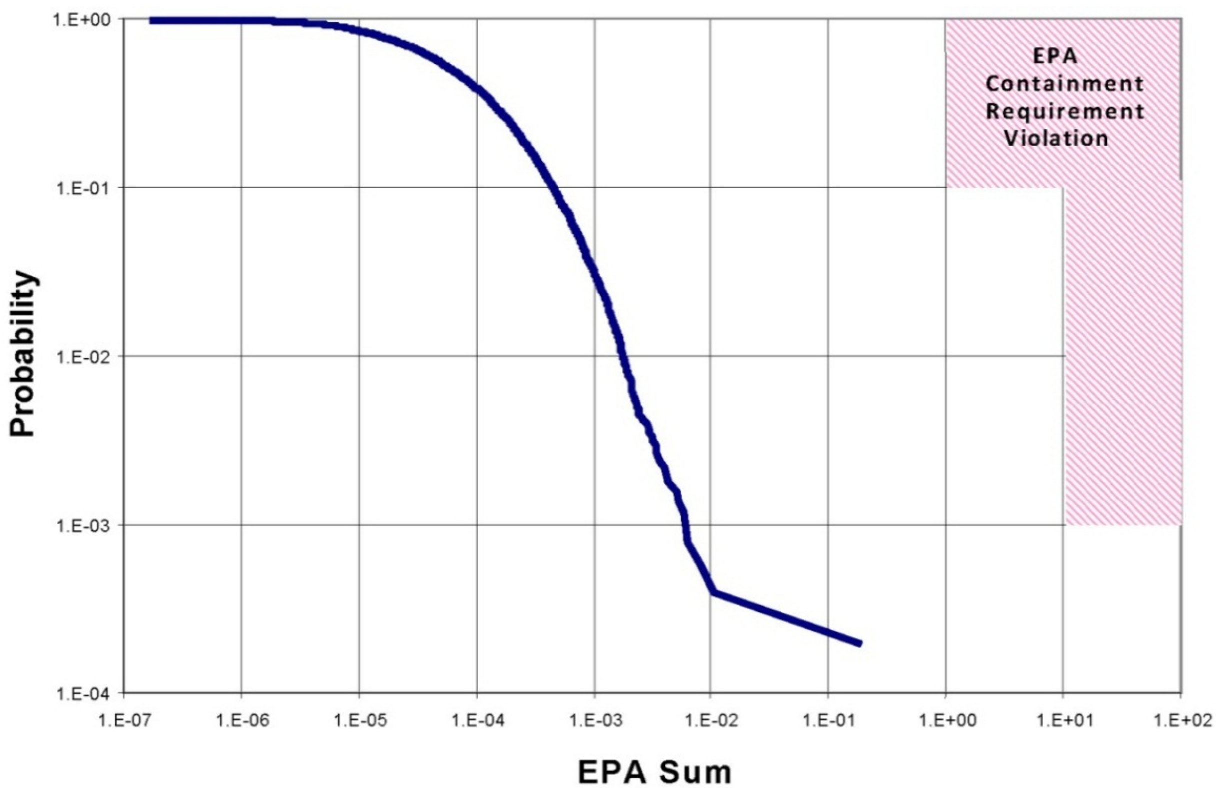


Figure 55. 2001 PA results for GCD system: CCDF for the PA Analysis of the TRU wastes in the GCD boreholes

6.5 Demonstration for Vitrified Fernald Waste (1997)

In January 1995, SNL was asked to evaluate the use of GCD boreholes at Nevada National Security Site for the disposal of vitrified Fernald byproduct material. In 1997 SNL completed a preliminary, or scoping, PA which included a description of the approach, regulatory analysis, selection of performance objectives, site and waste description, preliminary modeling results, and identification of issues and activities required to complete a full PA (Cochran, Brown, et al. 1997). Since this was not a full PA and it did not follow the probabilistic methodology, it will only be described briefly here.

The byproduct material is concentrated residue from processing uranium ore, which has been stored in three silos at the Fernald Environmental Management Project since the early 1950s and will be vitrified into 6,000 yd^3 (4,580 m^3) of glass "gems" prior to disposal. A significant portion of the byproduct material is composed of long-lived radionuclides, requiring some type of deep geologic disposal.

For the preliminary evaluation, a source term was developed, pathways and scenarios were developed, conceptual and mathematical models were developed, codes were identified, and a deterministic analysis was conducted. Based on this analysis, SNL concluded that performance standards would likely be met for the undisturbed case and the dose received by a hypothetical inadvertent intruder would likely be below the acute dose standard but above the chronic dose standard set by DOE Order 5820.2A. It is important to note that this hypothetical intruder would receive an unacceptable chronic dose independent of the site setting because it is primarily a function of the source term.

The preliminary PA left a number of issues remain unresolved; however, it did provide a strong foundation for developing a full PA that could support a final disposal decision. To date, a full PA has not been conducted for the disposal of vitrified Fernald byproduct material in the Nevada National Security Site GCD boreholes.

6.6 Significance of the DOE Greater Confinement Borehole Disposal PAs in the Historical Development of the PA Methodology

This iterative series of probabilistic PAs for the DOE special case wastes successfully demonstrated the use of the PA methodology for the GCD concept. It was a simple, efficient PA analysis that, in its final iteration, was implemented in a relatively simple computational model which was implemented in Microsoft Visual Basic macros in a Microsoft Access database, which was later benchmarked successfully against two other independently developed models.

On the basis of the Final GCD PA, the GCD boreholes at the Nevada National Security Site were approved as only the second site, after WIPP, to meet the environmental safety requirements of 40 CFR Part 191 for disposal of TRU waste, and this marked the first successful completion and acceptance of a PA for TRU waste under DOE self-regulation. The GCD boreholes at the Nevada National Security Site were the first site and are still the only site approved for intermediate-depth disposal of radioactive waste.

Since the time of these analyses, the GCD borehole disposal method has also been considered for greater-than-Class-C low-level radioactive waste (Tonkay, Joyce, and Cochran 2007; DOE 2011).

7. DEMONSTRATION FOR GEOLOGIC DISPOSAL OF DOE-OWNED SNF AND HLW STORED AT INL (1993–1998)

7.1 Introduction

Between 1993 and 1998, SNL evaluated the deep geologic disposal of HLW and SNF being stored at the Idaho National Laboratory (INL)[9] (Rechard 1993b, Rechard 1995, Rechard 1998). Initially, two generic fully-saturated geologic repositories, a bedded salt and a partially fractured granitic rock, were evaluated. However, the second and third PAs were for an unsaturated fractured tuff repository, basically the Yucca Mountain repository. The results of this series of PAs provided INL decision-makers with additional detail that could help guide them in preparing their stored wastes for permanent disposal. A timeline of the program, shown against the backdrop of other contemporary developments in PA is shown in Figure 56.

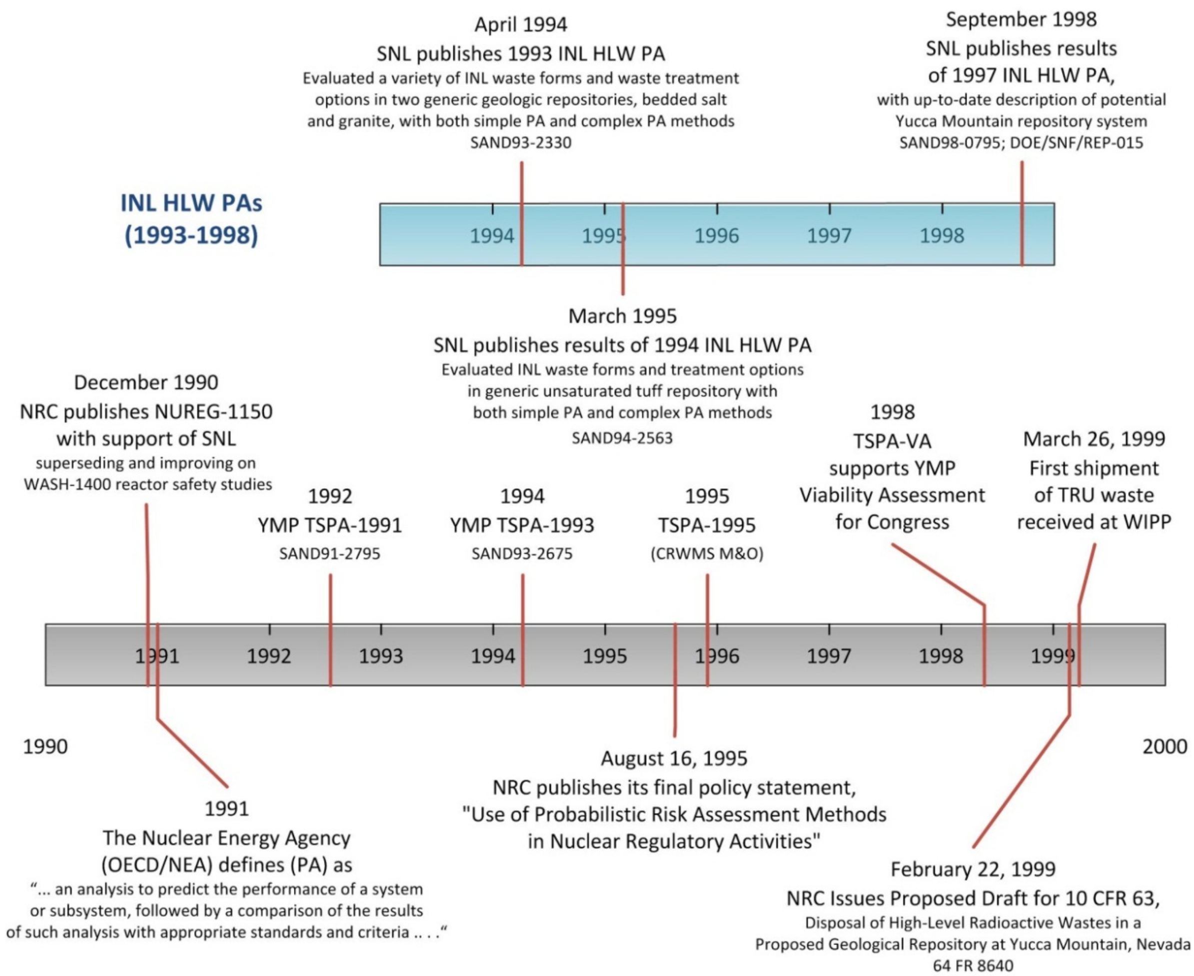


Figure 56. Timeline of PAs for DOE HLW at INL

[9] At the time of these studies, Idaho National Laboratory (the current name, since 2005, of the national laboratory) was known as Idaho National Engineering Laboratory and then, beginning in 1997, as Idaho National Engineering and Environmental Laboratory). For convenience, this report refers to the laboratory by its current name.

7.2 INL HLW PAs

The first two PA analyses of disposal of INL HLW were completed in 1993 and 1994 and both included both a simple and a complex PA, where “simple” meant that the wastes were evaluated individually in an analysis that used ordinary differential equations in a mathematical model and “complex” meant that the analyses used partial differential equations in a mathematical model. In the 1993 and 1994 iterations, the complex PAs used the codes and methodology developed for the WIPP Project (described in Section 4) and used the CAMCON system of linked codes. The simple PA relied heavily upon abstractions/simplifications of the complex processes taking place in the system, similar to the methodology employed on the YMP (described in Section 5). Both methods proved acceptable in evaluating disposal options and in developing waste management guidance for the DOE in handling its waste currently being stored at INL. The 1997 PA, however, did not include a simplified model. Instead, for the 1997 PA, which used the software configuration management approach used for the WIPP CCA PA rather than CAMCON in linking PA codes for total system analyses, the complex PA model developed by SNL could be compared to the YMP TSPAs then being developed by the DOE Office of Civilian Radioactive Waste Management, which used simplified, less interdependent PA models.

7.2.1 1993 Performance Assessment

In 1993, SNL evaluated two generic fully-saturated geologic repositories for the geologic disposal of HLW and SNF being stored at INL, one generic repository in bedded salt and the other in a partially fractured granitic rock (Rechard 1993b). The PA methodology followed the same methodology outlined previously and included the following elements:

1. System characterization
   - Waste containment system
   - Waste parcel description
   - Geologic barrier characterization
   - Repository design
2. Scenario development
   - FEPs development
   - Calculational design
3. Probability modeling
4. Consequence modeling
5. Regulatory assessment
6. Sensitivity/uncertainty analysis

Four PA analyses were conducted: (1) a complex PA for disposal in a bedded salt repository; (2) a complex PA for disposal in granite batholith repository; (3) a simple PA for disposal in bedded salt repository; and (4) a simple PA for disposal in granite batholith repository

The simple PA was a one-dimensional network using sophisticated, lumped-parameter models that included the following codes:

SALFLOW, GRFLOW → CLAMVD

The complex PA was a two-dimensional, finite-element modeling system based upon spatially distributed parameters that used the WIPP CAMCON (Rechard 1992) system of codes

(SANCHO → BRAGFLO) → PANEL → (GRASP-INV → SECO2D → SECO/TP) → [summation of type (a) consequences]

The waste forms included approximately 700 MTHM stored at the INL: graphite spent fuel, experimental low enriched and highly enriched spent fuel, and high-level waste generated during reprocessing of some spent fuel. Five different waste treatment options were studied; in the analysis, the options and resulting waste forms were analyzed separately and in combination as five waste disposal groups. When the waste forms were studied in combination, the repository was assumed to also contain vitrified HLW from three DOE sites for a common basis of comparison and to simulate the impact of the INL waste forms on a moderate-sized repository. The performance of the waste form was assessed within the context of a whole disposal system, against the containment requirements in 40 CFR 191.13, promulgated in 1985.

For the salt repository disposal system, the CCDFs for the waste parcel discharges were tightly grouped and were similar for the simple and complex PAs. The results from the complex PA were slightly higher at the lower values of activity discharge. Although, the activity discharges from the repository also had a tight grouping for the complex and simple PAs, the results from the complex PA were much lower. These differences were due to differences in the conceptual models for the simple and complex PAs.

For the granite repository disposal system, the CCDFs for the waste parcel discharges were slightly larger and more tightly grouped than those from the simple PA, which showed a separation of waste disposal groups 4 and 5 and disposal groups 2 and 3. The discharges from the granite repository for the simple and complex PAs were similar, with slightly more separation between disposal groups 4 and 5 and groups 1, 2, and 3 for the simple model and also more spread than the waste parcel discharges. Again, these differences were attributed to differences in the conceptual models used in the simple and complex PAs.

The results showed that both repository types had the potential to meet the performance criteria specified in 40 CFR Part 191. The salt repository met the criteria even without a moderately corrosion-resistant canister or chemically-adsorptive backfill. In the salt repository, differences among the individual waste forms were less pronounced than in the granite repository and the differences nearly disappeared when the waste forms were combined as a disposal group with vitrified HLW, suggesting that treatment of the relatively small mass of INL fuel and waste cannot substantially improve the performance of a moderate- or large-sized repository.

It was concluded that there was an extensive need for defensible data for future PA iterations, and it was recommended that future PAs evaluate an unsaturated volcanic tuff. The salt and granite disposal options would then be reviewed after the data, conceptual models, and computational models were improved so that more defensible predictions could be made.

7.2.2 1994 Performance Assessment

In 1994, SNL conducted another PA for the HLW waste stored at INL (Rechard 1995), but this time a hypothetical repository in unsaturated tuff was evaluated. The same objective existed, namely to assist the DOE in preparing its wastes for eventual permanent disposal. The hypothetical repository was similar to the potential repository at Yucca Mountain, Nevada. The performance criteria for the 1994 PA included: (1) containment requirements and individual protection requirements from 40 CFR Part 191, with an emphasis on maximum annual effective dose equivalent, and (2) technical criteria for the container and waste form as defined in 10 CFR Part 60.

The process for the 1994 PA was the same as that used in 1993 which was based upon the methodology developed for the WIPP PA. The 1994 PA took advantage of site characterization data used for the 1993 YMP TSPA, referred to as TSPA-1993 (Wilson, Gauthier, et al. 1994). A simple and complex PA were run, which is useful in establishing sensitive parameters for the PA. The simple PA took advantage of codes used in TSPA-1993 for the YMP, and the complex PA uses modified codes from the 1993 INL PA.

This PA also looked at the possibility and consequences of criticality in or near containers of highly-enriched uranium spent nuclear fuel. Ranges of temperature and water saturation were considered in preliminary reactor dynamics calculations in which phenomena were manually coupled. Other analysis approaches included summarizing data from historical criticality accidents to bound probable conditions and performing scoping calculations in which the repository was assumed to act like a reactor at steady state. The results showed the study's sensitivity and uncertainty analyses suggest that improved data could lower the estimates significantly.

The differences in the conceptual models between the simple and complex PAs are shown in Table 5 (Rechard 1995). These differences did not produce noticeable differences in the EPA summed normalized releases, which appear in releases from the waste package even though the same source-term model was used. The differences in results occur primarily because the corrosion rates were very sensitive to temperature and calculated temperatures near the disposal containers were different in the simple and complex PAs.

The PAs concluded that for a repository in unsaturated tuff, disposal of currently existing DOE SNF and HLW with minimal treatment or conditioning in a disposal container of carbon steel and Alloy 825, would behave similarly to commercial pressurized water reactor fuel and HLW in borosilicate glass. Furthermore, the waste complied with the overall dose and containment requirements of 40 CFR Part 191 over 10,000 years. ^{99}Tc was the major contributor to doses in the groundwater pathway in the first 10,000 years, but ^{237}Np was the most important at 100,000 years. The small inventory of ^{14}C in the DOE spent fuel helped in complying with the current containment requirements.

Table 5. Major differences in conceptual models in complex and simple PAs (1994 INL HLW PA)

	Complex PA	Simple PA
Saturated Zone	Dual porosity medium	Single porous medium
	Retardation of radionuclides by sorption on tuff	No retardation of radionuclides
	Three-dimensional model used directly in analysis	Abstraction of velocity distribution and dispersivity from three-dimensional model
Unsaturated Zone	Liquid flow	
	Two-phase flow	Single-phase flow
	Coupled heat convection and conduction	Uncoupled heat conduction
	Two-dimensional model	One-dimensional model
Gaseous flow	Identical to liquid flow	Abstraction of gas velocity from two-dimensional results reported in TSPA 1991 (Barnard, Wilson, et al. 1992)

7.2.3 1997 Performance Assessment

The earlier PA analyses examined specific treatment options and disposal in two hypothetical repositories and the direct disposal option in a relatively small but Yucca Mountain–like repository described above. A major focus of the 1997 PA was to improve the understanding of spent fuel performance in an unsaturated tuff repository by including the most current description of the potential Yucca Mountain disposal system. The 1997 PA (Rechard 1998) assembles data and then evaluates the performance after disposal of 13 separate DOE SNF categories in containers with defense HLW (i.e., the codisposal option) and two commercial spent fuel categories.

Data for site characterization in the 1997 PA were taken primarily from SNL's TSPA-1993 for Yucca Mountain (Wilson, Gauthier, et al. 1994), with updated data in several areas such as an order-of-magnitude increase in the average precipitation infiltration and a two order-of-magnitude decrease in neptunium volubility. In general, the data used in the 1997 PA are similar to those used in a TSPA-VA. Goals of 1997 PA (Rechard 1998) were to:

1. Identify the behavior, after permanent disposal, of the spent nuclear fuel (SNF) now under the jurisdiction of DOE Office of Environmental Management. It evaluated whether this DOE SNF performs better or worse than commercial SNF, which can be used as a benchmark in the absence of explicit acceptance criteria. The disposal system modeled is the potential repository at Yucca Mountain, Nevada, containing commercial SNF, defense HLW, and DOE SNF. The PA assumed the codisposal of DOE SNF, without treatment, with vitrified defense HLW, in which DOE SNF is packaged with defense HLW in the same disposal container.
2. Identify the most sensitive parameters through analysis of the results to determine which DOE SNF characteristics should be carefully estimated or measured and which could be neglected, after demonstrating their minor influence. Such information is useful for developing performance-based requirements for repository acceptance criteria, that is, defining characterization requirements only for those spent fuel types

and parameters that demand them, thus substantially reducing data gathering and preparation costs for DOE SNF.

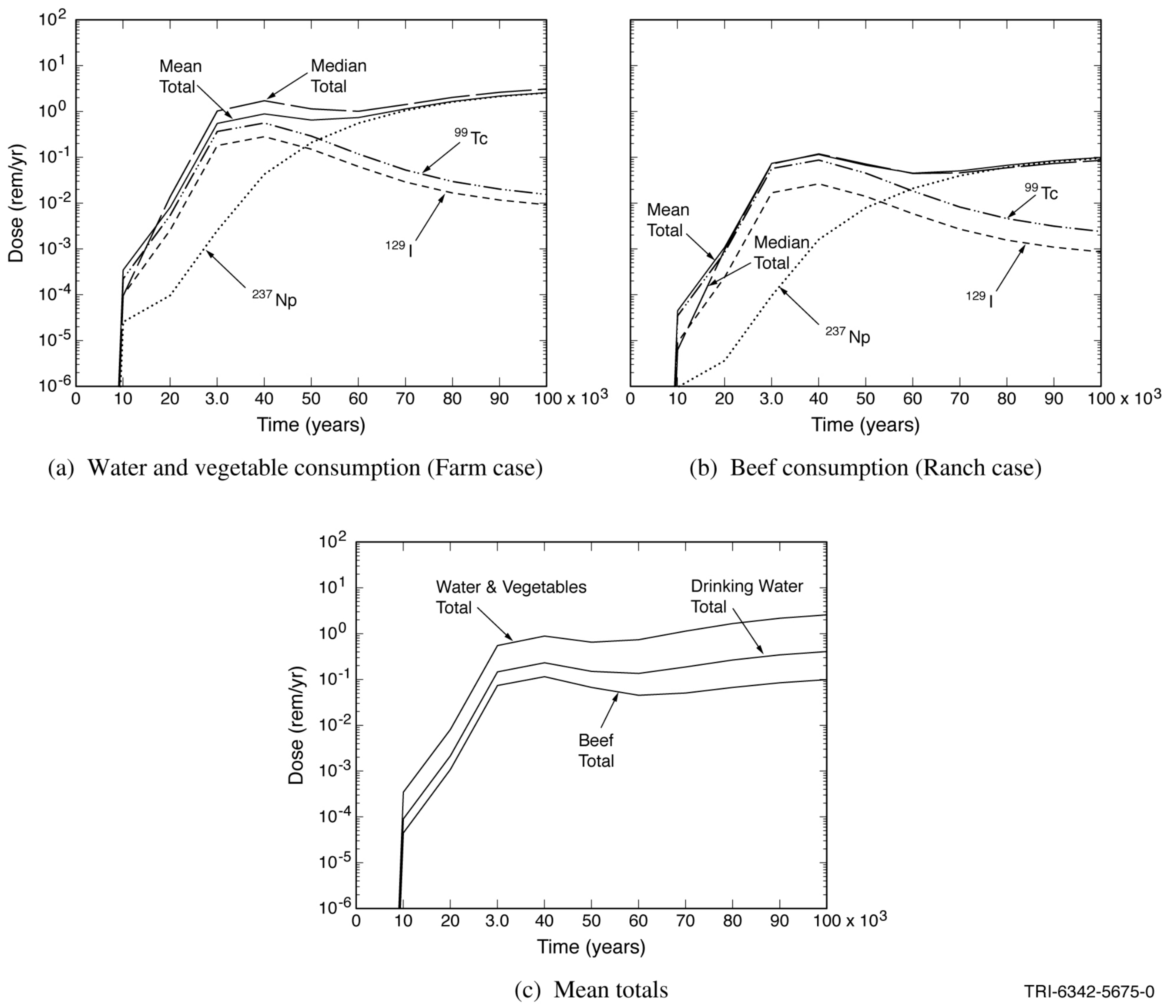
The methodology for the 1997 INL PA was the same as that used in the previous 1993 and 1994 PAs, which, in turn, were based upon the methodology originally developed for the WIPP PA. Among the major differences between the 1997 INL PA and the 1993 and 1994 PAs was modeling the actual Yucca Mountain site as the potential repository, with 75,320 MTHM of waste including 63,080 MTHM of commercial fuel, rather than the small, hypothetical repository modeled in the 1994 PA, located east of the Ghost Dance Fault and containing only DOE-owned SNF and HLW, thereby excluding effects from commercial SNF. In addition, the 1997 PA modeled 15 spent nuclear fuel categories, the codisposal option was used for the DOE-owned SNF and HLW, and dose calculations considered a 100,000-year performance period. Finally, the model for degradation of DOE SNF was enhanced in the 1997 PA. The upgraded model included transport of O_2 in order to determine whether O_2 was limited enough to reduce the rate of degradation of containers or waste.

The 1997 PA contained only a complex PA, and did not include a simplified model for comparison as was done in 1994. Instead, for the 1997 PA, the complex PA model developed by SNL could be compared to the YMP TSPAs then being developed by the DOE Office of Civilian Radioactive Waste Management, which used simplified, less interdependent PA models. To facilitate the comparison with YMP TSPAs, the 1997 INL PA used assumptions that matched those used by the management and operating contractor for the DOE Office of Civilian Radioactive Waste Management in PA sensitivity analyses for DOE SNF conducted in 1997 (CRWMS M&O 1997), namely, the 1997 INL PA neglected credit for cladding of SNF, assumed a similar radionuclide inventory, and used updated geologic data.

To improve traceability and repeatability, the 1997 PA used a centralized database for all parameters in its models. The database included about 3,024 total parameters: 2,755 assigned constants and 269 assigned distributions, 63 of which were varied in the 87 realizations run for the system analysis. The prior INL PA iterations had applied CAMCON, the system controller developed for the WIPP PAs, to link submodels together for the system calculation. For the 1997 INL PA, CAMCON was replaced with the software configuration management system applied shortly before on the WIPP CCA PA (see Section 4.4).

For the dose calculations, the 1997 PA calculated annual dose to a receptor for three cases: (1) a "ranch case," in which a rancher is exposed to radionuclides by means of beef consumption only; (2) a "farm case," in which a member of a farming family is exposed to radionuclides by means of drinking water and food consumption, as well as inhalation, and (3) a "small community case," in which an average resident is exposed through drinking water and consumption of locally grown farm products such as vegetables, fruits, dairy, and meat products. For all cases, dose was evaluated from peak concentrations of the transported radionuclides (^{129}I, ^{237}Np, and ^{99}Tc) at a 5-km boundary over a 100,000-year period, as if the rancher, farmer, or small community had drilled a water well into the saturated zone at the point on the 5-km boundary with the highest radionuclide concentrations.

Figure 57 presents median and mean total dose results for the farm case (a subsistence farmer consuming water and vegetable) and the ranch case (a rancher consuming beef) along with contributions from ^{129}I, ^{237}Np, and ^{99}Tc, and also shows the total mean annual dose for the ranch and farm cases as well as a drinking-water-only case. Calculated doses from ^{99}Tc and ^{237}Np dominated the results, with ^{99}Tc causing a temporary local peak (about 1 rem, for the farm case) that occurs at 40,000 years, with ^{237}Np driving doses above that earlier peak beginning at 70,000 years and continuing to increase through 100,000 years (to about 1.5 rem/yr, for the farm case). The doses primarily from drinking water for a critical subpopulation using the aquifer for its water supply were roughly seven times less than for the subsistence farmer (i.e., the farm case) but still more than for the rancher consuming beef (i.e., the ranch case). Similar to the farm case, the drinking-water-only case showed a temporary local peak dose rate of 250 mrem/yr from ^{99}Tc at about 40,000 years and a similar dose rate at about 70,000 years that continued to increase up to 400 mrem/yr at 100,000 years (Rechard 1998). (For comparison, the peak dose in contemporaneous PA sensitivity analyses for DOE SNF conducted by the YMP management and operating contractor (CRWMS M&O 1997) was 80 mrem/yr at 18,000 years from ^{99}Tc, but doses from ^{237}Np remained below that rate through the 100,000-year performance period.)



(a) Water and vegetable consumption (Farm case)

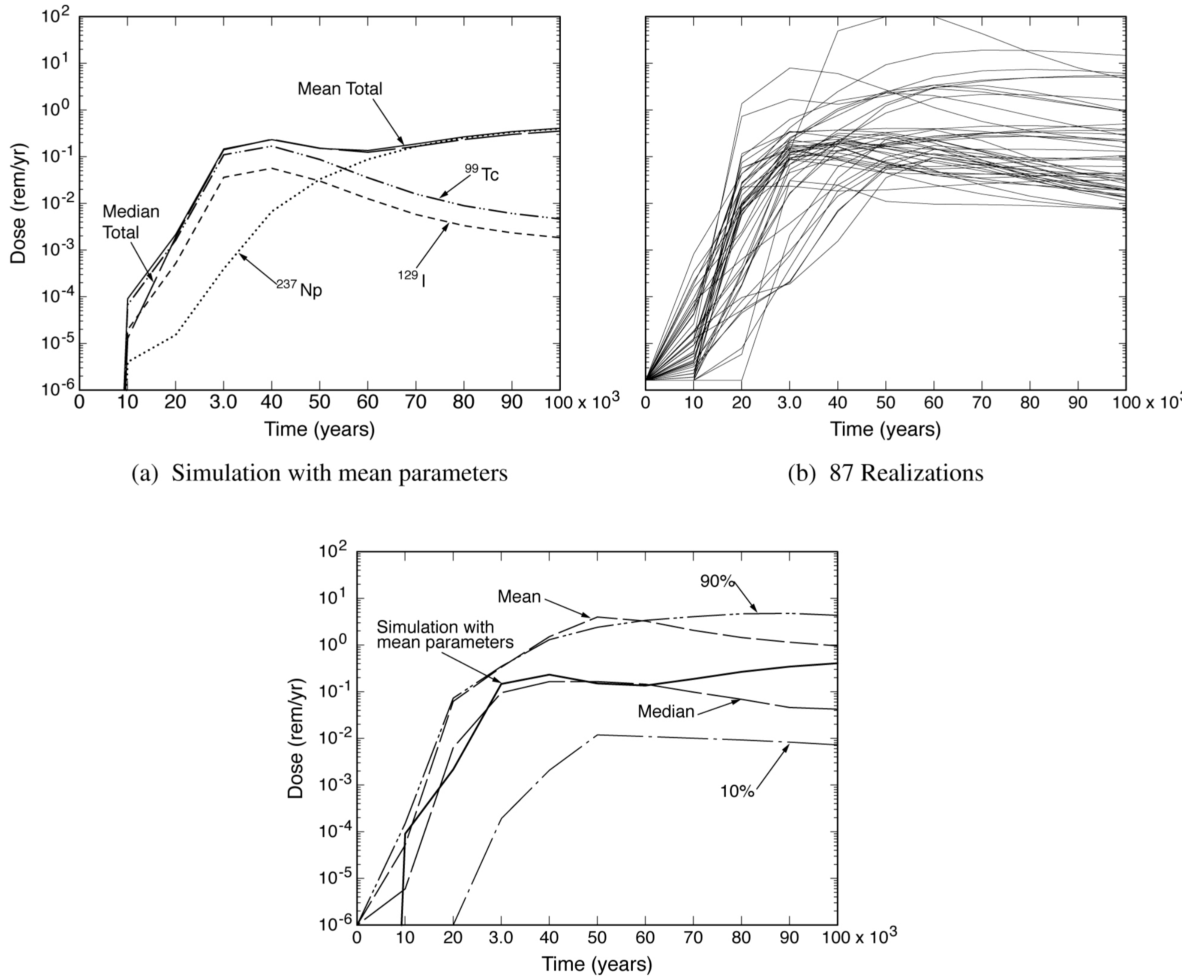
(b) Beef consumption (Ranch case)

(c) Mean totals

Figure 57. 1997 INL waste forms PA results: probabilistic annual dose as a function of time for (a) the "farm case" of water and vegetable consumption; (b) the "ranch case" of beef consumption; and (c) mean totals for both cases and for doses from drinking water

The contribution to the total ^{237}Np inventory from DOE SNF and defense HLW was about 680 curies (2% of the total). Because all cladding was assumed failed in the 1997 PA base case, the contribution of DOE SNF and defense HLW to the total inventory is also their contribution to the total dose.

As a measure of the nonlinearity of the modeling system, a deterministic (single-realization) simulation was run using mean parameter values for comparison with the probabilistic simulation with 87 realizations (Figure 58). The difference between the mean of the 87 realizations and the deterministic simulation using mean parameters was greatest at early times when ^{99}Tc dominates doses, implying that uncertainty in the nonlinear aspects of the source term model was dominating the variation. At later times, when two-phase flow and temperature have stabilized, uncertainty in the linear solubility limit appeared to dominate the variation.



(a) Simulation with mean parameters

(b) 87 Realizations

(c) Quantiles on 87 realizations

Figure 58. 1997 INL waste forms PA results: annual dose from drinking water as a function of time for (a) deterministic run using mean parameter values, (b) 87 realizations in 1997 PA, and (c) mean, median, and 10th and 90th percentiles of the probabilistic simulation compared to the deterministic simulation using mean parameters

In general, the consequence model was shown to be moderately nonlinear in representing the dose from ^{99}Tc. A difference of a factor of 20 existed between the ^{99}Tc peak of 4 rem/yr at 50,000 years for the mean of the 87 realizations and the peak of 200 mrem at 40,000 years for the deterministic mean realization, while a factor of 2.5 difference existed between ^{237}Np doses at 100,000 years for the mean of the 87 realizations (1 rem/yr) and for the deterministic mean-parameter simulation (400 mrem/yr). Because the variation in water flow was small in the first 10,000 to 20,000 years and, thus, could not have caused the variation in peak releases of ^{99}Tc, this difference in variation of ^{99}Tc and ^{237}Np implied that uncertainty in aspects of the source term (i.e., kinetics of release) more strongly influenced peak releases of ^{99}Tc than ^{237}Np.

Although the intention of the 1997 PA was to study only the relative performance of commercial and DOE SNF, the study's estimates of doses allowed for comparison with other studies. In addition, the study's sensitivity and uncertainty analyses suggested that improved data may lower the estimates significantly. The 1997 PA showed that the potential dose up to about 50,000 years depended on releases of ^{99}Tc, while at later times it is due to ^{237}Np. The uncertainties in the study's source term affected the estimates of peak releases of ^{99}Tc more strongly than the estimates for ^{237}Np. The 1997 PA showed that improving the accuracy of the alteration rate of the SNF matrix or the fraction of ^{99}Tc in gaps of the fuel would reduce this uncertainty in the estimated dose from ^{99}Tc. (The other two parameters influencing ^{99}Tc release rates were solubility of technetium and the amount of water flowing through the waste container.)

The results also indicated that the estimates of the doses from ^{237}Np may decrease if more accurate values for several parameters were provided. If the actual mean volubility of ^{237}Np is significantly lower than the assumptions in the 1997 PA, the longer-term doses would be lower. (Additional accuracy in the amount of ^{237}Np present was not shown to be necessary, especially for the 2% represented by the DOE SNF fuel.) The rate of fluid flow through the mountain and the number of containers contacted by water also contributed heavily to the doses from ^{237}Np. The PA results indicated that reduced values for any of these three parameters—solubility, fluid flow, and number of containers in contact with water—would improve the system's compliance with a future dose standard.

The results of the 1997 PA demonstrated that the mass and the activity of the DOE SNF in the repository are modest in relation to those of the commercial SNF and that that the unique characteristics of DOE SNF do not outweigh this relationship: they do not adversely influence the behavior of the disposal system. The effects of the characteristics unique to DOE SNF are further diminished because the radionuclides in the codisposal waste package are completely dominated by the defense HLW radionuclides. Therefore, DOE SNF would be expected to meet repository acceptance criteria if commercial SNF can meet them, and the direct disposal of DOE SNF could remain the primary option considered by the DOE Office of Environmental Management for disposal in the then-proposed geologic repository at Yucca Mountain.

7.3 Significance of INL HLW PAs in the Historical Development of the PA Methodology

The INL PAs were uniquely significant because they demonstrated two different methods of conducting a PA, the so-called “simple” PA with sophisticated, lumped-parameter models and the “complex” PA based on spatially distributed parameters. The complex PAs from the 1993 and 1994 analyses showed the benefit of including more physics in the high-level PA simulation with less reliance upon simplifications or abstractions. However, the simple PAs from the 1993 and 1994 analyses showed that simplifications of more complex models could accurately describe the behavior of a very complex system. The process for the 1997 PA was the same as that used in the complex PAs from 1993 and 1994, incorporating complex models directly into the probabilistic analysis to capture spent fuel behavior as accurately as possible, but the 1997 PA did not include a corresponding simple PA, as the earlier PAs had. Instead, SNL’s 1997 INL PA complemented the DOE Office of Civilian Radioactive Waste Management PAs for YMP by providing additional, detailed information for the YMP analyses (see Section 5), which used simplified models that have been abstracted from detailed simulations. To provide a complete picture of DOE SNF performance in the repository, the 1997 PA results could be compared with the results from contemporary YMP PA analyses. Contrasting the results of the two methodologies provided a benchmark-type comparison, helping to add confidence regarding modeling and serving to identify the relative benefits of having a streamlined PA system.

These PAs helped define appropriate requirements for waste characterization and waste treatment options for DOE-owned SNF being accepted for disposal as a part of the broader DOE National Spent Nuclear Fuel Program in developing a safe, cost-effective technical strategy for the management and disposition of the foreign and domestic spent nuclear fuel under DOE’s ownership.

8. OTHER APPLICATIONS AND ENHANCEMENTS OF THE SNL PA METHODOLOGY

SNL has implemented the PA process on several other waste management projects, including development of a PA methodology for long-term cover systems for uranium mill-tailing landfills (Ho, Arnold, et al. 2002); a PA for the mixed waste landfill on the Kirtland Air Force Base in New Mexico (Ho, Goering, et al. 2007); and preliminary PA investigations to address radioactive waste management issues in Egypt, Iraq, and Taiwan. In addition, as the Yucca Mountain repository program is being terminated, SNL has recently conducted three feasibility and scoping PAs for alternative SNF and HLW disposal approaches: disposal in deep boreholes (Brady, et al. 2009); disposal in a clay/shale repository (Hansen, Hardin, et al. 2010); and disposal in a granite repository.[10] Finally, SNL has outlined an approach to enhancing and adapting the PA methodology for application in safety analysis of carbon sequestration and storage, with a PA to run both forward and inverse calculations to support optimization of CO_2 injection and real-time site monitoring as an integral part of the system design and operation (Wang, Dewers, et al. 2010). This study demonstrated a prototype enhanced PA, and it described an optimization approach that continually fuses monitoring data into the PA model through model inversion and parameter estimation, with model calculations, in turn, turn guiding the design of optimal monitoring and carbon injection. This section will provide an overview of all of these projects.

8.1 PA Methodology for Long-Term Cover Systems and PA for Monticello, Utah, Mill Tailings Repository (2000–2002)

SNL applied the PA methodology to a repository site in Monticello, Utah (Ho, Arnold, et al. 2002), where a long-term cover system is being used to isolate long-lived uranium mill tailings from the biosphere. The location of the Monticello site is illustrated in Figure 59, and Figure 60 shows an aerial view of the double composite liner system at the base of the repository. Figure 61 provides a schematic illustration of the landfill cover and geology. Quantitative performance measures for this study included water percolation reaching the uranium mill tailings, radon flux at the surface, groundwater concentrations of ^{226}Ra, and dose. These performance measures were based on a number of requirements, primarily groundwater protection standards of 40 CFR Part 192 but also 40 CFR 264.301 and Utah Administrative Code R315-8-14.2, which set performance standards for the clay liner, and DOE Order 5400.5(II)(1.)(a), which sets an effective dose limit to a member of the public.

[10] A fourth study, investigating disposal of SNF and HLW in a repository in salt (Hansen and Leigh 2011), was also prepared. However, it examined the science available from WIPP and international salt repository programs to reach quantitative conclusions at a subsystem level and qualitative and comparative evaluations of general repository performance. Its conclusions with regard to the feasibility and merits of a geologic repository in salt are important, but no system PA was performed, so this report is not summarized here.

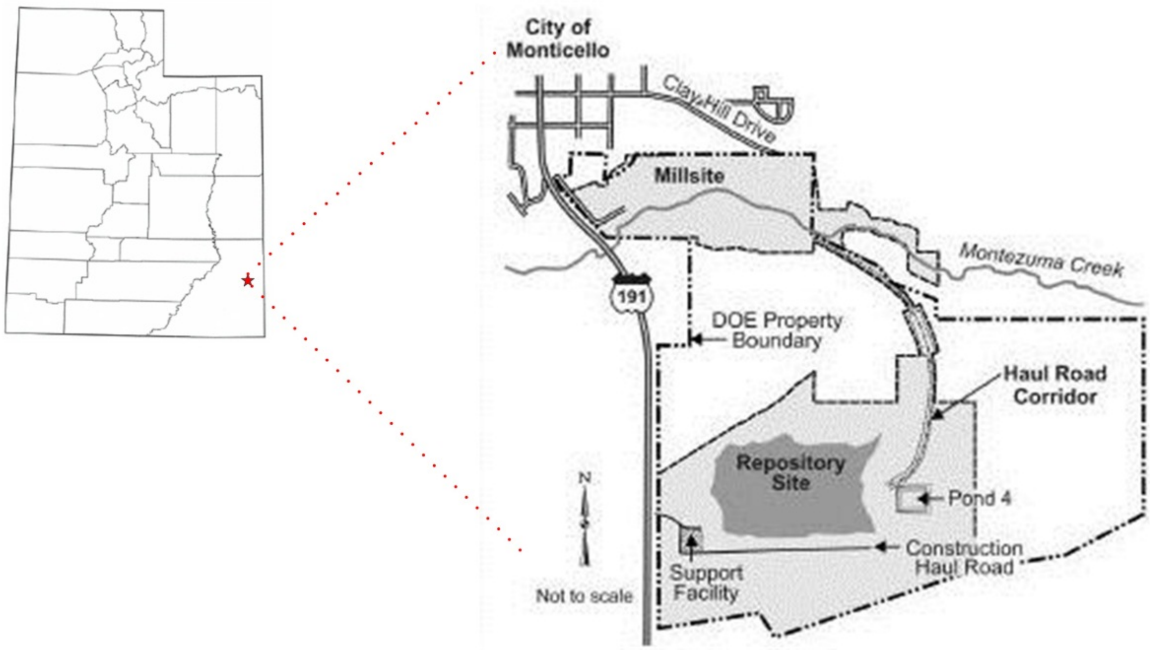


Figure 59. Location of the Monticello mill tailings repository site

Figure 60. Aerial view of the double composite liner system at the base of the Monticello mill tailings repository

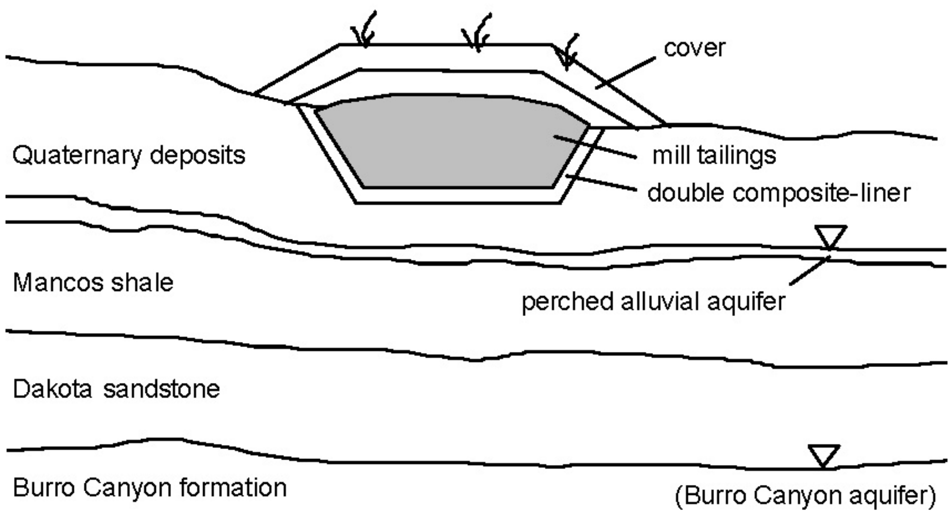


Figure 61. Schematic illustration of the landfill cover and geology

Eight scenarios were considered, and representative FEPs were identified for each scenario, but not all FEPs (e.g., human intrusion and disruptive events) were considered in this assessment (Ho, Arnold, et al. 2002). In total, 29 FEPs were considered and 18 were included in the PA (though it was recognized that others would likely be included in a more comprehensive PA). The scoping nature of this assessment did not allow for a full implementation of the rigorous FEPs identification and screening methods developed by Cranwell, Guzowski et al. (1990). FEPs were identified based on best professional judgment, and the eight scenarios were constructed based on relevant FEPs and to directly assess the relevant performance objectives. The eight scenarios represented four pathways for relative isolation, release, or exposure, as follows:

1. Infiltration percolates through the cover and reaches the mill tailings (scenarios 1 and 2).
2. ^{222}Rn gas diffuses from the mill tailings to the surface (scenarios 3 and 4).
3. ^{226}Ra leaches from the mill tailings and transports through the composite liner, the vadose zone, and into the shallow alluvial aquifer where water is used for agricultural purposes (scenarios 5 and 6).
4. ^{226}Ra leaches from the mill tailings and transports through the composite liner, the vadose zone, and into the deep Burro Canyon aquifer where water is used for agricultural purposes and drinking (scenarios 7 and 8).

Each of these pathways were repeated in two paired scenarios, the first simulating the present climate and repository conditions and the second representing a future conditions (i.e., with a future climate state and degraded performance of the repository’s composite liner).

SNL developed computer models to simulate relevant FEPs including water flux through the cover, source-term release, vadose-zone transport, saturated zone transport, gas transport, and exposure pathways. The component models were then integrated into a total-system PA model using FRAMES, software developed by Pacific Northwest National Laboratory with funding from DOE and EPA. PDFs of important input parameters were constructed and sampled in a stochastic Monte Carlo analysis. About 79 input parameters were used in the models, 31 of which were sampled. For most realizations were simulated using the integrated model to evaluate cover performance for both present and long-term future conditions. For each scenario (except for scenarios 1 and 2), 100 realizations were simulated using FRAMES. In scenarios 1 and 2, only 50 realizations were simulated because the HELP code was not integrated with FRAMES and had to be run manually for each realization (however, after the PA was completed, a method of automating stochastic HELP modeling runs as an integrated part of FRAMES was developed).

Transport of ^{226}Ra from the mill tailings to two different aquifers was evaluated under present and future conditions. Transport to both a shallow alluvial aquifer and a deeper regional aquifer was simulated independently. Groundwater concentrations and dose to a hypothetical receptor were used as the performance metrics in these simulations. In all simulations, the simulated groundwater concentration and dose were below the performance objectives (5 pCi/L and 100 mrem/yr). Peak cumulative dose from ^{226}Ra and its decay products was 0.78 mrem/yr from groundwater pathways in the alluvial aquifer. For groundwater pathways in the Burro Canyon aquifer, models simulating present conditions resulted in doses that were zero for all realizations; simulations of future conditions resulted in a peak cumulative dose of 8.62×10^{-10} mrem/yr (Ho, Arnold, et al. 2002).

Important parameters for these simulations included the low percolation fluxes (from the model of water percolation through the cover) and the relatively large sorption coefficients ^{226}Ra in both the vadose and saturated zones. A stepwise linear regression revealed that the most important parameters for water percolation through the cover were geomembrane placement quality, hydraulic conductivity of the topsoil layer, and wilting point of the clay layer.

An alternative evapotranspiration cover design was also evaluated and compared to the performance objectives. Although the alternative cover did not perform as well as the existing Monticello cover, the results indicated that the simulated percolation fluxes through the alternative cover had a very low risk of exceeding the performance objective. This application of the PA method demonstrates how alternative designs can be compared to minimize cost while ensuring adherence to relevant regulatory requirements and performance metrics. In addition, the results showed how important parameters could be identified with sensitivity analyses for use in prioritizing additional data collection and long-term monitoring studies.

8.2 SNL Mixed Waste Landfill PA (2005–2007)

The SNL Mixed Waste Landfill is a fenced, 2.6-acre landfill in the north-central portion of Technical Area 3 of Kirtland Air Force Base, located as shown in Figure 62 and Figure 63. It was used from March 1959 through December 1988 as the disposal area for LLW and mixed waste from SNL research facilities, receiving approximately 100,000 cubic ft of LLW containing approximately 6,300 curies of activity. The SNL Environmental Restoration Program investigated the Mixed Waste Landfill as part of its Resource Conservation and Recovery Act Facility Investigation since the late 1980s and early 1990s (Ho, Goering, et al. 2007).

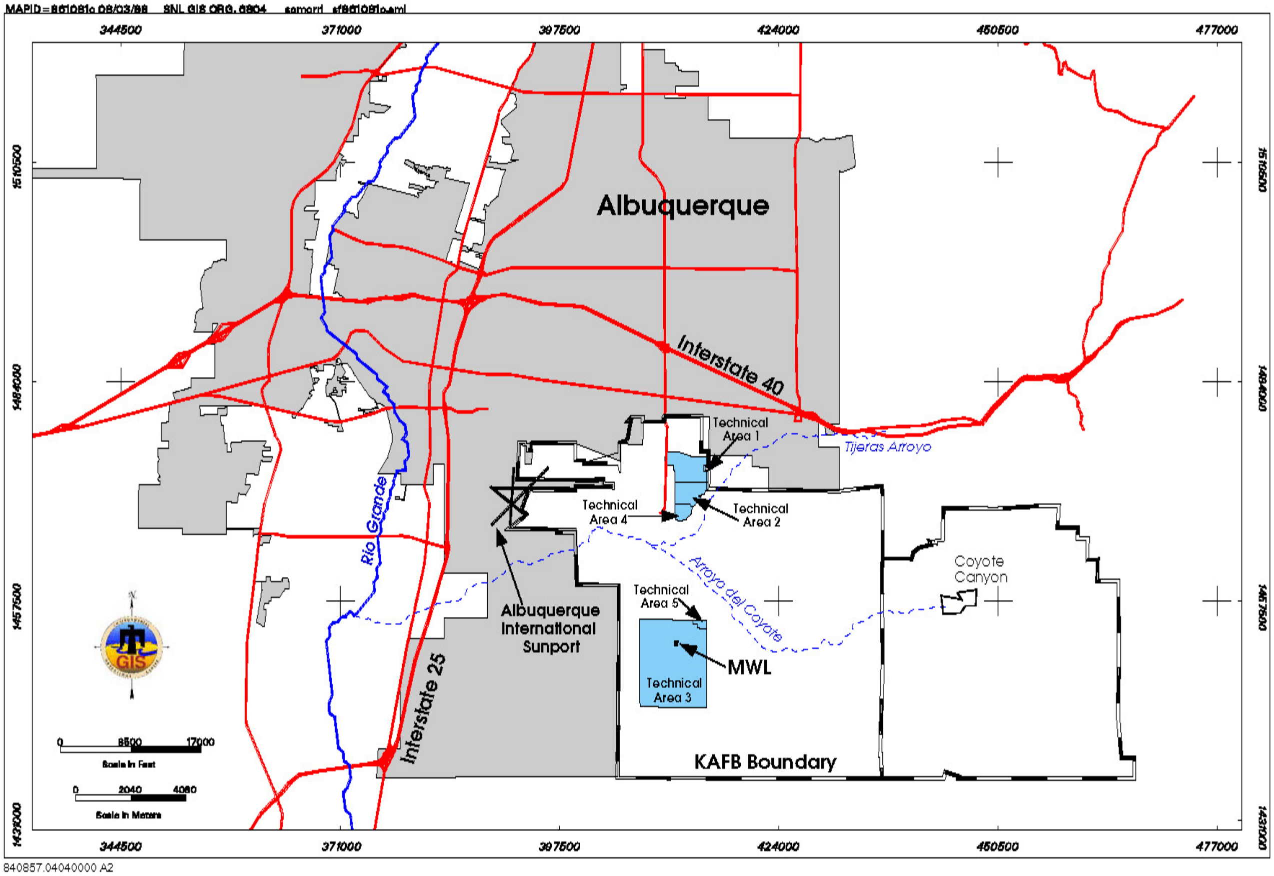


Figure 62. Location of the SNL Mixed Waste Landfill (MWL) in relation to Albuquerque, New Mexico

The Mixed Waste Landfill has been modeled since the 1990s; however, these earlier analyses were neither probabilistic nor were they comprehensive. In 2006, a probabilistic PA (Ho, Goering, et al. 2007) was performed to evaluate the fate and transport of radionuclides and volatile organic compounds from the Mixed Waste Landfill. The current analysis differed from previous analyses in several ways: (1) probabilistic analyses were performed to quantify uncertainties inherent in the system and models; (2) a comprehensive analysis of the performance of the MWL was evaluated and compared against relevant regulatory metrics; (3) sensitivity analyses were performed to identify parameters and processes that were most important to the simulated performance metrics; and (4) long-term monitoring requirements and triggers were recommended based on the results of the quantified uncertainty and sensitivity analyses.

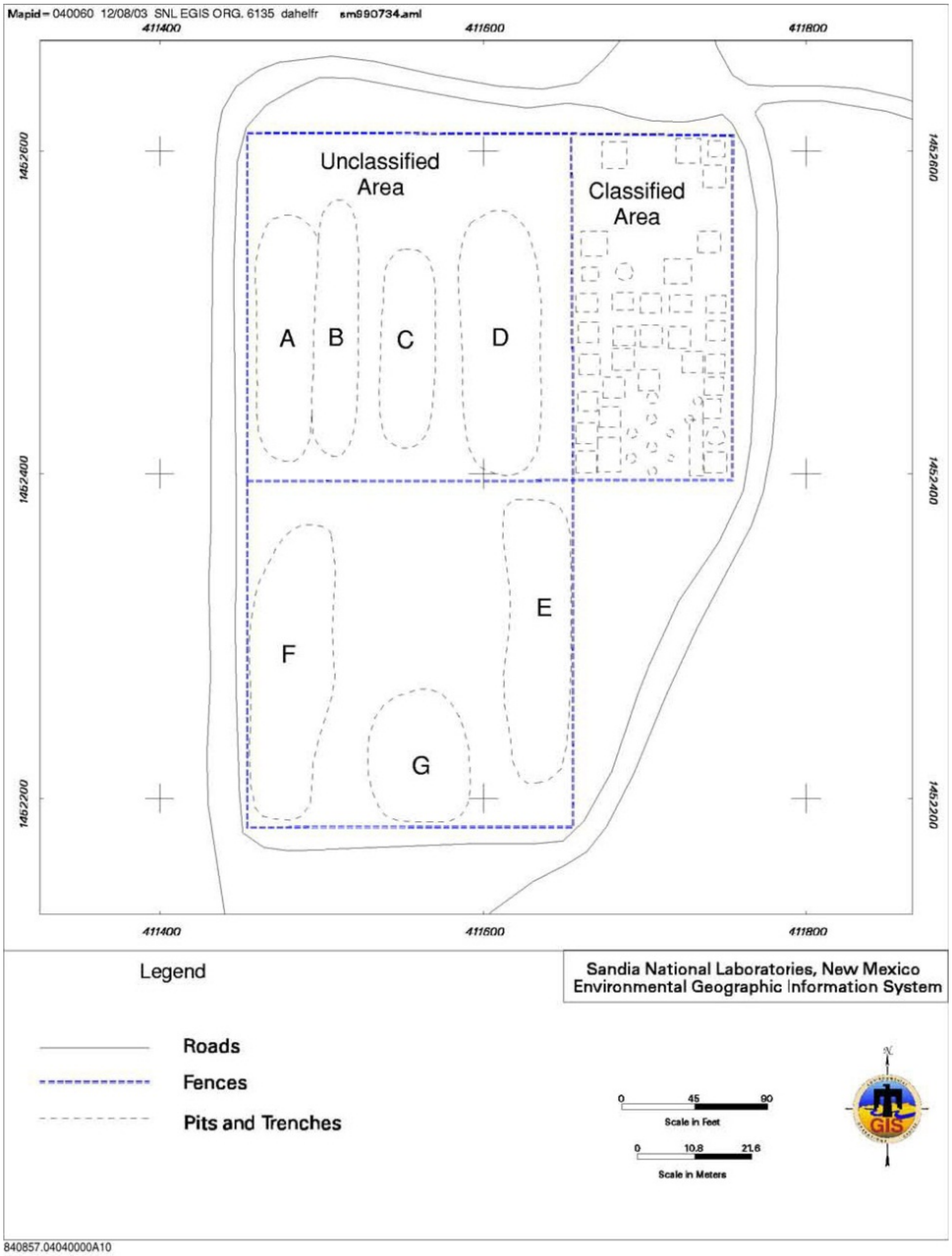


Figure 63. Mixed Waste Landfill layout

Rather than using the rigorous FEPs identification and screening methods developed by Cranwell, Guzowski et al. (1990), the Mixed Waste Landfill PA developed scenarios using a method much like that applied in the Monticello Mill Tailings PA, building scenarios directly responsive to regulatory performance objectives and limits. Relevant contaminants of concern were grouped into the following categories: (1) radionuclides, (2) heavy metals, and (3) VOCs. Table 6 summarizes the specific contaminants, scenarios, and performance objectives that were considered in this study. In general, the two pathways of concern include transport of volatile or gas-phase contaminants from the Mixed Waste Landfill to the atmosphere, and migration of aqueous-phase or vapor-phase contaminants through the vadose zone to the groundwater. For each of these primary pathways, relevant performance objectives and metrics were identified for each of the contaminants of concern. The chosen scenarios represent the most likely releases of contaminants from the Mixed Waste Landfill based on estimated inventories, contaminant properties, and previous studies.

Table 6. Summary of scenarios and performance objectives used in the PA of the Mixed Waste Landfill

Scenario/Description	Performance Objectives[a]
1. Water percolates through the cover to the waste	• Infiltration through the cover shall be less than 10^{-7} cm/s (a unit-gradient flow is assumed to equate infiltration to hydraulic conductivity) (40 CFR 264.301)
2. Tritium diffuses to the atmosphere and migrates via gas and aqueous phases through the vadose zone to the groundwater	• Dose to the public via the air pathway shall be less than 10 mrem/yr (excludes radon) (40 CFR 61.92) • Dose from beta particles and photon emitters shall be less than 4 mrem/yr (40 CFR 141.66) • Tritium concentrations in groundwater shall not exceed 20,000 pCi/L (40 CFR 141.66, Table A; tied to 4 mrem/yr)
3. Radon steadily diffuses to the atmosphere and migrates via gas and aqueous phases through the vadose zone to the groundwater	• The average flux of ^{222}Ra gas shall be less than 20 pCi/m^2 per second at the surface of the landfill (40 CFR Part 192) • Radon concentrations in groundwater shall not exceed 300 pCi/L (proposed EPA rules, 64 FR 59345–59378)
4. One or more radionuclides migrate via the aqueous phase through the vadose zone to the groundwater	• Maximum concentrations in groundwater of gross alpha particle activity (including ^{226}Ra but excluding radon and uranium) is 15 pCi/L (40 CFR 141.66) • Uranium concentrations in groundwater shall not exceed EPA MCL of 30 mg/L (40 CFR 141.66) • Dose from beta particles and photon emitters shall be less than 4 mrem/yr (40 CFR 141.66)
5. Lead and cadmium migrate via the aqueous phase through the vadose zone to the groundwater	• Lead concentrations in groundwater shall not exceed the EPA action level of 15 mg/L • Cadmium concentrations in groundwater shall not exceed the EPA MCL of 5 mg/L
6. PCE migrates through the vadose zone to the groundwater	• PCE concentrations in groundwater shall not exceed the EPA MCL of 5 mg/L (40 CFR 141.61)

Conservative values and assumptions were used to define values and distributions of uncertain input parameters when site data were not available. The PA used 113 input parameters; 75 were sampled distributions. For each of the six scenarios defined in the PA, 100 realizations of a 1,000-year period were simulated, and a sensitivity analysis was performed to compare use of 100 realizations against use of 200 realizations, and results showed that 100 realizations were sufficient to adequately represent the distribution of the simulated output.

In a small percentage of realizations or under the most conservative assumptions, some simulations showed the possibility of radon gas concentrations that would exceed regulatory limits. Based on the results, a focused set of monitoring triggers were proposed for the air, surface soil, vadose zone, and groundwater at the Mixed Waste Landfill; if a trigger were exceeded, then monitoring data would be collected to assess trends and recommend corrective actions, if necessary. The triggers established include numerical thresholds for (1) radon concentrations in the air, (2) tritium, gamma-emitting radionuclides, and heavy metal concentrations in surface soil, (3) volatile organic compound concentrations and moisture content in the vadose zone, and (4) uranium and volatile organic compound concentrations in groundwater.

By utilizing these triggers during long-term monitoring at the Mixed Waste Landfill, SNL and DOE will ensure that the Mixed Waste Landfill continues to protect human health and the environment and meet its commitments to the New Mexico Environmental Department.

8.3 Deep Borehole Disposal Preliminary Performance Assessment (2009)

In 2009, SNL conducted a study of deep borehole disposal of radioactive waste, considering several factors, including technical, regulatory, safety, cost, and performance factors (Brady, et al. 2009). A preliminary PA was included as part of the study. In the deep borehole disposal approach analyzed in this study, as illustrated in Figure 64, radioactive waste would be emplaced in solid form (spent fuel or glass) at the bottom of 3-km to 5-km deep boreholes in crystalline basement rocks—typically granites (found relatively commonly at a depth of 2 to 5 km)—using off-the-shelf oilfield technology, including simple disposal containers made of standard oilfield casing. The physical transport of radionuclides away from HLW and SNF those depths would be limited by low water content, low porosity and low permeability of crystalline basement rock, high overburden pressures that contribute to the sealing of transport pathways; and the presence of convectively stable saline fluids.

In the U.S., the 70,000 MTHM of waste currently proposed for Yucca Mountain could be accommodated in about 600 deep boreholes (assuming each deep borehole had a 2 km long waste disposal zone that contained approximately 400 vertically stacked fuel assemblies). The remainder of the projected inventory of 109,300 MTHM could be fit into about 350 additional boreholes. Because crystalline basement rocks are relatively common at 2- to 5-km depth, the U.S. waste disposal burden might be shared by shipping waste to regional borehole disposal facilities, which, if located near existing waste inventories and production, would minimize shipping. Given a typical disposal length of approximately 2 km and holes spaced 0.2 km apart suggests the total projected US inventory could be disposed in several borehole fields totaling about 30 km^2.

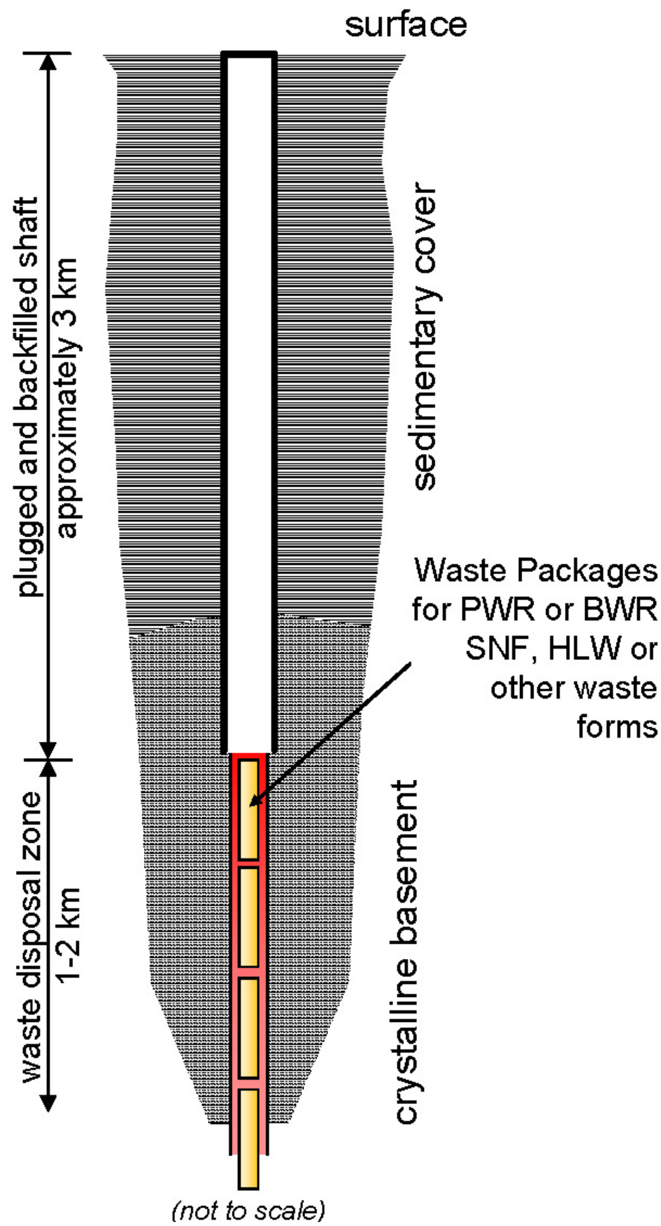


Figure 64. Schematic illustration of the Deep Borehole Disposal concept

In order to evaluate the system performance of a deep borehole disposal concept, it is necessary to adopt or develop a regulatory standard by which the performance can be measured. For the purposes of analysis, the NWPA was assumed to be amended to allow consideration of alternative disposal concepts, and new regulations were assumed to be similar in key regards to the current Yucca Mountain regulations. Thus, the primary overall performance measure of interest is mean annual dose to a hypothetical individual, with limits set at 15 mrem/yr for the first 10,000 years following disposal and 100 mrem/yr for the period between 10,000 years and 1 million years. Other details of the regulatory framework, including FEPs screening criteria, are also assumed from the Yucca Mountain-specific performance standards in 40 CFR Part 197 and 10 CFR Part 63, (with the exception of human intrusion scenarios, for which new regulatory requirements would need to be developed, and which were not included in this PA).

The Deep Borehole Disposal PA utilized a FEPs screening approach similar to that taken for both Yucca Mountain and WIPP to identify the significant FEPs that should be included in the quantitative PA. The FEPs list from the Yucca Mountain license application was adopted as a reasonable starting point for evaluation. Each of the 374 FEPs on the Yucca Mountain list was considered for potential relevance to deep borehole disposal; FEPs that may be unique to deep borehole disposal were considered and compared to the list to identify existing FEPs that capture the processes of interest and concern for boreholes. Notably, no new FEPs were identified in this process, which suggests that although the Yucca Mountain list was specifically tailored for a mined repository, it remains a useful starting point for preliminary analyses or other methods of geologic disposal. In addition, the preliminary FEPs screening effort included a qualitative estimate of the level of effort likely to be required to provide a robust basis for addressing each FEP in a full PA, in terms of the likely difficulty in researching and documenting the technical basis for FEPs screened out or, for FEPs screened in, the likely degree of difficulty in modeling them for PA.

The preliminary FEPs screening included about 110 FEPs into the preliminary PA. Consideration of the included FEPs shows that radionuclides emplaced in deep boreholes might reach the biosphere through three pathways: (1) up the borehole (including accidental release during emplacement); (2) along the annulus of disturbed rock; and/or (3) radially out through groundwater in the surrounding rock. Those pathways were developed as the three scenarios chosen for analysis in the preliminary deep borehole disposal PA:

- **Scenario 1: Transport in the borehole**. Hydrologic flow up the borehole transports radionuclides to a shallow aquifer from which they are pumped to the biosphere
- **Scenario 2: Transport in disturbed rock around the borehole**. Hydrologic flow up the annulus of disturbed rock surrounding the borehole transports radionuclides to a shallow aquifer from which they are pumped to the biosphere
- **Scenario 3: Transport in surrounding rock away from the borehole.** Hydrologic flow up through the crystalline basement and sedimentary cover transports radionuclides to a shallow aquifer from which they are pumped to the biosphere

The PA was developed for a single borehole 5 km deep, with 400 pressurized water reactor (PWR) assemblies (approximately 150 MTHM) vertically stacked down the length of the waste disposal zone (the bottom 2 km of the borehole). For the purposes of characterizing the waste, the relative radionuclide inventories for commercial SNF used in the YMP PA were considered representative of the entire US HLW and SNF inventory. Dose calculations for a hypothetical person living near the withdrawal well were based on biosphere dose conversion factors consistent with the lifestyle of the Yucca Mountain reasonably maximally exposed individual (RMEI), as specified by the EPA in 40 CFR Part 197.

Temperatures within the borehole and the host rock were simulated using a horizontal, two-dimensional model of thermal conduction implemented with the FEHM software code (Zyvoloski, et al. 1997), which was developed by Los Alamos National Laboratory for the Yucca Mountain Project. For SNF, the model used the heat output curves for a single average pressurized water reactor fuel assembly that has been aged for 25 years, as used for Yucca Mountain PA modeling. A separate calculation was also performed for vitrified HLW, with heat output curves from the current vitrified waste produced by reprocessing of commercial spent nuclear fuel in France (ANDRA 2005). Thermally driven fluid flow rates were estimated using a vertical, radial, two-dimensional model of coupled heat and fluid flow also implemented with the FEHM software code, as was the numerical model for groundwater pumping and radionuclide transport. Radionuclide solubilities were calculated using the PHREEQC code.

The conceptual model was implemented numerically in a Microsoft Excel spreadsheet. Based on the scenario analysis described above, the preliminary deep borehole PA was performed for a simplified and conservative representation of combined radionuclide releases from Scenarios 1 and 2.

The PA results suggested that radionuclides in spent fuel emplaced in deep boreholes will experience little physical reason to leave the borehole/near borehole domain. The vast majority of radionuclides, and the fuel itself, would be thermodynamically stable and would therefore resist dissolution into borehole fluids, or movement into and through the adjacent rocks. Thermal-hydrologic calculations indicated that, after an early window extending from the time of emplacement to about 150 years after emplacement (in the borehole), and about 600 years (to the top of the basement), there would be no vertical fluid flow to transport radionuclides towards the surface. Vertical transport velocities in the early flow window would be between 0.1 m/yr through the basement rock and 0.7 m/yr in the borehole. This means that total vertical fluid movement in and adjacent to deep borehole disposal zones should not exceed roughly 100 m. In the absence of advection, chemical diffusion cannot move radionuclides from boreholes through discontinuous, stagnant, and density-stratified waters over distances much greater than about 200 m in the 1 million years needed for the vast bulk of the radioactivity to decay away.

Simplified and conservative PA calculations indicate that radiological dose to a human receptor via the groundwater pathway would be limited to a single radionuclide (^{129}I) and would be negligibly small, with a peak dose of 1.42×10^{-10} mrem/yr occurring after 8,200 years, approximately 10 orders of magnitude below current regulatory limits for the Yucca Mountain repository system.

The deep borehole disposal PA results were based on a simple design concept and several bounding and conservative assumptions. For example, all waste was assumed to instantly degrade and dissolve inside the waste canisters; all waste was assumed to be PWR assemblies; no credit was taken for sorption or decay along the saturated zone transport pathway from the sealed borehole to the withdrawal well. Thus, this PA serves as a first approximation, pointing the way for further, more detailed and realistic assessments. More refined PAs may indicate peak doses that are lower or later, or both, than indicated by these preliminary results. The PA also helps identify additional design considerations for further research. For example, recognizing that ^{129}I is likely the only radionuclide of concern, releases might be lowered further from the already low predicted values above by development and deployment of sorbents, such as layered bismuth hydroxide, that sorb or sequester ^{129}I in the borehole or in the seals.

8.4 Shale Disposal Feasibility Preliminary Performance Assessment (2010)

In 2010, SNL conducted a study of the feasibility of high-level radioactive waste disposal in shale within the United States (Hansen, Hardin, et al. 2010).

In the late 1970s and early 80s, as numerous potential sites for radioactive waste disposal were being investigated, the western desert site now known as the Nevada National Security Site was considered attractive for a repository. Though the national repository program focused primarily on characterizing three different media (salt, basalt, and tuff), investigations at Nevada National Security Site included granite and argillite in addition to the volcanic tuff at Yucca Mountain. Limited characterization and modeling of argillite was completed in the SNL field testing of the Eleana argillite, also on the Nevada National Security Site (Eaton, et al. 1980, Lappin, Thomas and McVey 1981) and the Conasauga shale near Oak Ridge, Tennessee (Krumhansl 1979, 1983). SNL conducted the small-scale field tests of the Eleana argillite and Conasauga shale under the leadership of Oak Ridge National Laboratory, which, at the time, led the U.S. research and development efforts for repository investigations of shale and clay media.

The 1987 amendments to the Nuclear Waste Policy Act focused all further site characterization solely on Yucca Mountain. Because U.S. efforts have focused on the volcanic tuff site at Yucca Mountain, radioactive waste disposal in U.S. shale formations has not been considered for many years. However, advances in multiphysics computational modeling and research into clay mineralogy continue to improve the scientific basis for assessing nuclear waste repository performance in such formations. Importantly, several countries—key among them being Belgium, France, and Switzerland—have actively studied nuclear waste disposal in clay, shale, and argillite media for decades.

Hansen et al. (2010) selected representative material properties for their PA of a generic repository in shale drawing heavily from the data from these international programs, which have characterized several clay formations around the world. They focused on (1) Opalinus claystone at Mont Terri, Switzerland, (2) Callovo-Oxfordian argillite/mudstone near Bure, in eastern France, and (3) Boom clay at Mol, Belgium. More field work would be needed to characterize any particular clay/shale site in the United States, but initially international collaboration is be a key source of information as the U.S. contemplates alternative disposal approaches.

8.4.1 Shale Repository Design Concept

The generic repository design concept developed by Hansen et al. (2010) borrows strongly from the experience and expertise of these existing shale repository programs. Figure 65 illustrates sketches of the intended disposal process for the three comparable European repository studies, and a representation of possible disposal in a clay/shale formation in the U.S., which uses horizontal placement in an unlined and unbackfilled borehole, 0.7 m in diameter and 40 m in length. The disposal borehole is sealed at the proximal end with concrete and bentonite as depicted in the lower right panel of Figure 65, labeled “USA.” The generic repository features and dimensions are generally consistent with the French layout shown in the upper left panel of Figure 65, but could be changed in response to site specific analyses.

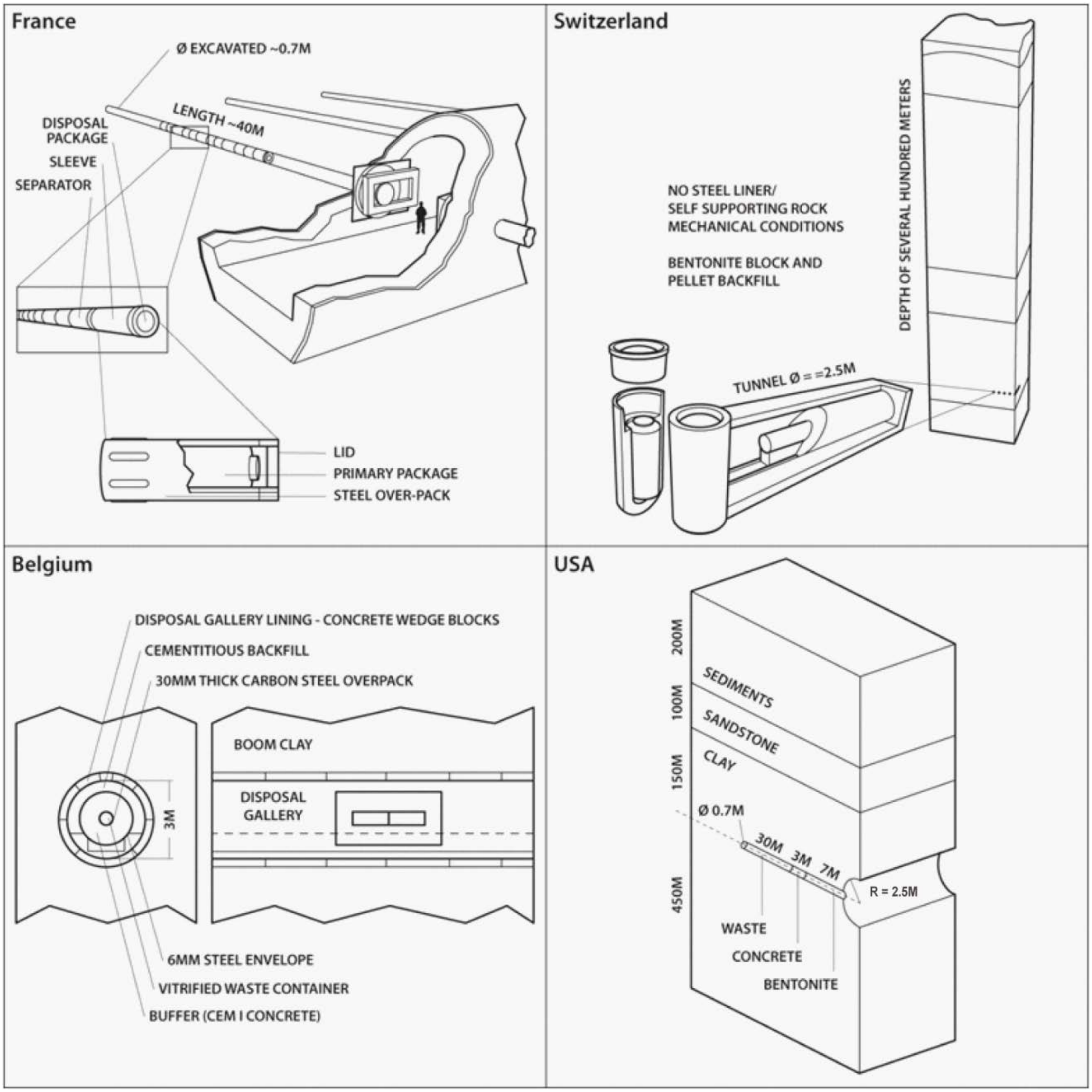


Figure 65. Schematic illustration of clay/shale disposal concepts from Belgium, France, and Switzerland, and generic USA conceptual design for preliminary analysis

The disposal boreholes would be spaced far enough apart to limit interaction of neighboring boreholes. Following an approach used extensively in the thermal management strategy for the Yucca Mountain repository license application (DOE 2008, SAR Section 1.3.1.2.5) and using thermal decay curves for spent nuclear fuel and HLW prepared for the Yucca Mountain, a thermal analysis indicated that, for older canisters of HLW, the emplacement borehole spacing could be as small as a few meters. Younger HLW canisters with greater heat output would not change this result significantly. For spent nuclear fuel, the reference borehole spacing of 20 m was shown to be sufficient to limit both the midpoint and borehole wall temperatures (e.g., less than 100°C).

The access drifts in the conceptual design are 5 m in diameter, which can accommodate waste packages containing spent nuclear fuel assemblies or canisters of vitrified HLW, but could be expanded to allow additional room for construction and waste handling equipment.

8.4.2 Shale Disposal Performance Assessment

As with the deep borehole PA, the regulatory framework was assumed to be similar in key regards to the current Yucca Mountain regulations. Thus, the primary overall performance measure of interest is mean annual dose to a hypothetical individual, with limits set at 15 mrem/yr for the first 10,000 years after repository closure and 100 mrem/yr for the period between 10,000 years and 1 million years. Other details of the regulatory framework, including FEPs screening criteria, are also assumed from the Yucca Mountain-specific performance standards in 40 CFR Part 197 and 10 CFR Part 63. Note that, while the depth of deep borehole disposal would likely change the regulatory requirement for human intrusion scenario, Hansen et al. (2010) assumed that it would remain applicable for a clay/shale repository and that implementation for a clay/shale repository would be inherently similar to the human intrusion scenario in the performance assessment for WIPP. However, human intrusion was not analyzed in the preliminary PA.

Like the Deep Borehole Disposal PA (Brady, et al. 2009), the Shale Disposal PA (Hansen, Hardin, et al. 2010) utilized a FEPs screening approach similar to that taken for both Yucca Mountain and WIPP to identify the significant FEPs that should be included in the quantitative performance assessment. For this analysis, a preliminary FEP list developed by the Used Fuel Disposition Program for Nuclear Energy Office of DOE was used. The Used Fuel Disposition Program FEP list was generalized from the FEP list for the Yucca Mountain license application (SNL 2008c, DOE 2008), as described in Section 5. Each of the 216 FEPs on the Used Fuel Disposition Program FEPs list was considered for potential relevance to disposal in clay/shale formations; the international Nuclear Energy Agency FEP catalogue specifically for argillaceous media (NEA 2003) was also reviewed to identify FEPs that may be unique to disposal in clay/shale. In addition, as with the Deep Borehole Disposal PA, the preliminary FEPs screening effort included a qualitative estimate of the level of effort likely to be required to provide a robust basis for addressing each FEP in a full PA, in terms of the likely difficulty in researching and documenting the technical basis for FEPs screened out or, for FEPs screened in, the likely degree of difficulty in modeling them for PA.

The preliminary FEPs screening included about 129 FEPs into the preliminary PA. Based on consideration of those included FEPs and the assumptions for the generic shale repository, radionuclides emplaced in deep boreholes might reach a hypothetical aquifer and the biosphere through two principal paths for a nominal scenario:

1. Advective transport through the short-lived excavation-disturbed zone and through or around shaft seals (hydrologic flow through the repository and up the shafts transports radionuclides to a shallow aquifer from which they are pumped to the biosphere); and

2. Diffusive transport in host clay/shale (diffusion, coupled with a small hydraulic gradient, transports radionuclides upward from the repository, through the clay/shale host rock, and to a shallow aquifer from which they are pumped to the biosphere).

Disruption of the repository through human intrusion was recognized as a third scenario to be considered. A stylized calculation specified by 40 CFR Part 197 would represent a borehole for hydrocarbon exploration drilled through the repository after repository closure and later abandoned, with a vertical hydrologic gradient that transports radionuclides to a shallow aquifer where they are pumped to the biosphere.

The first scenario was not considered for this preliminary analysis because of its short-term nature, the likely effectiveness of engineered seals, and the lack of a strong hydraulic pressure gradient to drive water through the repository and up the shafts. The third scenario was also not of interest for this work because it is stylized and only consequences are evaluated. Only the second scenario was considered in the generic performance analysis, using a one-dimensional advective-dispersive model formulation. A more complete or site-specific screening of the FEPs may identify additional scenarios of interest and may also show that some aspects of the chosen scenarios do not need further analysis.

The PA was developed for an assumed repository layout (the "USA" design shown in Figure 65), consisting of 0.7-m-diameter emplacement boreholes drilled horizontally from a 5-m-diameter horizontal access tunnel. The emplacement boreholes (i.e., the waste disposal zone) are at a depth of 450 m below the land surface. The overlying geologic units, from the repository to the ground surface, consist of clay/shale (150 m), a sandstone aquifer (100 m), and sediments (the top 200 m). The PA conceptual model provides that each waste package contains a single PWR assembly (equivalent to approximately 0.4 MTHM) in a 5-m-long package, emplaced horizontally in an emplacement borehole. As many as six such packages would be emplaced in each 40-m-long emplacement borehole. The repository would include 200,000 waste packages distributed in a horizontal array on a single emplacement level. Thermal loading was chosen to produce waste package temperatures approaching 100°C, and also to provide bounding calculations. Based on this design concept, a clay/shale repository could accept all waste from the current inventory for emplacement, with the use of up to 50 years of decay storage for the hottest spent nuclear fuel.

For the purposes of characterizing the waste for PA, the relative radionuclide inventories for commercial SNF used in the YMP performance assessment were considered representative of the entire U.S. HLW and SNF inventory. Dose calculations for a hypothetical person living near the withdrawal well were based on biosphere dose conversion factors consistent with the lifestyle of the Yucca Mountain reasonably maximally exposed individual (RMEI), as specified by the EPA in 40 CFR Part 197.

Dose is calculated by solving for concentration profiles at many time steps, and integrating the profiles to determine the total radionuclide mass (dissolved and sorbed) in the clay/shale layer, and above the clay/shale layer. The region beyond the clay/shale layer is represented by extending the solution to 10 km. In the conceptual model, the integrated radionuclide mass beyond 150 m is taken up immediately in water pumped from the aquifer, which is a "swept away" boundary condition that does not affect the diffusive flux within the clay/shale layer. This is based on an assumption that the concentration gradient changes slowly across the shale-sandstone boundary. The corresponding dose to the RMEI is based on the mass fluxes into the sandstone aquifer.

8.4.3 Implementation of Models

The conceptual model was implemented numerically in a Microsoft Excel spreadsheet. The numerical solution was used to calculate source concentrations and one-dimensional radionuclide transport for 29 selected radionuclides. The dose calculation was limited to the 12 radionuclides that could potentially transport far enough in 1 million years to contribute to dose (^{239}Pu, ^{242}Pu, ^{237}Np, ^{233}U, ^{234}U, ^{236}U, ^{238}U, ^{14}C, ^{79}Se, ^{99}Tc, ^{129}I, and ^{135}Cs).

While the overall model implementation was comparatively simple, its submodels represented notable advances in process model implementation and computation. The complex coupled thermal-hydrologic-mechanical-chemical calculations for a generic repository in clay/shale provided an opportunity to demonstrate the current capabilities of the SIERRA Mechanics software (Edwards 2002) as applied to a repository problem that requires many of the software's unique capabilities. The geometries, material properties, thermal loading, and other features of these calculations were chosen to represent potential repository designs.

The development of the SIERRA Mechanics code suite has been funded by the DOE Advanced Simulation and Computing program for more than ten years, supporting a variety of applications requiring high-performance multiphysics modeling. Sandia's Laboratory Directed Research and Development program has helped specialize modules of SIERRA Mechanics to geologic applications and coupled thermal-hydrologic-mechanical-chemical modeling. The goal has been development of massively parallel multiphysics capabilities to support the Sandia engineering sciences mission. SIERRA Mechanics was designed and developed to run on the latest and most sophisticated massively parallel computing hardware, with capability to span the hardware range from single workstations to systems with thousands of processors. The foundation of SIERRA Mechanics is the SIERRA toolkit, which provides finite element application-code services such as: (1) mesh and field data management, both parallel and distributed; (2) transfer operators for mapping field variables from one mechanics application to another; (3) a solution controller for code coupling; and (4) included third party libraries (e.g., solver libraries, communications package, etc.).

The SIERRA Mechanics code suite comprises application codes that address specific physics regimes. The two SIERRA Mechanics codes that are used for THMC coupling are Aria (Notz, et al. 2007) and Adagio (SIERRA Solid Mechanics Team 2009). The physics currently supported by Aria include the incompressible Navier-Stokes equations, energy transport equation, and species transport equations, as well as generalized scalar, vector, and tensor transport equations. The multiphase porous flow capability is a recent addition to Aria. Aria also has some basic geochemistry functionality available through embedded chemistry packages. The mechanics portion of the THMC coupling is handled by Adagio, which solves for the quasistatic, large deformation, large strain behavior of nonlinear solids in three dimensions. Adagio has some discriminating technology, developed at Sandia for solving solid mechanics problems, that involves matrix-free iterative solution algorithms for efficient solution of extremely large and highly nonlinear problems. This technology is especially suited for scalable implementation on massively parallel computers. The THMC coupling is done through a solution controller within SIERRA Mechanics called Arpeggio.

The repository geometry, material properties, thermal loading, and other features of this analysis were chosen to represent a plausible repository implementation concept. Further evaluation of tunnel deformation and stability, operational functionality, and the effects of excavation and heating on long-term performance would require development and application of site-specific constitutive models for the clay/shale. Even in this generic assessment, the value of three dimensional multiphysics calculations is demonstrated by identifying sensitive aspects of the underground setting.

8.4.4 Results

Thermal, hydrologic, and geochemical calculations suggest that radionuclides in a clay/shale repository will not migrate far from the disposal horizon. The great majority of radionuclides in the current waste inventory will be thermodynamically stable as solids and will therefore resist migration. Much of the inventory will decay before transport to the biosphere can occur.

Calculated aqueous radionuclide concentration profiles as a function of distance within the clay/shale layer at 1 million years after emplacement are shown in the left plot of Figure 66 (Hansen, Hardin, et al. 2010, Figure 4-2). The distance of 150 m is the boundary between the clay layer and the hypothetical sandstone aquifer. The radionuclide concentrations at that boundary over time are presented in the right plot of Figure 66 (Hansen, Hardin, et al. 2010, Figure 4-1). Of the 29 radionuclides considered in the analysis, the models suggest that only eight would reach the aquifer by 1 million years. In order of concentration (above 10^{-10} mg/L), those radionuclides are ^{129}I, ^{238}U, ^{236}U, ^{79}Se, ^{234}U, ^{233}U, ^{135}Cs, and ^{237}Np.

Only radionuclides with potentially significant concentrations at or near the 150 m boundary at 1 million years are included in the dose calculation (Figure 67) (Hansen, Hardin, et al. 2010, Figure 4-3). The calculated dose to the reasonably maximally exposed individual, based on the radionuclide mass flux into a hypothetical overlying sandstone aquifer, is 0.01 mrem/yr or less at 1 million years, which is far below the regulatory annual dose limit of 100 mrem/yr in the current regulations. This result is for PWR fuel; the dose for HLW would generally be smaller because of the smaller inventory of radionuclides per repository plan area. The PA for shale disposal predicts that the dose at 10,000 years is effectively zero.

These results are based on several simplifying assumptions:

- All waste is assumed to instantly degrade and dissolve inside the waste packages
- All waste is assumed to be PWR assemblies
- Unlimited availability of moisture for waste form degradation and transport
- No sorption on degraded waste package materials
- No credit is taken for horizontal transport to, or sorption or decay within, the sandstone aquifer
- The repository is assumed to be isolated from through-going hydrologic features such as faults or fracture zones that could provide preferential pathways for groundwater or radionuclides.

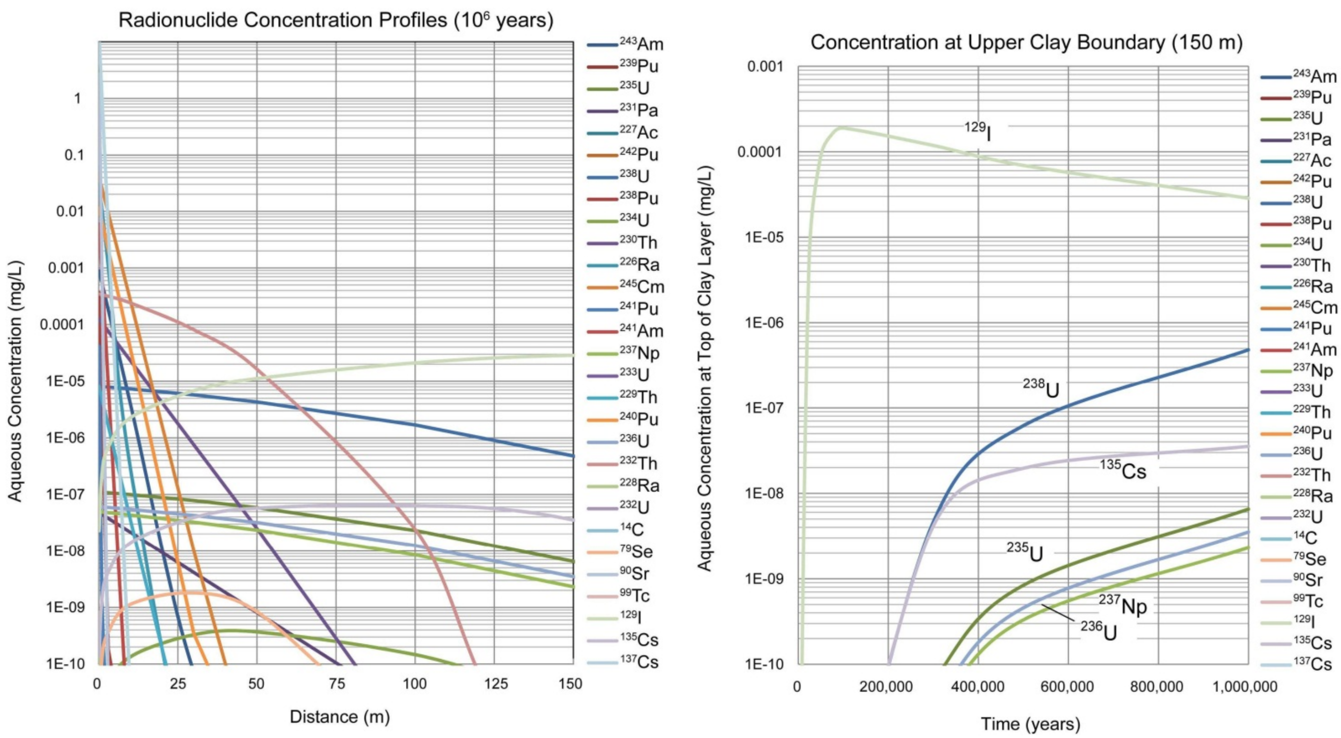


Figure 66. Aqueous radionuclide concentrations over a single waste package (left) within the clay/shale unit as a function of distance at 1 million years after emplacement; and (right) at the top of the clay/shale layer as a function of time

8.5 Granite Disposal Feasibility Preliminary Performance Assessment (2011)

Mariner et al. (2011) evaluated the feasibility of high-level radioactive waste disposal in granite within the United States, where there are many potential locations in granite formations with positive attributes for permanent disposal. Similar geologic formations have been extensively studied by international programs including Finland, Sweden, Canada, Switzerland, and Spain, with largely positive results over significant ranges of the most important material characteristics, including fracture permeability, stability, and geologic terrain. The granite PA study, supported in part by the Laboratory Directed Research and Development program at SNL and with additional support from the DOE Office of Basic Energy Sciences, drew significantly from advanced international work to establish functional and operational requirements for disposal of a range of waste forms in granite. Mariner et al. (2011) developed a preliminary scoping PA, based on the applicable FEPs identified by international investigators, to support generic conclusions regarding postclosure safety of granite repositories.

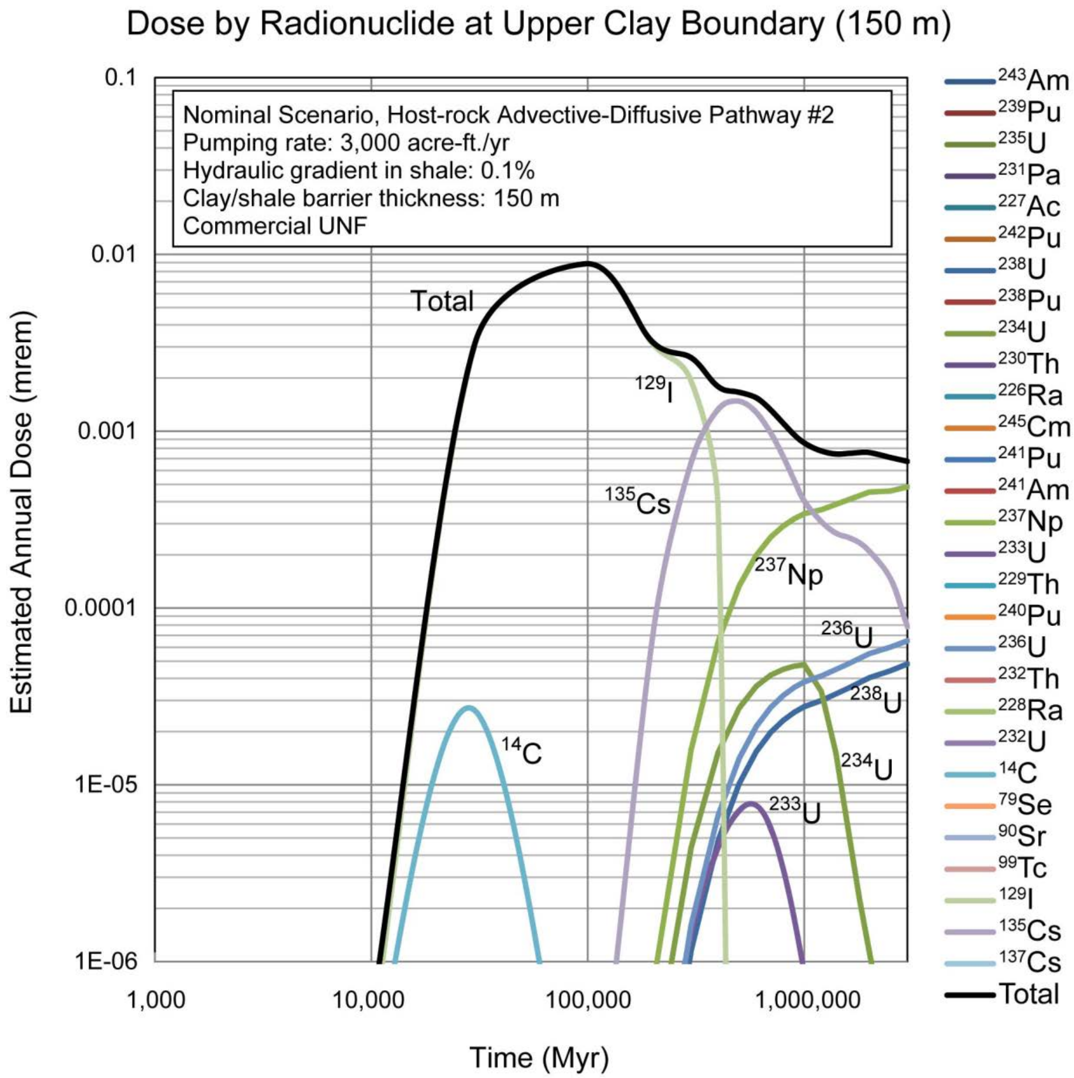


Figure 67. Dose calculation for 200,000 waste packages in clay/shale disposal, by radionuclide (for significant contributors) as a function of time

Disposal of HLW and SNF in suitable granite formations can be shown to be acceptable to meet anticipated regulations, although its long term performance is tied to engineered barriers. Vertically and laterally extensive granite formations exist in multiple locations in the contiguous 48 states. Temperatures near emplaced waste packages can be maintained below boiling and will decay to within a few degrees of the ambient temperature within a few decades (or longer, depending on the waste form). The host rock and engineered barriers provide a repository setting that strongly limits corrosion and degradation of the waste package and release of radionuclides to the geosphere. Under the conditions modeled, a granite repository can achieve containment sufficient to meet current regulatory requirements. The PA for granite disposal was based on the assumption that long-term standards for disposal in granite would be identical in the key aspects to those prescribed for existing and planned repositories.

While early U.S. research in geologic repositories focused on salt, repository research in granite formations increased in the late 1970s and early 1980s. Beginning in 1978, the U.S. developed an underground research laboratory at a depth of 420 m in the Climax monzonite stock, a granitic body at the Nevada National Security Site (then known as the Nevada Test Site). The Spent-Fuel Test–Climax used both commercial SNF and electric simulators to demonstrate the feasibility and safety of spent fuel storage and retrieval from a repository in granitic rock (Patrick 1986). Sites in granite media were considered the initial stages of the NWPA siting process, but were not carried forward into the screening process. Research into granite as a repository host medium continued in preparation for eventual siting of a second repository, until the Nuclear Waste Policy Amendments Act of 1987 directed that DOE characterize only the Yucca Mountain site in volcanic tuff, phasing out funding for all research programs for other repository geologic media and disposal strategies.

Although crystalline rock was no longer considered to be a potential repository host rock, limited work on crystalline concepts continued after 1987 as part of other research. As described in Section 7.2.1, the PA for INL HLW disposal included models for a generic granite repository and for a generic salt repository. For that study, the waste package outer layer was assumed to be Inconel 625 (69% nickel, 22% chromium, 9% molybdenum) rather than copper, the preferred material in the granite repository concepts described by Mariner et al. (2011).

8.5.1 Granite Repository Concept for Scoping PA

Based on disposal concepts in Sweden and Finland, the repository design concept for the preliminary granite PA was conceptualized as a network of tunnels with waste packages with copper outer barriers emplaced vertically in boreholes drilled in the tunnel floors. Copper is chosen as the outer barrier because it is highly resistant to corrosion in the chemically reducing environment of the repository horizon. The diameter and length of the emplacement borehole are larger than the waste package to accommodate a clay buffer that surrounds the waste package in the borehole. After emplacement, the emplacement tunnels are backfilled with a mixture of crushed rock and clay.

A repository would be deep enough below the present land surface to ensure that the waste is not exposed to the biosphere through erosion or shallow groundwater circulation during its hazardous period. By siting the repository at least 300 m beneath the present land surface in granite where fractures are sparse and hydraulic conductivity is low, erosion and shallow groundwater circulation would not threaten repository performance. Canister size and heat generation will strongly influence an actual repository design, including the extent of the underground facility and the minimum vertical thickness of the host formation. The geochemical environment expected in emplacement boreholes will also influence the design, including backfill and seal systems. Although retrievability is facilitated in a granite repository by the long-term stability of granite, for the purpose of the scoping study, retrievability was not a design priority.

8.5.2 FEPs Analysis and Scenario Development

For a full PA, a more comprehensive FEPs screening analysis would need to be conducted, but for scoping analyses of a generic repository in granite, Mariner et al. (2011) considered the FEPs list developed for the DOE Used Fuel Disposition Campaign, which was developed from international FEP lists and currently includes 208 FEPs potentially relevant to a wide range of disposal system alternatives. In addition, they examined the scenarios developed in international programs for repositories in granite. The preliminary screening analysis was done based on current U.S. regulations and based on the assumptions described above. The analysis identified 116 FEPs that would likely have a screening decision for inclusion. The primary pathway for potential release of radionuclides to the biosphere was expected to be through the granite formation. It was recognized that a more complete screening of the FEPs may identify additional scenarios of interest, and may also show that some aspects of the chosen scenarios do not need further analysis. For a generic, preliminary PA, five scenarios were identified:

1. **Nominal Scenario—The waste packages and engineered barrier system perform as designed.** In the nominal scenario, consistent with the base or expected scenarios for granite repository analyses for Canadian and Swedish waste disposal, no releases occur because the engineered barrier system will protect the waste packages from significant damage, leaving the waste form contained within the waste package during the entire performance period, and preventing radionuclide release.
2. **Defective Waste Package Scenario—A major defect in a waste package allows early radionuclide release.** In this scenario, a waste package in the capture zone of the future groundwater well was assumed to have a major defect in the canister and its contents at the time of emplacement, was modeled as having no barrier performance capability. One waste package was selected instead of multiple waste packages based on estimates of undetected defect rates and the number of waste packages in the capture zone. Exposing the full inventory of a defective waste package is a pessimistic assumption because undetected defects would likely be small, so that the waste package and its contents would still provide some performance, and, in addition, complete failure of all cladding in the defective waste package is extremely unlikely. But for purposes of the analysis, at the time of repository closure the entire waste form was assumed to be exposed to water and beginning to degrade. In this scenario, the buffer, backfill, and seals perform as designed, causing the primary pathway to be through the granite formation. Released radionuclides diffuse through the bentonite buffer and migrate to the well via a near-field fracture and a far-field fracture zone.
3. **Buffer Failure Scenario—Deep circulation of glacial melt waters causes buffer erosion.** In this scenario, corrosion of a number of waste packages was enhanced by advectively flowing corrosive groundwater due to buffer erosion caused by hydrologic changes brought on by the next glacial climate cycle. This scenario assumed that the earth's glacial cycle continues such that an ice sheet or glacier advances over the top of the repository site and then, during a subsequent warming period, retreats at approximately 100,000 years in the future. The warming period was assumed to cause deep penetration of melt waters at the repository site as the ice retreats. The increased flow conditions at depth were assumed to last approximately 25,000 years and to sufficiently erode the buffer to expose one quarter of the waste packages to advective groundwater flow. The increased corrosion rate due to flowing groundwater causes a small fraction of these waste packages to fail within 1 million years, based on a distribution of corrosion rates appropriate for advective conditions. Internal waste package components, such as the insert and fuel cladding, were pessimistically assumed to fail when the canister fails, initiating degradation of all waste in the failed packages. The backfill and seals could be affected in such a scenario, but in this analysis they were assumed to perform as intended. However, for emplacement boreholes containing failed

waste packages, the buffer was assumed to be completely gone, so radionuclides released from breached waste packages were assumed to migrate directly to the host rock.

4. **Shear Movement Scenario—An earthquake causes a displacement that ruptures waste packages.** This scenario was not simulated in the preliminary PA because it was assumed that, as a result of site-selection criteria, the chosen repository location would be in granite with a low probability of significant earthquakes or major glacially induced faulting. In a full PA, site-specific calculations would be needed to confirm exclusion of this scenario.
5. **Disruptive Human Intrusion Scenario.** In a stylized calculation specified by 40 CFR Part 197, a borehole is drilled through the repository and later abandoned, and a vertical hydrologic gradient transports radionuclides to a shallow aquifer from which they are pumped to the biosphere. This scenario is presently required by regulation, but Mariner et al. (2011) considered it less likely to happen in granite and did not analyze it further in their preliminary PA.

8.5.3 Implementation of Models

The scoping PA adopted a number of generic assumptions regarding the repository environment and potential exposure pathway. The repository was assumed to be located in granite where there is a low hydraulic gradient and deep in the saturated zone where reducing conditions would persist even during periods of deep penetration of glacial melt water. Migration of radionuclides along the excavation disturbed zone and through tunnels and shafts was assumed to be insignificant due to effective backfilling and sealing; the excavation disturbed zone was expected not to form a continuous conductive flow path. A future groundwater well is assumed to be constructed downgradient of the repository and within 500 m of the repository footprint. The conceptual model based on this primary pathway is illustrated in Figure 68 (Mariner, et al. 2011, Figure 4-1).

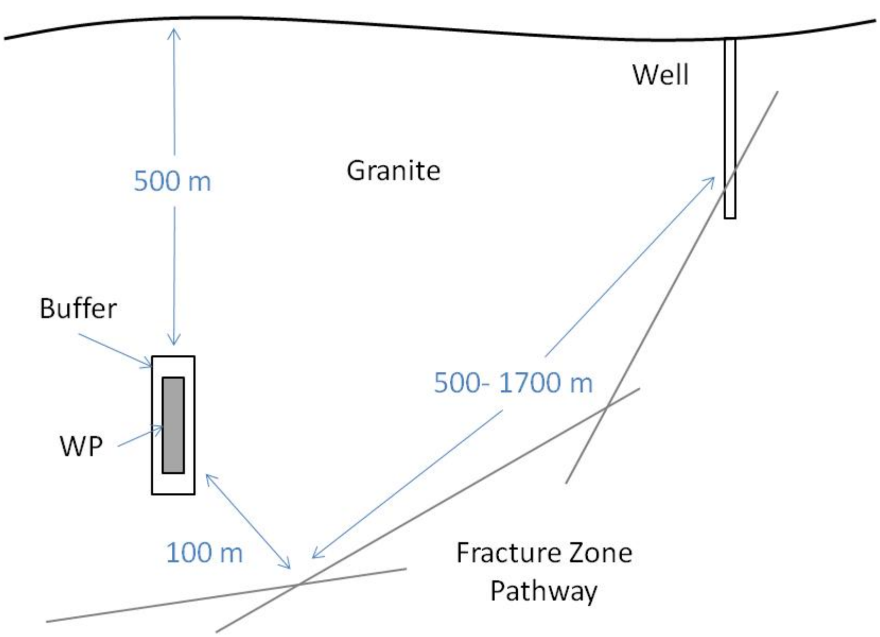


Figure 68. Granite repository PA conceptual model (2011)

The repository is modeled at a depth of 500 m in sparsely fractured granite. The modeled layout is based on the vertical emplacement design and emplacement spacing derived from the repository program in Sweden, with emplacement tunnels 40 m apart and a spacing of 6 m between emplacement boreholes, which contain waste packages with a copper corrosion barrier 50 mm thick. As a simplifying assumption in the preliminary analysis, the affected waste packages in each scenario were assumed to contain SNF. Flow through 500 m to 1,700 m of a hydraulically conductive fracture zone in the granite was represented by a dual-porosity fracture flow model with advection occurring in the fractures and diffusion and sorption occurring in the diffusion porosity. Flow through the fracture zone was modeled as flow through a block of highly fractured granite, which was divided into conduits, each of which captured flow from 100 waste packages. The receptor was assumed to be a family that uses water from a well drilled into the granite within 500 m of the lateral extent of the repository and which was assumed to capture flow from fractures intersecting the emplacement boreholes of 3,000 waste packages (i.e., from 30 fracture zone conduits).

Failure rates for the copper canisters in the buffer failure scenario are sampled from a cumulative probability distribution. For both the defective waste package and buffer erosion scenarios, the canister contents, including the SNF cladding, are effectively treated as nonexistent after failure of the copper canister and provide no further performance. Once the canister is breached, the release of radionuclides from each waste package is limited by (1) waste form degradation rates that initiate upon waste package breach (except for instant release fractions); (2) radionuclide solubility; and (3) when the bentonite buffer is intact, radionuclide diffusion into the buffer surrounding the waste packages (otherwise, in the buffer failure scenario, release is limited by the flux of water flowing through fractures intersecting the waste package borehole).

All modeling for the preliminary granite PA was done in GoldSim. The PA of the generic granite repository was conducted for two scenarios: (1) the defective waste package scenario, and (2) the buffer failure scenario. The defective waste package scenario was analyzed deterministically using the best estimate values for the model parameters; the buffer failure scenario analysis was probabilistic, using the Monte Carlo sampling technique for appropriate model parameters.

In the defective waste package scenario, one waste package was modeled as having failed at the time of repository closure, exposing the full inventory of the waste package, which was assumed to contain commercial SNF. The potential performance of the fuel cladding was not considered. In this scenario, the bentonite buffer, backfill, and seals perform as designed and the primary transport pathway for released radionuclides is through the geosphere. Released radionuclides diffuse through the bentonite buffer and migrate to the well via a near-field fracture and a far-field fracture zone.

For the buffer failure scenario, 25% of the waste packages upgradient from the receptor well were assumed to become exposed to advective groundwater flow at 100,000 years (the time at which the warming trend in the glacial cycle is assumed to cause deep penetration of melt waters at the repository site, eroding the buffer). A probability distribution of higher copper canister corrosion rates was used for waste packages exposed to advective groundwater flow. The canister contents, including fuel cladding, were assumed to provide no barrier performance capability after failure of the canister. For simplification, all failed waste packages were assumed to contain SNF. The probabilistic analysis for the buffer failure scenario was conducted using GoldSim, with a total of 1,000 realizations. Three model parameters were sampled in the analysis: (1) waste package corrosion rate; (2) commercial SNF fractional degradation rate; and (3) the far-field fracture zone length.

When assumptions were made in the granite repository PA, the goal was to make them reasonable and realistic. However, conservative assumptions were adopted in some instances to ensure simplicity and to accommodate uncertainty in a generic assessment. The more conservative assumptions in the analyses included:

- Once the copper canister is breached, the copper canister, insert, and fuel cladding were modeled as if they completely disappeared, ignoring continuing performance contribution of the structural insert and fuel cladding, which, based on data on Zircaloy cladding, may have a performance lifetime of at least 100,000 years.

- Corrosion products from the canister and its contents were not modeled as retarding radionuclide release, though the release of many radionuclides is likely to be retarded due to strong adsorption to corrosion products.

- No lateral dispersion. All radionuclides were modeled as remaining within the confines of the modeled conduits and migrating toward the receptor well.

- In the buffer failure scenario, complete removal of the buffer for 25% of the waste packages after the first glacial period, a percentage based conservatively on probability distributions assuming spalling. In comparison, the most recent PA analyses for Sweden's granite repository program (SKB 2011) estimate a much lower fraction in the reference evolution consistent with limited or no spalling. Within one million years, those anayses estimated that only about 0.4% of deposition boreholes become exposed to advective conditions in the reference evolution; in the most unfavorable cases simulated for the Forsmark repository in Sweden, this percentage increased but remained below 10%.

8.5.4 Results

For the for the defective waste package scenario, the PA calculated radionuclide release rates (radionuclide mass flux) (1) from the failed waste package to the buffer; (2) from the buffer to the near-field granite; (3) from the near-field granite to the far-field granite fracture zone; and, finally, (4) release rates from the far-field granite fracture zone. The resulting annual dose rate to the receptor for the defective waste package scenario is shown in Figure 69 (Mariner, et al. 2011, Figure 4-10). Consistent with from the far-field radionuclide release rate, ^{129}I is the dominant radionuclide for dose, and the peak dose rate is calculated to be about 4.8×10^{-2} mrem/yr at 180,000 years. The dose rates from ^{36}Cl and ^{14}C were much smaller.

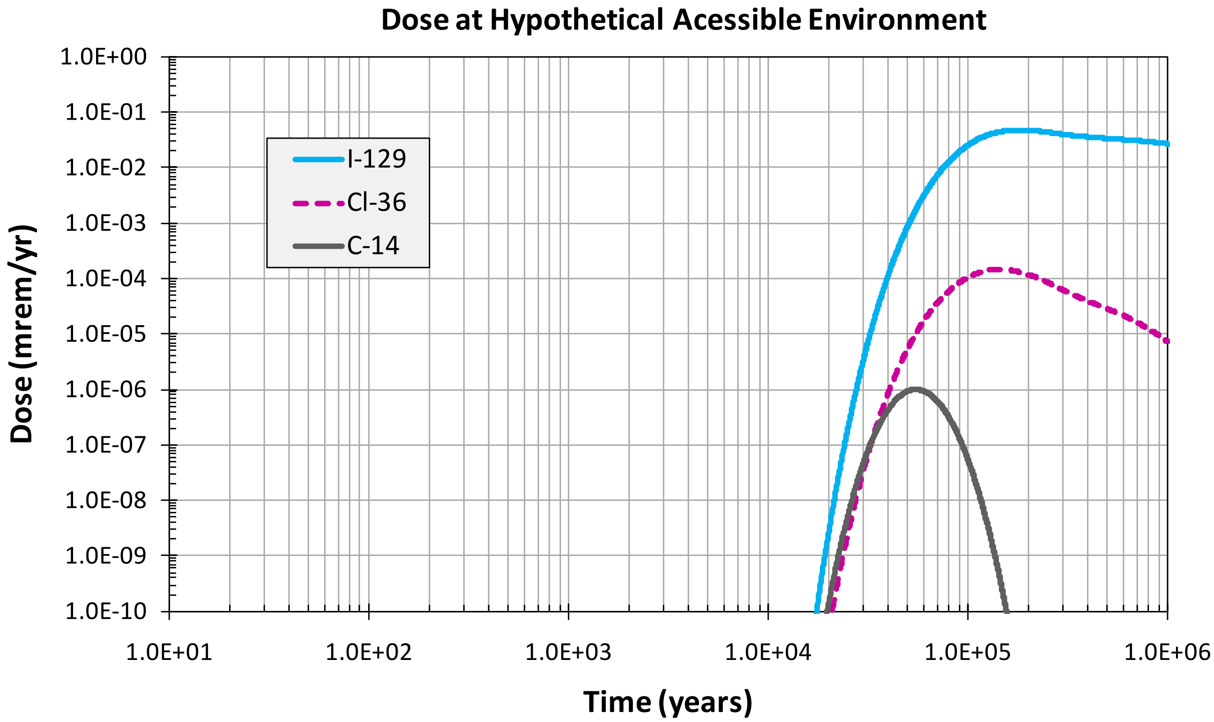


Figure 69. Granite repository PA results for the defective waste package scenario: Dose rate to the hypothetical receptor

In the probabilistic analysis for the buffer failure scenario, only nine of the 1,000 realizations produced a value for waste package corrosion rate that was great enough to fail waste packages within the performance period of 1 million years. For those realizations, all 750 waste packages assumed to be impacted (i.e., 25%of the 3,000 waste packages) were modeled as failed at the same time calculated by the sampled corrosion rate.

For the for the buffer failure scenario, the PA calculated mean releases (radionuclide mass flux) (1) from all failed waste package groups to the near-field granite; (2) from the near-field granite to the far-field granite fracture zone; and, finally, (3) from the far-field granite fracture zone conduits. The resulting annual dose rate to the receptor for the defective waste package scenario is shown in Figure 70 (Mariner, et al. 2011, Figure 4-14). ^{129}I is the dominant dose contributor, and its contribution continues to increase over the entire analysis period. The peak dose rate from ^{129}I is about 0.08 mrem/yr at 1 million years.

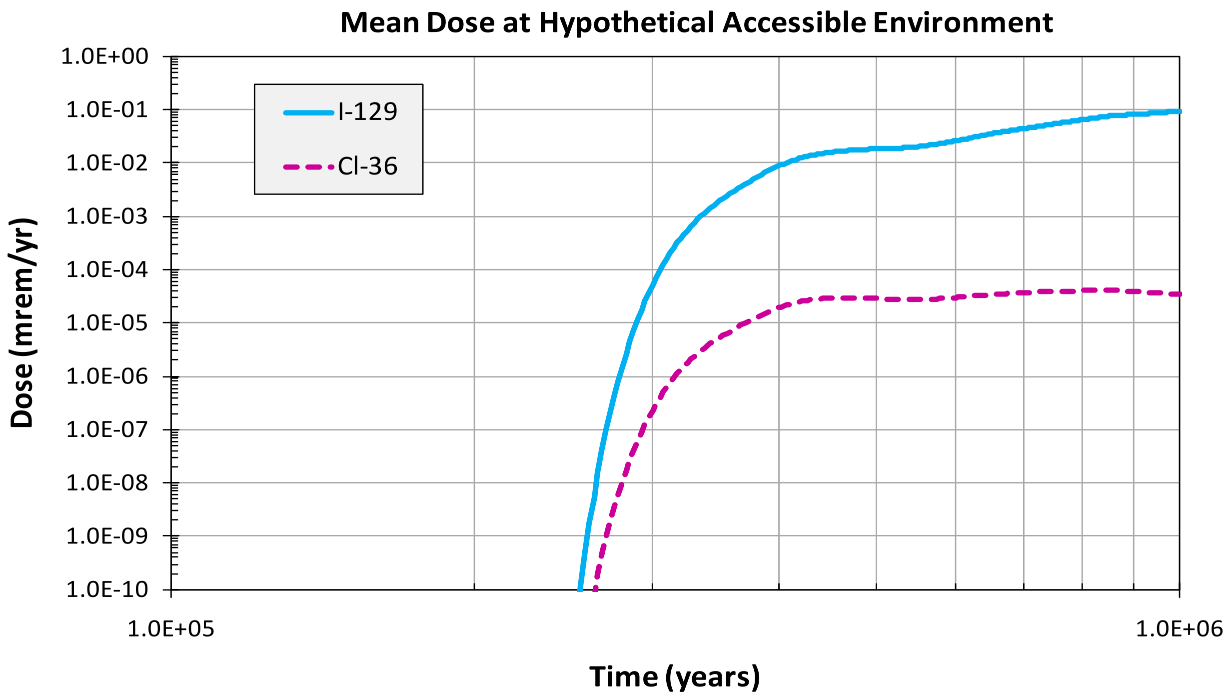


Figure 70. Granite repository PA results for the buffer failure scenario: Dose rate to the hypothetical receptor

Analysis of the relative performance contribution of system features and processes indicated that the most important simulated processes for preventing release of radionuclides from the repository would be canister corrosion, waste form degradation, and radionuclide precipitation, which, in turn, depend highly on reducing conditions and the presence and properties of the canister and buffer. The buffer acts as a diffusive and sorptive barrier to radionuclide transport; however, the results suggest that the buffer's role in limiting waste package canister corrosion rates is more important to repository performance. Once the radionuclides enter the geosphere, fracture flow velocities, matrix diffusion, adsorption, and radioactive decay would be most important to the dose rate at the receptor well.

Consistent with results of safety assessments performed for sites in Sweden, Finland, and Canada, the results of the generic granite repository PA indicate that a granite repository could satisfy established safety criteria. The PA suggested that a small number of FEPs would largely control the release and transport of radionuclides. A proposed site for a granite repository would require a specific design concept, reliable data, and a detailed PA; nevertheless, due to the favorable results from international safety assessment models and the preliminary conclusions of the generic safety analysis presented by Mariner et al. (2011), and because suitable granite bodies are widely available across the U.S., a mined granite repository is likely to be a feasible option for the disposal of SNF and HLW.

8.6 Egypt, Iraq, and Taiwan: PAs and General Waste Management Support for International Radioactive Waste Management Programs

8.6.1 Egypt: Integrated Management Program for Radioactive Sealed Sources

The Integrated Management Program for Radioactive Sealed Sources is a joint project between Sandia National Laboratories, and the Government of Egypt, funded by the U.S. Agency for International Development. It was initiated, in part, in response to an incident in May 2000 involving an ^{192}Ir source that was mismanaged by workers, resulting in exposures of many Egyptian villagers to excessive radiation and in two deaths, including a nine-year-old child who found it and brought it home, believing it to be precious metal. Radioactive sealed sources have been used in Egypt for over 50 years in a wide range of peaceful applications, including, most significantly, oil exploration and medicine. In addition to concern over accidental mismanagement of radioactive sealed sources, terroristic misuse of such sources as components of dirty bombs is a concern in Egypt as it is around the world.

Egypt currently stores hundreds of disused radioactive sealed sources, some of which contain long-lived radionuclides such as ^{241}Am and ^{226}Ra, or high activities of intermediate half-lived nuclides such as ^{137}Cs and ^{90}Sr (Cochran, Hasan, et al. 2004). While near-surface storage or disposal may be appropriate for most of the radioactive sealed sources, for long-lived or high-activity sources, geologic isolation is needed. The analyses of the GCD borehole disposal approach, as described in Section 6 of this report, demonstrate the potential utility of intermediate-depth disposal in thick arid alluvium such as are widely found in the deserts of Egypt. The PA for GCD compliance assessment (Cochran, Beyeler, et al. 2001) demonstrated that intermediate-depth disposal in thick arid alluvium may isolation long-lived radioactive sealed sources from the biosphere for thousands of years.

Based on siting criteria derived from the GCD borehole approach, Egyptian scientists selected six preliminary sites, and reduced them to three. With assistance from Sandia and the IAEA, they will choose a single site for further characterization and safety assessment (Cochran, Carson, et al. 2006). The Egypt Atomic Energy Authority and Sandia partnered in the development of an assessment of site-specific safety based on the prior experience and analyses of the GCD boreholes at the Nevada National Security Site. GoldSim was selected to construct the probabilistic system model for the preliminary PA for the Egypt Atomic Energy Authority GCD study, with uncertain parameters defined as a distribution, utilizing Monte Carlo simulation with Latin Hypercube sampling, and utilizing the GoldSim contaminant transport module with transport and source term elements (Mattie and Cochran 2004).

In cooperation with the Egypt Atomic Energy Authority, SNL also supported a study of hydroxyapatite, $Ca_{10}(PO_4)_6(OH)_2$, which has a high affinity for the sorption of many radionuclides and which is being considered as a reactive backfill material that could help ensure containment and prevent migration of radionuclides from a GCD borehole (Hasan, et al. 2004). Hydroxyapatite has many properties that make it an ideal material for use as a backfill, including low water solubility, high stability under reducing and oxidizing conditions over a wide temperature range, availability, and low cost. The testing assessed differences in important containment-relevant properties of hydroxyapatite that can vary depending on the material's source and method of preparation.

8.6.2 Iraq Nuclear Facility Dismantlement and Disposal Program

Since 2006, Sandia National Laboratories has been a part of a U.S. Department of State team implementing the Iraq Nuclear Facility Dismantlement and Disposal Program, along with other U.S. participants, including the DOE, the EPA, the NRC, Texas Tech University and others. Iraq never had a disposal facility for radioactive wastes, and the effects of the two Gulf Wars, lack of upkeep, loss of records, and looting in the aftermath of the second Gulf War resulted in an enormous radioactive waste problem (Cochran, Daneels and Kenagy, et al. 2007, Cochran and Daneels 2009). As part of this program, SNL provided training and technical consultation, introducing Iraqi scientists and representatives of the Iraqi government to modern decommissioning and waste management practices, and supporting the International Atomic Energy Agency as they assist the government of Iraq.

Sandia provided training and technical consultation to Iraqi scientists and engineers, including support in establishing training programs for radiological workers; guidance in concepts for project management planning and help in developing a project management plan for the Stage 1 Decommissioning of the Active Metallurgical Testing Laboratory facility at Al Tuwaitha; recommendations on quality assurance; a conceptual design for a sorting and storage facility for radioactive waste, based on facilities at Sandia but tailored to the situation at Al Tuwaitha; and detailed advice regarding purchase and use of radiation protection equipment for remediation work at Al Tuwaitha. In addition, Sandia provided training on monitoring groundwater at liquid radioactive waste tanks, which included both classroom instruction and field trips to observe operating equipment.

Sandia also provided training, which included Jordanian nuclear officials in addition to the Iraqis, detailing concepts for siting, licensing, constructing, and operating permanent radioactive waste disposal facilities appropriate to the climate and regional geology and hydrology in Iraq and Jordan. This training was greatly informed by the experience and PA studies from the GCD program on the Nevada National Security Site, as described previously in Section 6. Sandia led the Iraqi scientists on a tour of the Area 5 Radioactive Waste Management Site where the GCD boreholes are located, as well as the EnergySolutions LLW disposal facility near Clive, Utah, which is one of the three commercial facilities licensed by the NRC for disposal of LLW

In 2005, when the IAEA began organizing an international effort to help Iraq address its radioactive waste problems, the government of Iraq had almost no plans, no procedures, no teams, and no infrastructure to initiate decommissioning or remediation work. In July 2008, Iraq had begun the on-the-ground dismantlement of the Active Metallurgical Testing Laboratory facility at Al Tuwaitha (Cochran, Daneels and Kenagy, et al. 2007), demonstrating the great progress made by the Iraqi government with support of the international community, the IAEA, and the U.S. Department of State's Iraq Nuclear Facility Dismantlement and Disposal Program, through which Sandia was able to contribute their experience and expertise.

8.6.3 Taiwan: Low-Level Radioactive Waste Disposal, Preliminary Analyses

Taiwan currently has three operating nuclear power plants with a total of six reactors, and a fourth plant scheduled for startup in late 2011. Low-level radioactive wastes from the nuclear power plants requiring permanent disposal are produced by operational activities and will be produced by decommissioning. Temporary storage exists at each nuclear power plant site and on the island of Lanyu, but these are not permanent options. Taiwan anticipates generating 966,000 55-gallon drums of LLW over the lifetime of the four nuclear power plants, based on 40 years of operation. In addition, Taiwan needs a method for disposal of other medical and research wastes.

After working directly with the Taiwan Institute of Nuclear Energy Research since 1998 in an exchange of technical information and geologic repository experience, Sandia assisted Taiwan in (1) providing a regulatory analysis of LLW final disposal, (2) development of LLW performance assessment capabilities using NRC-sponsored codes and other computational tools, and (3) conducting performance assessments for two potential LLW final disposal sites using available site and initial conceptual design information (Arnold, et al. 2007).

Performance objectives for the preliminary PA were based on regulations in Taiwan and comparisons to those in the United States. Probabilistic performance assessment models were constructed based on limited site data using software including:

- GoldSim, providing the probabilistic model framework

- BLT-MS (NRC's Breach, Leach, and Transport—Multiple Species code), used here to simulate waste-container degradation, waste-form leaching, and advective-diffusive transport through the host rock)

- FEHM, for modeling groundwater flow and transport velocities

- HELP (EPA's Hydrologic Evaluation of Landfill Performance model), applied here in evaluating infiltration through the disposal cover system.

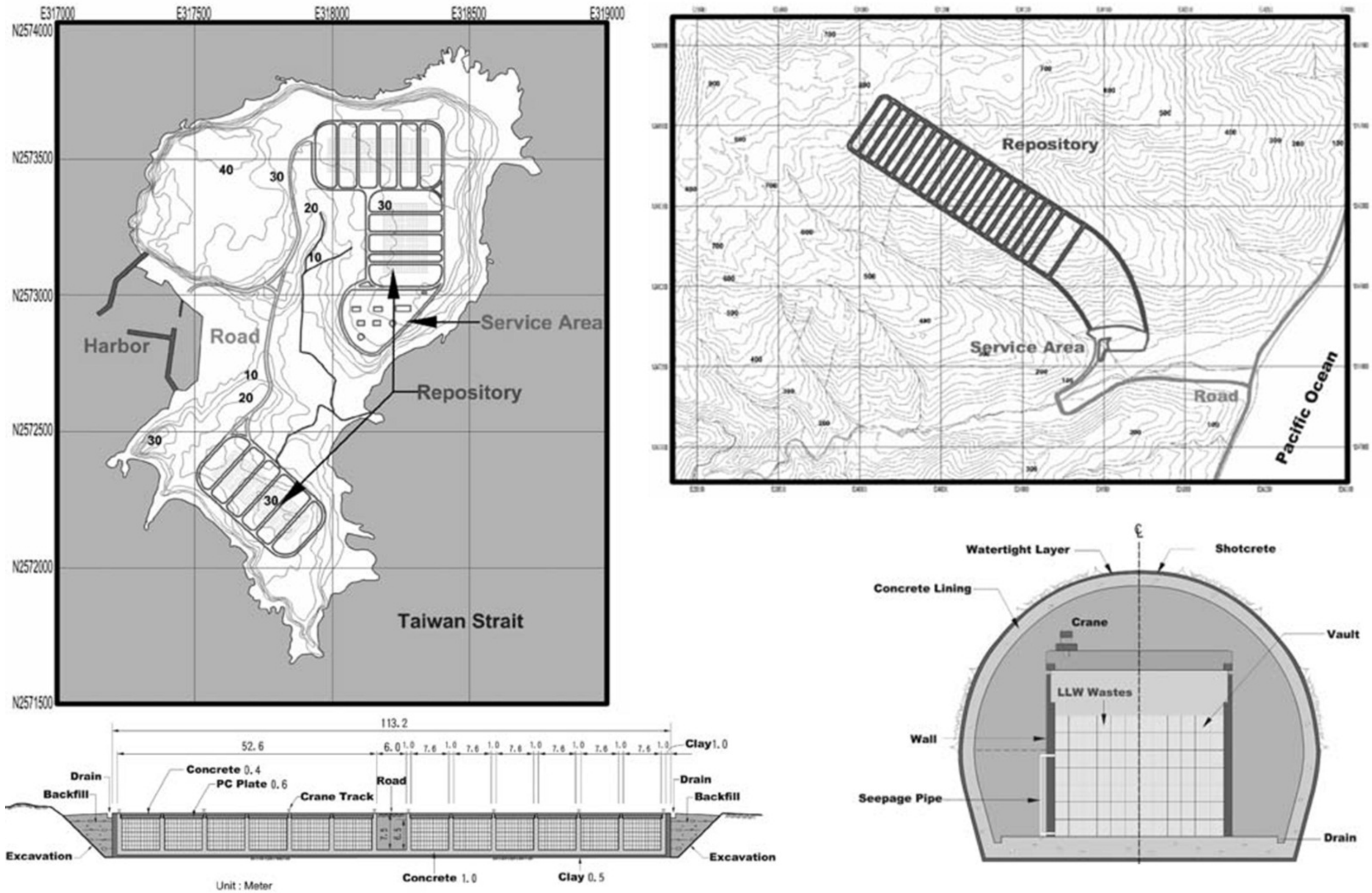
Preliminary performance assessment analyses were conducted for two representative sites in Taiwan: (1) a near-surface disposal system on a small island off the western coast of Taiwan, with basalt bedrock and interbedded sedimentary rock, and using an engineered cover system to limit infiltration; and (2) a mined cavern disposal system located along the southeastern coast of the main island, with a mined tunnel system about 500 to 800 m below the surface in bedrock consisting of argillite and meta-sedimentary rocks. These two sites are shown in Figure 71 (Arnold, et al. 2007, Figures 4 and 5). Though Sandia presented and recommended the methodology for formal FEPs analysis to the Taiwan Institute of Nuclear Energy Research, it was not implemented in this preliminary PA. The Institute of Nuclear Energy Research has made some progress on adopting the formal FEPs methodology, but may defer full implementation to future stages of the program. The conceptual models, therefore, are based on general knowledge of the site, literature data, and limited site characterization data (hydrogeologic and geochemical data, for example, were not available for these analyses; literature data were used instead or processes were omitted from the model). Some future scenarios (e.g., typhoon events, a sea level rise of as much as 2 m in the next 300 years, and seismic activity) may need to be assessed in the PA in future analyses. The compliance point is assumed to be 100 m from the edge of the disposal cell. The deterministic calculations were run for 10,000 years, but the probabilistic models calculated results for the first 1,000 years.

Near-Surface Disposal System on the Island Site—The conceptual model for the near-surface disposal system on the island site consists of three separate disposal areas, with the waste inventory assumed to be split evenly among the three disposal areas. For purposes of the preliminary PA, only one of these disposal areas was considered; correspondingly, only one-third of the estimated total LLW inventory was considered in the preliminary PA. The base-case cover design includes two concrete caps 60 cm and 40 cm thick over the disposal cell with an earthen cap above the concrete.

The waste inventory in the source term was derived from information provided by the Taiwan Institute of Nuclear Energy Research along with a number of assumptions where information or data did not exist. Their chosen design concept assumed that all the waste would be encapsulated in concrete and grout. Concrete and grout generally have about a 300 year failure time or greater. Localized corrosion of the waste drums would likely be occurring during this time, but releases would not occur due to the concrete/grout encapsulation, so generalized corrosion was invoked as the release mechanism with a failure time of 300 years.

The first conceptual model of the site involves a very conservative assumption about the leaching process after failure of the waste drums. The assumption is that all the waste is subject to rinse release, allowing the radionuclides to freely mix with the incoming infiltration water, with little credit given for source term controls. The rationale behind invoking this conceptual model is that if the performance of the system under this set of unrealistic assumptions demonstrates compliance with the standard, then the site is likely suitable because the engineered barrier system would further improve performance. A conceptual model with more realistic source-term assumptions reduced estimated doses by half in comparison to the rinse-release model. A third conceptual model was designated the baseline case model because it incorporates the best estimates of the waste release mechanisms as well as a representation of the engineered barrier system performance, including effects of concrete and grout. A fourth conceptual model addressed the possibility that all wastes disposed at the facility might be required to be solidified, resulting in wastes being subjected to a diffusion release mechanism. These models were implemented deterministically. The results for the latter three models were nearly identical, reducing the peak dose by about half in comparison to the rinse-release model.

Modeling of infiltration for the near-surface disposal system indicated percolation of about 45 mm/yr through the base-case cover design; however, sensitivity analysis was also conducted showing that the simulated percolation flux is reduced to about 3 mm/yr if a high-density polyethylene geomembrane is added to the design below the drainage layer. Results of example calculations indicate peak simulated concentrations to a receptor within a few hundred years of disposal, primarily from highly soluble, nonsorbing radionuclides.



Island Site for Near-Surface Disposal System Concept
and Cross Section of Disposal Cell

Site for Mined Cavern Disposal System Concept
and Cross Section of Disposal Tunnel

Figure 71. Island near-surface disposal system site and disposal concept (left) and mined cavern disposal system site and concept (right)

The four base-case analyses used the two-dimensional BLT-MS transport model. An alternative model, using the one-dimensional transport model implemented in GoldSim, consisting of three pathways from different locations within the disposal cell, with a vertical pipe segment representing flow in the unsaturated zone beneath the disposal cell and a horizontal pipe segment for flow in the saturated zone for each pathway.

Finally, a probabilistic analysis was run for the baseline conceptual model varying two parameters, the container general corrosion rate and the radionuclide-specific effective diffusion coefficients. The analysis was run for 100 realizations over a period of 1,000 years. Because the container corrosion rate was treated as uncertain, releases begin earlier (reflecting an earliest container failure time of approximately 220 years) compared to the deterministic baseline analysis, where releases were modeled to occur at 300 years. The combined effects in this example yielded a higher normalized dose on average, than that of the expected case as represented by the deterministic base case results. Given the high number of assumptions in the design and site properties for the near surface disposal site, the uncertainty study displays the importance of an iterative approach that couples design and feedback to the safety assessment.

Mined Cavern Disposal System—The conceptual model for the mined cavern disposal system consists of are a series of 21 disposal tunnels, 400 m long and spaced 63 m apart. In the center of each a concrete vault is constructed, divided into 10 disposal cells where waste is disposed in 55-gallon galvanized drums. Each disposal cell holds 4,704 waste drums, yield a total repository capacity of 987,840 drums. The waste inventory is modeled as being split evenly amongst all of the disposal tunnels. Disposal cells are grouted, a concrete cap is placed over each disposal cell, and, when the entire vault is filled, the tunnel is backfilled. The models are reasonably representative of the specific site being considered in the southeast portion of Taiwan in the side of a mountain, though, as with the island site, the availability of detailed site data is limited and additional characterization would be needed for a PA after site selection. In addition, the model was set up to account for effects from the concrete and grout surrounding the waste drums. The concrete/grout has a different porosity and molecular diffusion coefficient than the surrounding host rock. Diffusion of the radionuclides through the concrete/grout should impede the release to a degree. This conceptual model is designated as the baseline case model in that it incorporates the best estimates of the waste release mechanisms and includes a representation of the engineered barrier system performance.

The model domain for the mined cavern site is considerably larger than that of the near-surface disposal design for the island site. Two conceptualizations of the mined cavern site were developed in order to facilitate the objectives. One conceptualization attempted to honor as much of the specificity of the tunnel design as possible, resulting in a relatively large and computationally burdensome finite-element grid, having a total 29,988 finite-element nodes and 29,480 finite elements. The second, simplified conceptualization was coarser but easier to implement in a probabilistic framework because it is less computationally burdensome (with 10,578 finite-element nodes and 10,280 finite elements, a little over a third of the nodes and elements in the detailed model). A comparison of the two conceptual models allowed conclusions to be drawn as to the representativeness of each model. The behavior of the two representations was slightly different, with the simplified model suggesting more diffusion-dominated behavior, releasing radionuclides to the far field more slowly after container failure and predicting slightly higher dose rates after about 400 years. But these differences were small enough to suggest that the simplified model would be adequate for implementation in probabilistic calculations.

As with the analyses for the near-surface island site, a conservative rinse release case based on the simplified model was run. In another variation on the simplified conceptual model, the operational waste inventory was assumed as split between a nonsolidified waste fraction with a 50% rinse and 50% diffusion release specification and a solidified waste fraction with diffusion release, the decommissioning waste then was split between a solidified non-metal waste fraction with rinse and diffusion release and a metal waste fraction with a dissolution release. Yet another variation of this conceptual model assumes that all the waste is solidified (e.g., grout added to the waste in the disposal drums) and subject to a diffusion release. The results suggested that conceptual models that invoke either all rinse or even partial rinse release mechanisms exhibit very similar behavior. The conceptual model that has some dissolution release of metal waste in addition to diffusion release had essentially the same response as just diffusion release. Therefore, the results indicated performance benefits from solidifying most of the waste, with the potential exception of the metal waste from decommissioning.

An alternative model was constructed using a one-dimensional matrix-diffusion transport model implemented in GoldSim, rather than the two-dimensional BLT-MS model implemented in the baseline conceptual model. The alternative model consisted of 21 pathways, each one originating from one of the 21 disposal tunnels. The one-dimensional transport model results indicate a higher peak normalized dose value (by a factor of approximately 2) and a slightly higher normalized dose rate out to times of about 2,500 years in comparison to the two-dimensional BLT-MS model.

The probabilistic analysis for the mined cavern site was developed the same way it had been for the near-surface island site, by varying the same two parameters (i.e., container general corrosion rate and the radionuclide-specific effective diffusion coefficients) using the same distributions for 100 realizations over a period of 1,000 years. Because the container corrosion rate was treated as uncertain, releases begin earlier (reflecting an earliest container failure time of approximately 220 years) compared to the deterministic baseline analysis, where releases were modeled to occur at 300 years. The combined effects in this example yielded a higher normalized dose on average, than that of the expected case as represented by the deterministic base case results. Given the high number of assumptions in the design and site properties for the near surface disposal site, the uncertainty study displays the importance of an iterative approach that couples design and feedback to the safety assessment.

Results—The baseline deterministic and probabilistic results are provided in Figure 72 for both the near-surface disposal site for located on a small island in the Taiwan Strait and the mined cavern site located in the southeast of the main island of Taiwan, near the coast (Arnold, et al. 2007, Figures 55, 56, 73, and 74). Because the modeling is at such an early stage and the general lack of site data for each site under consideration and the uncertainty that is created in these preliminary modeling results, dose values are presented as normalized quantities. The maximum total dose estimate from the rinse release case for the near-surface island site (i.e., shown in black in the upper left plot in Figure 72) was used as a divisor for all dose estimates to put all the output on a relative scale without having to publish a dose estimate. To compare a dose estimate against any standard of compliance with these preliminary results would not be appropriate. Likewise, though normalized dose results allow such comparisons, caution should be used in drawing firm or final conclusions regarding the relative suitability of these two disposal systems at these two sites based on these preliminary PA analyses.

Keeping that caveat in mind, these preliminary assessments indicate values of peak simulated dose from the near-surface disposal system nearly two orders of magnitude greater than the mined cavern disposal system. This is significant when also considering that the source term modeled in the near-surface disposal system was only one-third that of the mined cavern configuration due to the fact that only one of the three disposal cells was considered for the near-surface disposal site.

These differences in the performance assessment analyses at the two sites are primarily due to differences in the volume of groundwater in which the radionuclides are dissolved for calculating their concentrations. The near-surface site is located on an island with limited volumetric groundwater flow rates through an aquifer that is probably relatively thin. Precipitation of about 1 m/yr at the island site is significantly less than the average of about 2.6 m/yr at cavern site on the southeast coast of Taiwan's main island, resulting in less recharge to the groundwater flow system. There are differences in the BLT-MS model setup for the two sites that lead to greater numerical dispersion and associated dilution for the mined cavern disposal site, but these differences in model domain size largely reflect probable differences in the physical groundwater flow system.

It is possible that some of the difference in performance assessment model results for the two sites can be accounted for by the preliminary nature of the underlying groundwater flow models. However, Arnold et al. (2007) concluded that the physical layout of the repository designs and the nature of the groundwater flow systems at the two sites support the model results that indicate much larger groundwater volumetric flow rates intersecting the LLW at the mined cavern site, relative to the near-surface disposal site on the small island.

8.7 Preliminary Development of an Enhanced PA System for Geologic Carbon Sequestration (2010)

Carbon capture and geologic sequestration is one important approach to help mitigate impacts of atmospheric carbon emission currently being investigated at a number of sites in the United States and around the world, including a few sites where CO_2 has been injected into geologic formations for sequestration. Recent efforts have been made to apply the existing probabilistic PA methodology developed for geologic repositories for nuclear waste to evaluations of the effectiveness of subsurface carbon storage. However, most of the existing modeling effort focused on detailed physical and chemical processes, with little attention to uncertainty quantification of the model predictions, and systematic application of the PA methodology to geologic carbon sequestration systems is still undergoing development, and could benefit from enhanced capabilities.

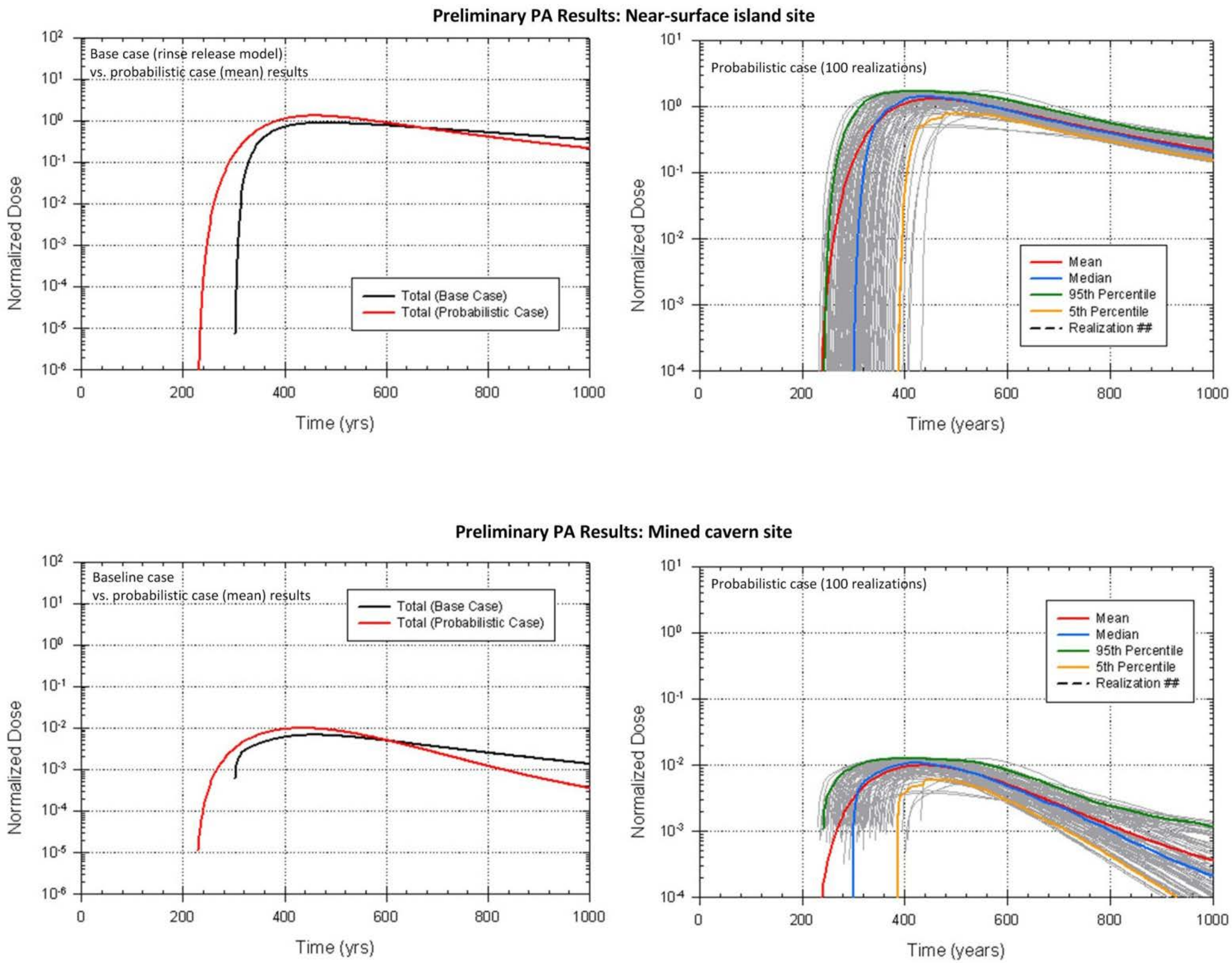


Figure 72. Taiwan LLW PA Results: Normalized dose for deterministic and probabilistic models for both the near-surface island site and the mined cavern site

With support provided by the SNL Laboratory Directed Research and Development program, Wang, Dewers, et al. (2010) outlined a new methodology for an enhanced PA system, performed a preliminary FEPs analysis for a hypothetical geologic carbon sequestration system, developed a prototype PA model using TOUGH2 and DAKOTA codes, and successfully implemented that system model in a probabilistic analysis based on the Frio pilot injection project near the Texas Gulf Coast, which utilizes the Oligocene Frio Formation as storage reservoir and the overlying Miocene Anahuac Formation as caprock. The enhanced PA model was designed to be generally applicable to geologic carbon sequestration pilot projects currently underway as part of the DOE/National Energy Technology Laboratory partnerships for carbon sequestration (NETL 2011).

8.7.1 Enhanced PA System Methodology

In the evaluation of a nuclear waste repository, a PA model is essentially a forward model that samples input parameters and runs multiple realizations to estimate future consequences and determine important parameters driving the system performance. The enhanced PA system methodology, shown in Figure 73 (Wang, Dewers, et al. 2010, Figure 2), provides PA model able to run both forward and inverse calculations to support optimization of CO_2 injection and real-time site monitoring as an integral part of the system design and operation. The forward model components represent the typical steps of the existing PA methodology, starting with FEPs evaluation, development appropriate computational models for the selected FEPs and scenarios, and then constraining model input parameter values and their uncertainty distributions based on field observations and laboratory experimental data. As with the typical PA approach, the PA analysis is then completed by uncertainty quantification and sensitivity analysis, typically performed using multiple Monte-Carlo simulations. The overall process remains iterative, with preliminary results used to inform and adjust subsequent iterations.

The enhanced PA approach extended the existing PA methodology by adding the inverse model components shown in Figure 73. These inverse components provide necessary tools for process optimization of CO_2 injection, updating of parameter estimates as new data are obtained, as well as optimization of long-term system performance. "Data fusion," is the direct integration of multiple land- and satellite-based sensors into an adaptive integrating modeling system. It involves the combination of data from multiple sources in a structured fashion, delivering inferences or meaning more efficiently and accurately than by review of separate sources of information delivered independently. For example, pressure monitoring from CO_2 injection wells could be combined with satellite land surface monitoring to highlight and evaluate areas of potential release. Adaptive modeling and mesh or parameter refinement could also utilize the updated pressure information to further calibrate the site CO_2 model. The timescale for such analysis and updating would be dependent on site and injection characteristics.

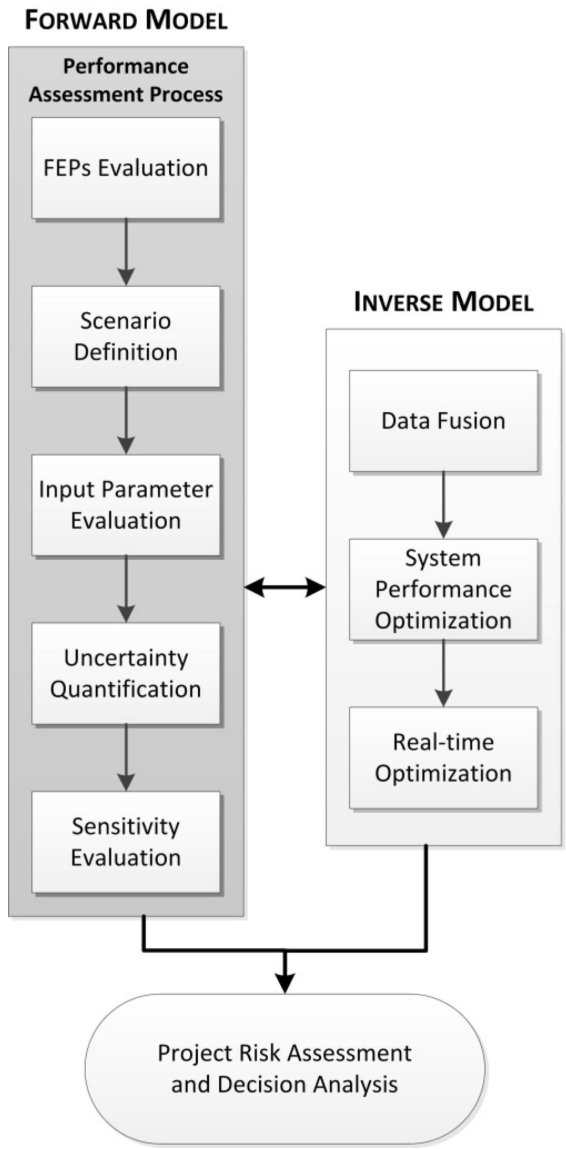


Figure 73. Enhanced PA methodology for geologic carbon sequestration

8.7.2 PA Development and Implementation

For the preliminary PA, a simplified conceptual model was developed based on the Frio pilot injection project near the Texas Gulf Coast, currently being investigated for suitability for carbon storage. That pilot project is utilizing the brine-filled Oligocene sandstone of the Frio Formation as a storage reservoir and the overlying Miocene Anahuac Formation as caprock. Figure 74 provides a schematic illustration of how the sequestration system was modeled (Wang, Dewers, et al. 2010, Figure 4). The overall system contained an injection well and reservoir, with system flow ultimately leading to the biosphere through a fault, abandoned unsealed well, or caprock. For simplification, fixed hydraulic pressures were imposed on the top surface of the shallow sandstone formation, while no flux condition was imposed on the bottom surface of the host rock. Fixed hydraulic pressures were imposed on both left and right sides, which were assumed to coincide with faults.

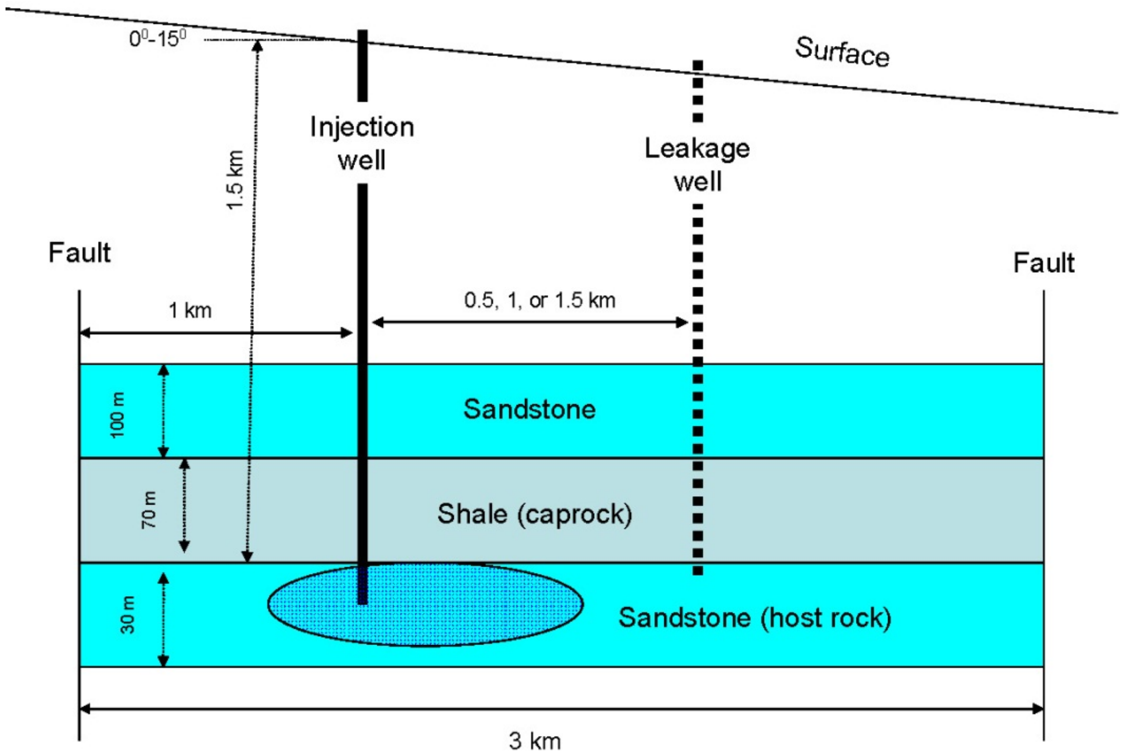


Figure 74. Schematic illustration of the modeled carbon sequestration system

The FEPs analysis performed by Wang, Dewers, et al. (2010) used the open-access generic CO_2 storage FEP database developed by Quintessa (2010) as its basis. The list consisted of 134 FEPs to be considered; because the analysis was preliminary and for a generic system, some FEPs (e.g., effects from human activity upon the system and impacts on humans and flora and fauna) were not considered, but most others had a preliminary screening decision applied for them.

The base-case scenario for the analysis included caprock overlying the potential injection zone, and a vertical fault on the boundary of the injection zone, with 30-year duration for injection, and a total simulation time of 100 years. The scenario also included the potential for CO_2 release from an abandoned borehole in the formation (considered to be located at 1 km from the injection well). Five other scenarios were included:

1. Multiple abandoned, leaking boreholes—injection to base case system with the potential for release from multiple abandoned and degraded closed wells
2. Leaking caprock—overpressure in the system due to injection rates allows leakage through the caprock
3. Impact on upper groundwater aquifer—injection through the system escapes to hypothetical upper groundwater aquifer and impacts the water chemistry in the aquifer.
4. Heterogeneous injection zone—an injection zone with heterogeneities in permeability/porosity.
5. Surface facility optimization—incorporating surface facilities, including cost and alternative injection rates.

Input parameters were chosen to loosely correspond to the Frio pilot injection project in Texas. There were 41 input parameters (including five uncertain parameters) representing the physical configuration and geologic setting, operational conditions, hydrologic properties, fluid properties, and the time scale for simulation.

The prototype model was developed by coupling TOUGH2 (Pruess, Oldenburg and Moridis 1999), software widely used for multiphase, multicomponent reservoir simulation, with an uncertainty quantification and optimization code, DAKOTA (Design Analysis Kit for Optimization and Terascale Applications) (Adams, et al. 2010). DAKOTA is a software toolkit that provides a flexible and extensible interface between simulation codes and iterative analysis methods used in large-scale systems engineering optimization, uncertainty quantification, and sensitivity analysis. The DAKOTA toolkit can perform parameter optimization using gradient and nongradient-based methods. It can also be used to conduct sensitivity analysis with the purpose of investigating variability in response to variations in model parameters using sampling methods such as Latin Hypercube sampling, among others. Further capabilities of the toolkit include uncertainty quantification with sampling, analytic reliability, and stochastic finite element methods; and parameter estimation with nonlinear least squares methods. These capabilities may be used on their own or as components within system models. Specific to this study, a DAKOTA-based nondeterministic sampling algorithm was implemented for the enhanced PA system framework. The overall sampling flow involves embedded TOUGH2 functional evaluations within a DAKOTA run. First, a set of uncertain parameters with assigned probability distributions was specified in the DAKOTA input parameter file. A sample was drawn using Latin Hypercube sampling and was processed by an input filter routine to transcribe each sample element, comprising a value for each uncertain parameter, into a formatted template file compatible with TOUGH2. After each sample element was executed, an output filter extracted the pertinent output values via an output filter routine and returned them to DAKOTA.

DAKOTA is designed to support large-scale, computationally intensive simulations. Different levels of parallelism are available in DAKOTA; for this enhanced PA system framework, a hybrid parallelism is assumed. Using DAKOTA provided a level of parallelism at functional evaluation level, allowing three concurrent serial TOUGH2 jobs to be executed at any given time, as long as the computational CPUs are available, thereby shortening the overall calculation cycle. Such coupling could be refined further and expanded to run in parallel on the high-performance computational clusters at SNL.

To demonstrate feasibility of their approach, Wang, Dewers, et al. (2010) constructed a two-dimensional model with a 3,000 m by 200 m rectangular simulation domain, 1,330 m below ground surface (Figure 74). The reservoir was 30 m thick, with both homogeneous and heterogeneous properties. The overlying caprock, with homogeneous properties, was 70 m thick. The aquifer at the top of the domain was 100 m thick. A finite volume grid used in the numerical simulations is superimposed upon the three spatial regions. Injection of super critical CO_2, a leaky well scenario, caprock leakage, and brine and CO_2 migration driven by injection-related rise in pressure are three aspects of the conceptual model that were investigated by numerical means.

A deterministic case was run using an injection rate of 0.35 kg/s distributed equally over five cells of the model grid. The deterministic example illustrated the physics behind the leakage scenarios of leaky well and leaky caprock. In order to minimize simulation times, the DAKOTA realizations assumed homogeneous reservoir properties and assumed zero salinity throughout the simulation domain—assumptions which had little effect on the overall leakage realizations. The multiple realization case exercising the coupled DAKOTA–TOUGH2 system was run for 27 realizations using sampled values of the five uncertain parameters: (1) injection rate, (2) abandoned well productivity index, (3) caprock permeability, (4) caprock porosity, and (5) caprock residual liquid saturation. In all cases, a sampled constant amount of CO_2 is injected for a period of 30 years and the simulation continued to a total time of 100 years to observe CO_2 movement in the injection zone and through the abandoned well and caprock.

The horsetail plots of pressure and saturation for 27 realizations showed a spread in results, a direct effect of the range of sampled parameters. Pressure builds up rapidly in the injection well element and the surrounding elements as more CO_2 is injected and brine is pushed out. The increased CO_2 injection also results in increases in gas saturation. The magnitude of the increases depends primarily on the injection rate and to a lesser extent on the other sampled parameters. Thus, vectors with low injection rates show lower pressure buildup. With time, the pressure buildup decreases as the brine is pushed further and fluid moves to the abandoned well and the caprock, away from the injection well. The assumption of a homogenous reservoir with a permeability of 1.94×10^{-13} m^2 also facilitated movement of fluid away from the injection well. Higher gas saturations in the injection well were maintained by high capillary pressure conditions. When the capillary pressure in the injection well element reaches the maximum (2×10^7 Pa in the reservoir), brine flow into the element is reduced, thereby maintaining high gas saturations are maintained.

Pressure and gas saturation in the abandoned well build up early as a result of fluid (CO_2 and brine) movement, leading to in increased CO_2 leakage. Leakage from the abandoned well increases until around 30 years, when CO_2 injection is stopped. The peak leakage rate is a big portion of the total injection rate, indicating that for this preliminary model the abandoned well is a major conduit for CO_2 migration. In a more realistic model, the heterogeneity in the host rock and the caprock and conditions of the abandoned well would likely control fast migration. The model showed some CO_2 migration into the caprock overlaying the reservoir and also into the aquifer above it, with this movement occurring during the injection period of the 30 years, but showed a small amount of reverse flow after 30 years, when CO_2 injection is stopped.

8.7.3 Conclusions from the Preliminary PA

The prototype enhanced PA system utilizing TOUGH2 and DAKOTA software developed by Wang, Dewers, et al. (2010) lays the foundation for development of a new generation of PA tools for effective management of geologic carbon storage activities. The prototype demonstrates that the PA tools developed by SNL for evaluation of performance of geologic repositories for radioactive waste on the scale of up to 1 million years can be applied to evaluation of geologic carbon storage and even adapted to assist in the real-time monitoring and management of such systems.

The scope of the preliminary PA was limited, but served to identify areas of interest for further development. For example, a more complete evaluation and model development is needed for FEPs that were not included in the initial PA as well as development of additional scenarios.

Further, having successfully demonstrated the application of the DAKOTA toolkit, linkages to various data sources to provide rapid updating of site carbon storage models and develop adaptive modeling tools to further calibrate site CO_2 models would demonstrate the feasibility of the inverse model components of the enhanced PA methodology. Finally, the full breadth of the DAKOTA toolkit can be utilized for additional optimization of the key parameters for a geologic carbon sequestration system and can even be updated for specific PA requirements.

9. CONCLUSIONS AND CURRENT DEVELOPMENTS

Over nearly 40 years, Sandia National Laboratories has developed and applied a PA methodology that has informed key decisions concerning radioactive waste management. This experience includes not only the Waste Isolation Pilot Plant and Yucca Mountain repository projects, recent programs that have wide public recognition, but also less well-known past programs including the development and demonstration of the NRC's initial PA capabilities for both high-level and low-level wastes in a variety of geologic media, the Subseabed Disposal Project, PAs for wastes stored at the Idaho National Laboratory, and PAs for Greater Confinement Disposal boreholes at the Nevada National Security Site, as well as recent, smaller-scale PA studies in support of multiple international collaborations for radioactive waste management.

These efforts have produced a generic PA methodology for the evaluation of total waste management systems that has gained wide acceptance within the international community. More importantly, this methodology has been used as an effective management tool to evaluate different disposal designs and sites in a variety of geologic media; inform development of regulatory requirements; identify, prioritize and guide research aimed at reducing uncertainties for objective estimations of risk; and support safety assessments. As shown by the breadth of PA applications described in this report, the SNL PA methodology is designed to be adaptable to evaluate analyses of different strategies and options that might be proposed to manage the back-end of the nuclear fuel cycle, including both repository and disposal strategies as well as analysis of the disposal implications of diverse waste forms that may be generated. SNL PAs have been demonstrated to successfully evaluate environmental safety and support licensing of radioactive waste disposal. The SNL PA methodology has been successfully applied to computationally simple PAs, such as for GCD borehole disposal, and to very complex PAs, such as for the WIPP and YMP. In fact, the development of the SNL PA methodology for radioactive waste disposal has helped advance the science of probabilistic analysis and computation, being among the early applications of LHS techniques, and, as outlined in these concluding summaries, continues to extend that science with new software to utilize massively parallel processing computing capabilities.

The most important lessons learned from review and comparison of the SNL PA programs include:

- Development of a comprehensive FEP list during the earliest iteration of PA ensures that site characterization and engineering research programs can be appropriately planned; without a sufficiently comprehensive FEP list, scenarios are likely to be revisited and revised significantly at each iteration, resulting in disruptions to both the research program and the modeling effort. The FEPs list for WIPP remained largely consistent and served as a firm foundation for research and PA; in comparison, the FEPs program for Yucca Mountain was not as robust (initial planning allowed for and anticipated changes to the FEPs list on an ongoing basis), and the program struggled somewhat until a comprehensive FEPs list was established in 2001.

- The iterative PA approach can provide a structured framework for the management very large and complex projects, organizing and assessing a vast quantity of data and information in an integrated manner. Beginning with the early PAs for the SDP and continuing through other PAs, particularly the WIPP and YMP, PA has been used to provide consistency between system elements and related research and development activities. PA has been demonstrated as a very effective management tool of evaluating and prioritizing data and other system information so that scientific and engineering activities are focused on those most important to meeting the performance requirements. Use of PA in program decision-making promotes efficient use of scientific and engineering resources and more quickly optimizes system performance.

- In the longest programs, WIPP and YMP, PA was effective in helping to manage the necessary transition from science to compliance. During the science phase both projects focused the technical organization on (1) the scientific and research work needed to understand the behavior of the disposal system and (2) the use of that information in the total system analysis needed to evaluate compliance with the applicable regulatory requirements and safety standards. In the "compliance" phase the emphasis shifted to (1) the use of the scientific and technical information and of the total system analysis in the preparation of the regulatory safety case (i.e., CCA for WIPP and the license application for YM) and (2) the defense of the safety case and its technical basis within the processes established by the pertinent regulatory authorities (Bonano, Kessel and Dotson 2010).

- Comparison of PAs, especially the WIPP, YMP, SDP, and GCD programs, shows that, all else being equal, simple repository geologies tend to lead to simpler, quicker, and less costly PA programs. In addition, a simple geologic setting tends to foster more transparent communication and interaction with stakeholders. While this finding has no bearing whatever on the relative safety of repositories, it has important ramifications on the sociopolitical aspects of siting and licensing geologic repositories, and those sociopolitical aspects have proven to be as important as technical aspects of repository development.

The two most well-known series of PAs—for WIPP and for the Yucca Mountain repository—represented many years of intensely focused site-specific study, so it is may be easy to overlook the fact that some of the earliest applications of the PA methodology were not site-specific studies; rather, they were generic analyses of diverse geologic media. Indeed, the PA studies conducted by SNL in support of developing NRC's regulatory capabilities were conducted in the late 1970s and early 1980s, before Yucca Mountain had been singled out for consideration as the repository site for SNF and HLW, so the SNL PA methodology was designed for versatility. Those preliminary analyses for NRC helped formalize the methodology and demonstrate its regulatory value for mined geologic disposal, even as PA was being pioneered in a quite different application—subseabed disposal.

With Yucca Mountain no longer considered an option for disposal of SNF and HLW, a comprehensive approach will be needed to manage the analyses and evaluations of the different strategies and options that might be proposed to address the federal government responsibilities for management of radioactive waste. This new national approach toward SNF and HLW management will require development of new standards and regulations flexible enough for application to repositories in different geologic media and perhaps even different repository strategies, such as deep borehole disposal. SNL's PA methodology provides tools to carry out total system analyses to address these challenges and includes methods for propagating the effect of important sources of uncertainties in the analysis as well as for sensitivity analyses identifying the most critical system components with respect to the performance measures. Using the PA methodology will help identify technically sound nuclear waste management strategies that reduce overall cost and prioritize activities by focusing scientific and engineering efforts on what is most important to repository performance.

Generic PA Analyses of Alternative Geologic Media for Repository Options. A renewed effort to identify solutions for SNF and HLW management will be aided by the recent scoping and feasibility PA analyses for shale and granite repository disposal (Hansen, Hardin, et al. 2010, Mariner, et al. 2011) and deep borehole disposal (Brady, et al. 2009) as well as the preliminary study of feasibility for a salt repository for SNF and HLW disposal (Hansen and Leigh 2011), which performed quantitative analyses at a subsystem level and made qualitative and comparative evaluations of general repository performance. These studies demonstrate that each approach is a viable option for a renewed national program for radioactive waste management. In addition, a preliminary generic PA modeling approach has been developed to evaluate disposal options in common computational framework (Clayton, Freeze, et al. 2011). But, perhaps more importantly, these studies made important progress toward even more fundamental issues, including:

- Helping to identify issues and assumptions in regulatory standards that would necessarily be revisited in a redefined radioactive waste management policy; and

- Testing new software and computational strategies that had not been exercised previously because the licensing process of mature repository programs such as WIPP and YMP requires stability.

Advances in Computation and Modeling. In support of the DOE Office of Nuclear Energy Advanced Modeling and Simulation (NEAMS) Campaign, SNL is presently working to develop the Waste Integrated Performance and Safety Codes (IPSC). The goal is to develop an integrated suite of modeling and simulation capabilities to quantitatively assess the long-term performance of waste forms in the engineered and geologic environments of a radioactive waste storage or disposal system. The Waste IPSC will provide this simulation capability for a range of disposal concepts, waste form types, engineered repository designs, and geologic settings, and also for a range of time scales and distances. The Waste IPSC will include advanced high-performance computing capabilities (e.g., parallel processing, advanced solution techniques, high-speed supercomputers), consideration of the inherent uncertainties, and robust verification, validation, and software quality requirements. The SNL work for the Waste IPSC is documented by Freeze, Argüello and Howard, et al. (2010), Freeze, Argüello, and Bouchard, et al. (2011) and Wang, Argüello, et al. (2011).

To date, the development of radioactive waste disposal system models (both in the U.S. and internationally) has generally been limited to a moderate level of fidelity (i.e., models rely on approximations and/or surrogate representations of the thermal, hydrologic, chemical, mechanical, biological, and radiological processes and their couplings, often based on empirical relationships), each application has been focused on a very specific disposal system concept/design, and PA-model results have been focused on estimating system-level performance (e.g., dose). These moderate-fidelity PA models have typically been developed from a set of loosely linked submodels, each of which describes a very specific (and often uncoupled) process. As a result, current PA models have been relatively inflexible to changes in designs or disposal conditions and are generally validated for only that narrow range of designs and disposal conditions. The very long time scales to be considered lead to additional difficulties in validating the surrogate models and approximated couplings. As a consequence, current PA models often require significant conservatisms necessary to account for the model approximations and large uncertainties (Freeze, Argüello and Howard, et al. 2010). The high-performance computing capabilities of the Waste IPSC are intended to facilitate the application of coupled high-fidelity models to represent a variety of disposal concepts and sites while reducing the computational difficulties and uncertainties in current PA models, thereby increasing the realism represented by the results.

Wang, Argüello et al. (2011) performed a gap analysis to identify and conduct detailed analyses of candidate codes and tools to support the development and integration of the Waste IPSC. The gap analysis indicated that significant capabilities may already exist in the existing coupled thermal-hydrologic-chemical codes, but there was no single code able to fully account for all physical and chemical processes involved in a waste disposal system and large gaps existed in coupling of important repository processes. To advance code selection and code development for the Waste IPSC, Wang, Argüello et al. (2011) recommended, first, that high-fidelity, fully coupled thermal-hydrologic-chemical-mechanical-biological-radiological codes be built using SNL's existing SIERRA codes and platform, and, second, that DAKOTA be used to build an enhanced PA system framework, with a modular code architecture and key code modules for PAs.

Computational limits in PA calculations have been centrally important in the history of PA, as evidenced by the frequent references in this report to key advances and innovative approaches in addressing the computational burden associated with PA implementation (e.g., the use of Latin hypercube sampling as a means to reduce the number of Monte Carlo simulations employed to propagate the effects of parameter uncertainty though the calculations). Computer technology has so advanced in recent years that the computational burden may no longer be the limit it once was. These rapid and ongoing technological advances such as massively parallel computers as well as high-powered software such as the SIERRA codes have removed one of the key technical challenges in PA and will increase the power and utility of PA. For example, preliminary and comparative analyses of different waste disposal methods and siting alternatives as well as the efforts to understand the performance of the disposal system may include greater model detail and require less preliminary judgment or analysis on which processes or phenomena are important to performance. Use of PA as a decision-analysis tool for management to identify, prioritize, and guide areas of research for cost-effective reduction of uncertainties for objective estimation of risk (e.g., as in the WIPP SPM program) or to identify parameters for a repository performance confirmation program will be less costly in terms of time and money required for the calculations. Similar benefits will be found in the use of PA for assessment of new regulatory performance measures if they are needed for a new disposal concept such as deep borehole disposal.

Ironically, these technological advances may present new challenges to the practice of PA for compliance-directed safety assessments. Namely, high-performance computing resources and high-powered software codes tend to be in a perpetual state of improvement, while the current environment of regulatory compliance and quality assurance requires stability, repeatability, and transparency. Hardware, operating systems, software codes, and databases must be available for quality assurance audit and even review by regulators and interveners long after compliance calculations are run. To utilize advancing computational technologies in future PAs in assessing regulatory compliance of nuclear waste disposal systems will require not only the technical know-how but also the practical understanding and experience of the regulatory environment, quality assurance principles and requirements, and stakeholder interests and concerns. If they can be developed, regulatory structures and quality assurance approaches that anticipates and accommodates rapid evolution and improvement in the computational approaches applied in PA will optimize the benefits produced by computational advancement.

Expanding the PA Methodology to New Applications. Such computational advances could contribute to further development of other enhanced PA systems such as the one outlined by Wang, Dewers, et al. (2010) and described in Section 8.7 of this report, which is designed for real-time optimization of carbon storage and sequestration system for safety and effectiveness, as well as the enhanced PA system developed by Ames et al. (2010) for optimization modeling of nuclear energy fuel cycles for the efficient use of uranium resources and minimizing radioactive waste products. Both of these studies were developed under SNL's Laboratory Directed Research and Development program, the discretionary research and development investment program at SNL established to serve as a proving ground for new concepts, to support high-risk but potentially high-value research and development, and to foster creativity and stimulate exploration at the forefront of science and technology.

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