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Historical Background on Assessing the Performance of the Waste Isolation Pilot Plant

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Abstract

In 1979, six years after selecting the Delaware Basin as a potential disposal area, Congress authorized the U.S. Department of Energy to build the Waste Isolation Pilot Plant (WIPP) near Carlsbad, New Mexico, as a research and development facility for the safe management, storage, and disposal of waste contaminated with transuranic radioisotopes. In 1998, 19 years after authorization and 25 years after site selection, the U.S. Environmental Protection Agency (EPA) certified that the WIPP disposal system complied with its regulations. The EPA's decision was primarily based on the results from a performance assessment conducted in 1996. This performance assessment was the culmination of four preliminary performance assessments conducted between 1989 and 1992. This report provides a historical setting and context for how the performance of the deep geologic repository at the WIPP was analyzed. Also included is background on political forces acting on the project. For example, the federal requirement to provide environmental impact statements and negotiated agreements with the State of New Mexico influenced the type of scientific areas that were investigated and the engineering analysis prior to 1989 for the WIPP.

Acknowledgments

This report is a greatly expanded version of a figure showing milestones originally prepared for a report on the performance assessment process (Rechard, 1995). The figure was updated and presented separately in 1998 (Rechard, 1998). The original figure included technical milestones, but emphasized the political and policy turmoil acting upon the WIPP Project. Although the information from that figure is included here, the emphasis in this report is on the technical aspects of the project. The WIPP has benefited from continuity in the technical support provided by Sandia National Laboratories (in addition to the strong community support by Carlsbad and management support by Westinghouse); thus, more technical information has been added and the information on the original figure has been rearranged into several different figures covering specific technical aspects. The usefulness of this history is due in part to D. E. Munson, M. S. Tierney, and W. D. Weart, of Sandia, who reviewed the text; D. Brewer of Sandia, who compiled the WIPP budget history; F. C. Allan and S. Halliday, librarians at Sandia who helped with searches; C. S. Crawford, of ASAP, Inc., who verified references; and Tech Reps, Inc. personnel, J. M. Chapman, who edited the text, S. K. Best, who generated the figures, and L. C. Tartaglia, who provided desktop publishing.

Contents

1 Introduction 1
1.1 Characterization of WIPP Disposal System 1
1.2 Compliance Assessment of WIPP 2
1.3 Organization of Historical Material 3

2 Performance Goals for Waste Isolation Pilot Plant 5
2.1 Criteria for Selecting a Site 5
2.1.1 Early Disposal Methods 5
2.1.2 Selection of Bedded Salt 5
2.1.3 Selection of Repository Location 6
2.1.4 Segregation of TRU Waste 7
2.2 Preparation of Environmental Impact Statement 7
2.3 Establishment of Regulatory Risk Goals 9
2.3.1 Historical Events 9
2.3.2 Containment Requirements of 40 CFR 191 10

3 Site Characterization 12
3.1 Selection of Site 12
3.2 Site Characterization Studies for EIS 13
3.3 Long-Term Site Characterization Studies 14
3.3.1 Site Characterization Studies at Repository Horizon 14
3.3.2 Site Characterization Studies for Stipulated Agreement 16
3.4 Site Characterization Studies for CCA 18
3.4.1 Fluid Flow in Culebra 18
3.4.2 Sorption Studies in Culebra 18

4 Waste Properties 20
4.1 Changes in Purpose of WIPP Project 20
4.2 Transuranic Nuclear Waste 20
4.2.1 Current Waste Description 20
4.2.2 Modification of TRU Waste Definition 22
4.2.3 TRU Waste Inventory 22
4.3 Waste Characterization Studies 22
4.3.1 Initial EIS Waste Studies 22
4.3.2 Container Corrosion Studies 23
4.3.3 Characterization for Performance Assessment Studies 23

5 Design, Construction, and Behavior of Repository 26
5.1 Design of Repository 26
5.1.1 Early Conceptual Design 26
5.1.2 Detailed Design 26
5.2 Construction of Repository 28
5.3 Experiments and Model Development of Repository Behavior 29
5.3.1 Experiments with Salt Creep Behavior 29
5.3.2 Modeling of Repository Behavior 30
5.4 Repository Backfill 30
5.4.1 Use of Backfill in Disposal Area 30
5.4.2 Shaft Backfill and Sealing Methods 31
5.4.3 Panel Sealing Methods 31

6 Hazard Identification and Scenario Development 32
6.1 Description of Hazard Identification 32
6.1.1 Categories of Hazardous Agents 32
6.1.2 Scenarios 32
6.1.3 Steps of Hazard Identification and Scenario Development 32
6.2 Hazard Identification After Site Selection 33
6.2.1 Features, Events, and Processes Considered 33
6.2.2 Screening Calculations 33
6.3 Hazard Identification and Scenario Development for EIS 33
6.3.1 Features, Events, and Processes Considered 33
6.3.2 Construction of Scenarios 34
6.3.3 Hazards Related to Waste Acceptance Criteria 34
6.4 General Scenario Development 34
6.4.1 Defining Universe (FEPs List) 35
6.4.2 Screening Universe and Forming Scenarios 35
6.4.3 Including the Human Intrusion Event 35
6.5 Hazard Identification for Preliminary PAs 35
6.5.1 Hazards Identified in 1980s 36
6.5.2 Features, Events, and Processes Retained 36
6.5.3 Undisturbed Scenario 36
6.5.4 Human Intrusion Scenarios 36
6.6 Hazard Identification for 1996 PA 39

7 Consequence Analysis 40
7.1 Description of Consequence Analysis for the WIPP 40
7.2 Analysis for Site Selection and EIS 40
7.2.1 Site Selection Analysis 40
7.2.2 EIS Analysis 41
7.2.3 NRC Analysis 41
7.2.4 Supplemental EIS Analysis 42
7.3 Analysis Logistics for PA 43
7.3.1 Iteration of Calculations 44
7.3.2 Detailed Modeling Style 44
7.3.3 CAMCON Development for PA 44
7.4 Consequence Models in Performance Assessments 45
7.4.1 Direct Cutting and Brine Release Modeling 47
7.4.2 Repository Fluid Flow Modeling 47
7.4.3 Source Term Modeling 48
7.4.4 Culebra Transmissivity Modeling 49
7.4.5 Culebra Flow and Transport 50

8 Probability Evaluation and Sensitivity Analysis 51
8.1 Scenario Probabilities and Parameter Selection for the EIS 51
8.1.1 Scenario Probabilities 51
8.1.2 Parameter Selection 51
8.2 Scenario Probabilities for Performance Assessments 51
8.3 Parameter Uncertainty in Performance Assessments 52
8.3.1 Data Bases for Model Parameters and Results 52
8.3.2 Quality Assurance Procedures 52
8.3.3 Parameter Selection 53
8.3.4 Describing Parameter Uncertainty 53
8.4 Sensitivity Analysis 54
8.4.1 LHS Technique 54

8.4.2 Sensitive Parameters 55
8.4.3 Project Guidance from Sensitivity Analyses 55

9 Compliance Assessment and Summary 57
9.1 Assessing Compliance of the WIPP 57
9.1.1 Predicted Doses for EIS 57
9.1.2 Releases from Preliminary Performance Assessments 57
9.1.3 Performance Assessment for CCA 57
9.2 Summary 58

References 61

Figures

1-1 Juxtaposition of several categories of events for the WIPP 4
2-1 Early history of nuclear waste disposal related to the WIPP 6
2-2 Performance goals were added or changed several times over the 25-yr history of the WIPP Project 8
2-3 Uncertainty in model parameters p (expressed as a probability density function [PDF], or cumulative distribution function [CDF]) results in uncertainty in the results 10
3-1 Physical setting of the WIPP in Delaware Basin 13
3-2 Well tests conducted to characterize transmissivities of the Culebra Dolomite Member of the Rustler Formation 14
3-3 Stratigraphy above and below the WIPP repository 15
3-4 Estimated permeability of Salado Formation near repository horizon continued to decrease over the years 16
3-5 Site characterization studies at the WIPP 17
3-6 Stratigraphic subdivisions of Culebra Dolomite Member of Rustler Formation discerned after 1997 18
3-7 Ranges of K_ds (m^3/kg) for plutonium and uranium in 1991, 1992, and 1996 performance assessments 19
4-1 Development of project goals and characterization of waste properties 21
4-2 Solubility of radioisotopes used in calculations between 1989 and 1996 for (a) plutonium and (b) uranium 25
5-1 Designs for the WIPP repository, including (a) 1977 conceptual design and (b) final construction plans for WIPP repository 27
5-2 Events associated with design, construction, and modeling of the WIPP repository 28
5-3 Comparison of measured and calculated creep closure in Room D 30
6-1 Technique to construct scenarios from various features, events, and processes changed between WIPP Environmental Impact Statement in 1979 and performance assessment calculations after 1989 34
6-2 Engineering analysis of the illustrated behavior of the disposal system 37
6-3 Undisturbed scenario used since 1989 38
6-4 E1E2 human intrusion scenario used since 1990. Intrusion borehole plugs are placed to maximize brine flow through the repository 38
7-1 Boundaries of models constructed for analysis of groundwater flow and transport of radioisotopes in the Culebra at the WIPP 42
7-2 Schematic of CAMCON modeling system completed in 1991. By 1996, CAMCONexec was replaced by computer specialists for run control and management of files 46
7-3 Changes in components of the exposure pathway ("consequence") model for immediate releases after exploratory drilling for evaluating compliance of the WIPP 47
7-4 Changes in components of the exposure pathway model for long-term releases via leaking borehole and Culebra brine aquifer 48
9-1 Changes in the distribution of cumulative releases (normalized by the EPA limits) for (a) direct releases to the surface during drilling and (b) releases via groundwater pathway through Culebra 58
9-2 Annual outlays for construction and assessments of the Waste Isolation Pilot Plant (WIPP) (private communication with D. Brewer, SNL, from Sandia budget records) 60
9-3 History of expenditures and human resources for WIPP Project at Sandia National Laboratories 60

1.0 Introduction

In 1998, 25 years after selection of the Delaware Basin in southeastern New Mexico as a potential disposal site for radioactive waste, the U.S. Environmental Protection Agency (EPA) certified that the Waste Isolation Pilot Plant (WIPP), an underground geologic disposal system, complied with its regulations. This report presents a historical summary of the system characterization, engineering analyses, and scientific investigations undertaken by the WIPP Project over the past 25 years. Many paths were followed to discern which phenomena were important at the WIPP, and often these paths were initiated in response to evolving notions of what kind of scientific information was significant, what level of understanding was required, and how society could use this scientific information to decide whether deep geologic disposal of nuclear waste was acceptable.

1.1 Characterization of WIPP Disposal System

The choice of New Mexico as a potential disposal site for nuclear waste in 1973 (Claiborne and Gera, 1974) was one in a series of episodes in which New Mexico had figured prominently with regard to nuclear phenomena. In 1942, the Manhattan Engineering District selected New Mexico for assembling the scientists, engineers, and technicians who would develop the first atomic bomb. The location selected later became what is now known as Los Alamos National Laboratory and Sandia National Laboratories[1]. As a direct consequence of that development, the first atomic explosion took place in 1945 in the desert near Alamogordo, New Mexico (Serber, 1992). Sixteen years later, in 1961, scientists with the Gnome Project detonated a device in the Delaware Basin in bedded salt near Carlsbad, New Mexico, as part of the Plowshare Program, which was exploring nonmilitary uses of nuclear explosives (Teller, 1959; Gard, 1968). Several months after the explosion, as part of the test, engineers made an excavation in the cavity that had been created.

After a potential disposal site in bedded salt in Kansas had been rejected in the early 1970s, New Mexico citizens invited the Atomic Energy Commission (precursor to the U.S. Department of Energy[2]) to consider the bedded salt deposits in the Delaware Basin in southeastern New Mexico. The search for a specific site in the basin occurred between 1973 and 1976 (Powers et al., eds., 1978). The site was then characterized by Sandia National Laboratories for the U.S. Department of Energy (DOE) in support of its Draft Environmental Impact Statement (EIS), which was completed in 1979 (DOE, 1979).

Nearly ten years later, in December 1979, Congress authorized the DOE to construct the WIPP for eventual disposal of radioactive waste (Public Law 96-164). Shortly before October 1979, Bechtel National had begun designing the surface and underground facilities. The first shaft was drilled in 1981, and full construction began in 1983. Construction of the WIPP facility was substantially complete at the end of 1988.

The wastes intended for the WIPP included waste contaminated with transuranic (TRU) nuclear elements and hazardous chemicals generated during the production of nuclear weapons. The Atomic Energy Commission (AEC) has been segregating and storing these wastes above ground for eventual disposal since 1970 (Perge, 1982).

Along with construction, a suite of characterization studies were initiated in the 1980s in response to agreements with the State of New Mexico. The final site characterization studies were completed in the 1990s. The characterization of the WIPP provided input for the 1996 performance assessment, an engineering analysis documented in the Compliance Certification Application (CCA), which was submitted to the EPA in October 1996 (DOE, 1996a; EPA, 1996b).

1 Los Alamos National Laboratory, a multiprogram laboratory in Los Alamos, NM, was first known informally as Los Alamos Laboratory; it was officially named the Los Alamos Scientific Laboratory in 1948 and then renamed the Los Alamos National Laboratory in 1979. Sandia National Laboratories is the multiprogram laboratory located in Albuquerque, NM, and Livermore, CA. The Albuquerque laboratory was originally referred to as Los Alamos' Z Division, and then as the Sandia branch of Los Alamos. It became Sandia Laboratories in 1949 and Sandia National Laboratories in 1979.

2 The Atomic Energy Commission (AEC) was formed by the *Atomic Energy Act of 1946* (Public Law 79-585). The Energy Research and Development Agency (ERDA) and the Nuclear Regulatory Commission (NRC) were formed by splitting the Atomic Energy Commission in the *Energy Reorganization Act of 1974* (Public Law 93-438). ERDA became the U.S. Department of Energy (DOE) in the 1977 *Department of Energy Organization Act* (Public Law 95-91).

1.2 Compliance Assessment of WIPP

Over the past 25 years, the process for assessing performance of a deep geologic repository for radioactive waste developed concurrently with the characterization of the WIPP in New Mexico. The WIPP Project's first major analysis was for the EIS in 1979 (DOE, 1979). In 1985, the EPA[3] promulgated its radiation protection standard for the management and disposal of spent nuclear fuel, high-level and transuranic wastes, in Title 40 of the Code of Federal Regulations Part 191 (40 CFR 191) (EPA, 1985a). In 1986, the DOE asked Sandia to assess the performance of the WIPP (Krenz, 1986). The assessment process that evolved by 1996 for this regulation included developing a scientific understanding of the current status of the repository and the surrounding geologic barrier (the disposal system) through sufficient site characterization. Yet, an important aspect of the regulatory assessment was a set of calculations illustrating possible behavior well into the future. The assessment also required that the calculations include uncertainty concerning model parameters and model form; hence, the analysis was probabilistic.

The overall process of assessing whether a nuclear waste disposal system meets a set of performance criteria is known as a performance assessment, a term defined in 40 CFR 191. Similar to other risk assessments, the performance assessment process consists of determining the answers to the following three questions (Kaplan and Garrick, 1981; Helton et al., 1997c, 1993b; Rechard, 1995):

1. What hazards can occur?
   Process: Identify sources of unwanted outcomes through hazard identification and scenario development.

2. What are the consequences potentially caused by these hazards?
   Process: Evaluate the consequences or unwanted outcomes by determining
   - the pathway by which a hazard reaches a receptor (e.g., humans or the general environment) and
   - the response (e.g., fatality, injury, or no effect) to the level of hazard that eventually reaches the receptor

3. What is the probability of these unwanted outcomes?
   Process: Evaluate the probability of consequences by determining the probabilistic description of the uncertainty in both the pathway to the receptor and the response of the receptor.

Describing and quantifying the answers to these three basic questions are the three main steps of a performance assessment. Four other steps complement them. First, if performance of a system is being evaluated for the first time, then an initial step must determine appropriate performance measures. Then, the system must be characterized or otherwise defined. After probabilities and consequences have been calculated, a separate step is to combine them for use as input to management decisions. In the case of a compliance assessment for a nuclear waste repository, the results are compared with probabilistic risk criteria so that a decision can be made on whether society will accept the nuclear waste disposal site. Finally, a sensitivity analysis on the model parameters is run, if appropriate, to identify significant parameters for use by decision makers. In summary, a performance assessment includes up to seven steps:

0. Definition of performance criteria
1. System definition and/or characterization
2. Hazard identification and scenario development
3. Probability evaluation
4. Consequence evaluation
5. Performance characterization and compliance assessment
6. Sensitivity analysis

As with any scientific modeling or policy process, steps may overlap. More importantly, an analyst may need to cycle through several of the steps when building an appropriate model. Hence, the steps are not always truly sequential. However, the discretization is useful as a means of describing the process and so is used here. The computational mechanics of the seven-step process are described in Rechard (1995).

Because the main purpose of a performance assessment is to serve as input to a management decision, it is an engineering analysis with constraints on time and resources specified by the decision makers (or tolerated by representatives of society) rather than a scientific

3 Congress formed the U.S. Environmental Protection Agency in 1970, transferring to it from other agencies the responsibilities of research, monitoring, standard setting, and enforcement activities related to the environment.

analysis, which is in principle constrained only by human curiosity.

1.3 Organization of Historical Material

The historical events summarized above are more fully described in the following sections according to the steps of a performance assessment. The discussion begins with the definition of performance goals. Next discussed is characterization of the three main components of the disposal system, i.e., site, waste and facility. Site characterization for the WIPP (Section 3) includes identifying and selecting a specific site and studying its geologic and hydrologic setting. The waste characterization studies (Section 4) describe the amounts and types of waste intended for the WIPP, and examining its various properties. Facility design studies (Section 5) evaluate the thermal/mechanical properties of the salt and also various engineering components such as backfill and shaft sealing. The engineering analysis of the WIPP disposal system is then presented according to the remaining steps of a performance assessment (Figure 1-1).

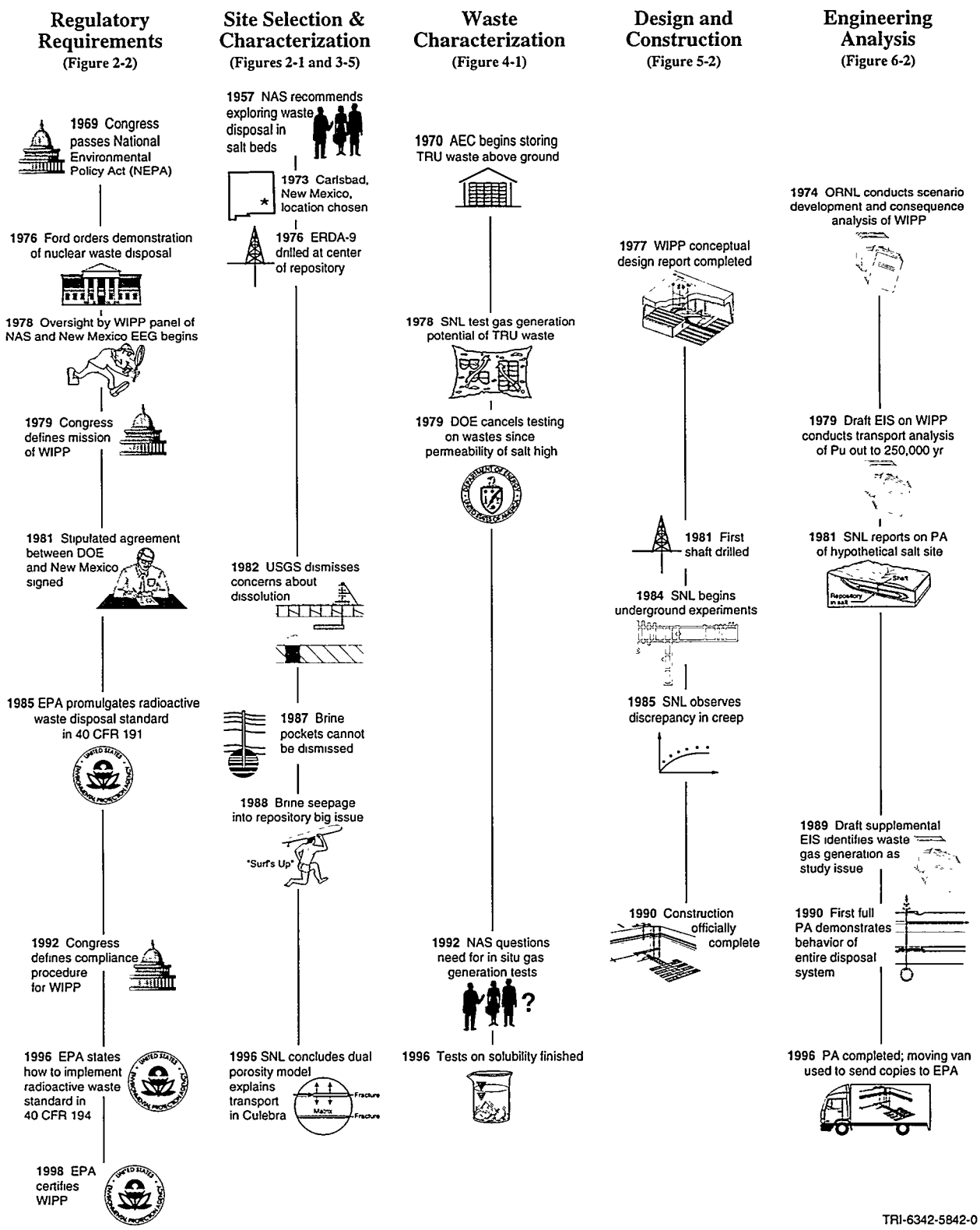

Figure 1-1. Juxtaposition of several categories of events for the WIPP.

2.0 Performance Goals for Waste Isolation Pilot Plant

From 1955 through the 1960s, the AEC explored options for storage and disposal of nuclear waste. The scientific and engineering studies for deep geologic disposal in salt eventually led to the selection of the salt beds of New Mexico. Later, in the mid 1970s through the early 1980s, an assessment process and risk-based performance criteria evolved to determine the acceptability of the risk of the WIPP disposal system.

2.1 Criteria for Selecting a Site

2.1.1 Early Disposal Methods

In the early years of nuclear research, it was very important to recover all of the radioisotopes produced, particularly ^{239}Pu and ^{235}U. At that time, many members of the staff at various facilities were analytical chemists whose primary task was to detect and retain isotopes (Perge, 1982). At Los Alamos National Laboratory, the site and methods for disposal of waste that remained were initially the same as for any other waste: it was disposed of in nearby canyons. Around 1944, the Manhattan Engineering District decided to bury solid nuclear waste in shallow trenches and augered holes at Los Alamos (NAS/NRC, 1957) and in railroad cars, trenches, and underground caissons at the Hanford Reservation in Washington state.

The AEC, formed in 1946 (Public Law 79-585), continued the practices of the Manhattan Engineering District. The AEC also constructed storage tanks in the late 1940s at Hanford and completed a nuclear waste storage complex, the Radioactive Waste Management Complex (RWMC) at Idaho National Engineering and Environmental Laboratory[1] (INEEL) in 1952. By the mid-1950s, eight reactors were operating at Hanford and five were running at the Savannah River Plant, decreasing the need to recover every detected gram of plutonium. Furthermore, in the late 1950s and early 1960s, the AEC began to refurbish, modify, and clean up its facilities, e.g., Los Alamos began to replace its temporary wooden buildings, and so the amount of plutonium-contaminated waste in need of disposal rapidly increased. Initially, waste had been kept near the facilities and test sites where it was produced (by the mid-1950s Hanford had 30 burial grounds [Perge, 1982]). However, in the 1960s the AEC did try to reduce the number of burial grounds at the facilities to help manage the wastes.

2.1.2 Selection of Bedded Salt

Studies of permanent disposal options began in 1955 when the AEC asked the National Academy of Sciences (NAS) to examine the disposal issue. In 1957, the NAS reported that while various options and disposal sites were feasible, disposal in salt was the most promising method to explore (NAS/NRC, 1957). NAS reaffirmed that recommendation in 1961 (Carter, 1987, p. 64; Boffey, 1975; Claiborne and Gera, 1974). By 1966, frustration at the lack of a formal waste policy at the AEC provoked strong criticism from the NAS about the AEC's disposal practices (Carter, 1987, p. 64; Boffey, 1975; Claiborne and Gera, 1974). At that time, the reasons for using salt beds were as follows: (1) salt can be found in regions of tectonic stability, (2) the existence of salt demonstrates the absence of fresh circulating groundwater, (3) salt is easy to mine, and (4) fractures are readily healed (i.e., salt readily consolidates and entombs the waste as the result of its plastic properties). From 1961 through the early 1970s, Oak Ridge National Laboratory[2] (ORNL) conducted radioactive-waste disposal experiments, most notably Project Salt Vault in an abandoned salt mine near Lyons, Kansas, from 1963 to 1967. Some of the experiments had used actual spent nuclear fuel, which was retrieved afterwards and sent back to INEEL (Bradshaw and McClain, eds., 1971) (Figure 2-1). Based on the results of these tests, the Committee on Radioactive Waste Management, which had been established in 1968 at the request of the AEC and later became a permanent board, concluded in their 1970 report that bedded salt was satisfactory and the safest choice then available for nuclear waste disposal (U.S. Congress, 1970a; 1970b; NAS/NRC, 1970).

1 Idaho National Engineering and Environmental Laboratory (INEEL), a multiprogram laboratory in Idaho Falls, Idaho, originated as the National Reactor Testing Station in 1949. It was named the Idaho National Engineering Laboratory in 1974, and then renamed the Idaho National Engineering and Environmental Laboratory in early 1997.

2 Oak Ridge National Laboratory, a multiprogram laboratory in Oak Ridge, Tennessee, was originally named the Clinton Engineering Works in 1942. It was renamed the Clinton National Laboratory in 1947, and the Oak Ridge National Laboratory in 1948.

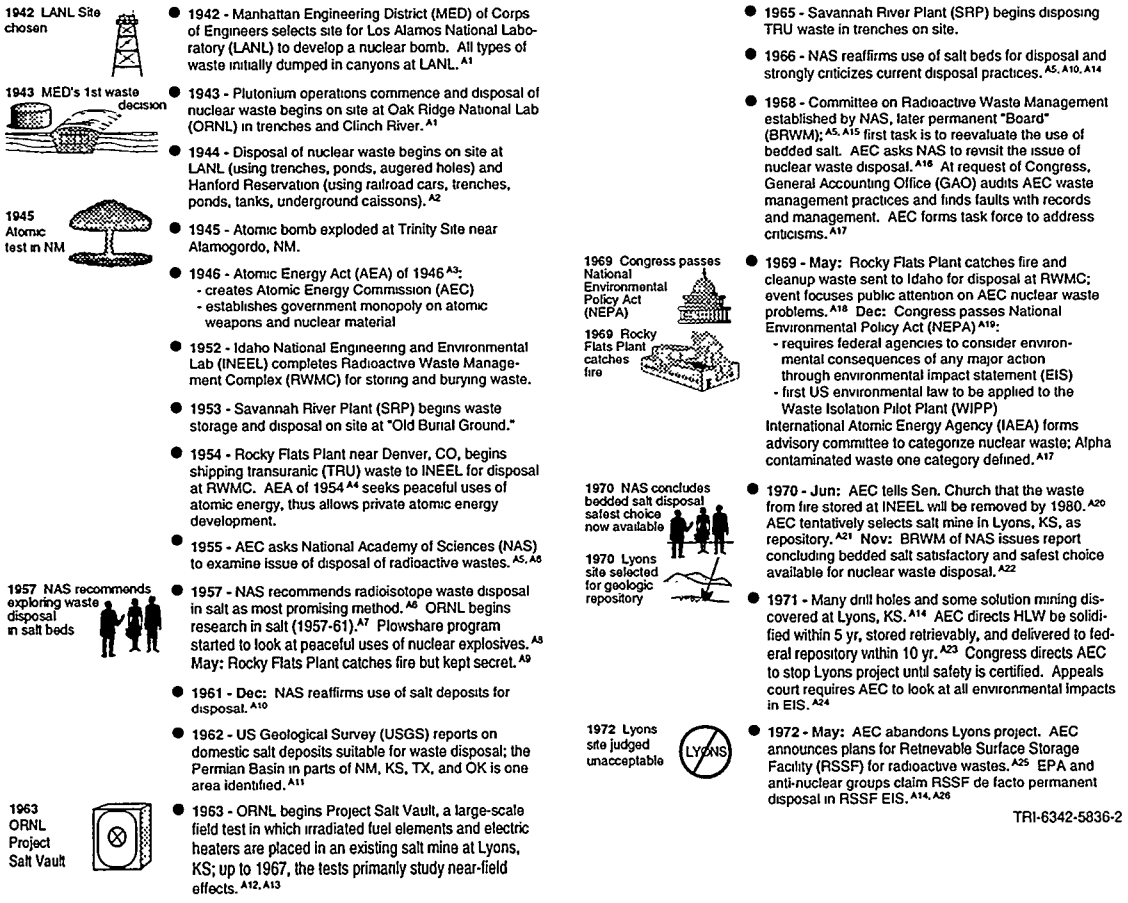

Figure 2-1. Early history of nuclear waste disposal related to the WIPP.

2.1.3 Selection of Repository Location

In May 1969, the Rocky Flats Plant, built by the AEC in 1951 to machine plutonium and other metals for nuclear weapons, caught fire. Located only 26 km (16 mi) from Denver, Colorado, the fire and subsequent cleanup attracted public attention. The press reported that the waste from the cleanup was eventually to be sent to Idaho (Carter, 1987, p. 66). For the first time, the public and many Idaho state officials learned that TRU waste from the Rocky Flats Plant had been routinely buried in Idaho since 1954. Media attention over the events created high visibility for the issue of waste disposal from nuclear weapons production. Officials of the State of Idaho complained that the state was becoming the nation's waste disposal site for defense-related waste by default; in addition, the Radioactive Waste Management Complex there was not ideal for long-term disposal of nuclear waste because of its location near the Snake River and its associated aquifer. These problems were described in reports to Senator Church of Idaho by the U.S. Geological Survey (USGS), the U.S. Public Health Service, the Federal Water Quality Commission (later part of the EPA), and the Bureau of Sport Fisheries and Wildlife (Perge, 1982); hence, the AEC quickly moved to find a more suitable site. The AEC assured Senator Church that the waste from Rocky Flats stored in Idaho would be removed by 1980 (Lipschutz, 1980). In June 1970, the AEC tentatively selected the abandoned Kansas salt mine at Lyons, the site of the ORNL experiments, as a demonstration repository (AEC, 1971).

Although salt has many advantages, a disadvantage is the coexistence of economic minerals, which are deposited along with the salts, and hydrocarbons. In 1971, a large number of previously unknown drill holes for mineral exploration and some solution mining were discovered very near the proposed repository (Carter, 1987, p. 69). Already faced with general opposition to the site from Kansas state officials and anti-nuclear groups, the AEC now had a technical reason to look for a new site. Soon after, Congress directed the AEC to stop work on the Lyons project until safety was certified.

In May 1972, the AEC officially abandoned the Lyons project and announced plans for a Retrievable Surface Storage Facility primarily for high-level waste (Metlay, 1978); however, anti-nuclear groups and the EPA, through comments on the draft EIS for the facility, claimed the storage facility as *de facto* permanent disposal (MRSRC, 1989). Although the RSSF plan was not officially withdrawn until formation of ERDA in January 1975, the early criticism prompted the AEC to continue to search for a suitable disposal site for the less controversial TRU waste (Anderson et al., 1973; Jones et al., 1973, Carter, 1987, p. 177; Brokaw et al., 1972; Mytton, 1973; Bachman et al., 1973; Merewether et al., 1973; Hite and Lohman, 1973; Bachman, 1973). At the time, several states offered to host the TRU waste in other abandoned mines (Perge, 1982). New Mexico citizens, with the tacit approval of Governor Bruce King, invited the AEC to consider the salt beds of southeastern New Mexico. Based on previous experience with the Gnome project, ORNL and the USGS were able to recommend to the AEC the extensive salt beds of the Delaware Basin (Barnes, 1974).

The experience at Lyons provided two important site-selection criteria: (1) an absence of boreholes or solution mining near the repository, which meant ensuring minimal conflicts with other mineral resources by means of a buffer zone of one mile (ORNL originally used two miles) from existing deep wells, five miles from existing potash mines, and avoidance of known hydrocarbons and potash reserves, and (2) the presence of advantageous political characteristics such as public and government support in the area, low population density, and few or no land use conflicts. These two criteria were added to other criteria such as presence of high quality salt at a depth between 300 and 800 m to avoid potential problems with erosion, dissolution, or rapid salt creep.

2.1.4 Segregation of TRU Waste

In 1968, the Government Accounting Office (GAO) evaluated waste management practices within the AEC (Perge, 1982). Like the 1966 NAS Committee, the GAO criticized the documentation and organizational aspects of the AEC's disposal practices, especially for high-level waste. In response, the AEC formed a management task force that eventually recommended that liquid high-level waste be solidified and that low level waste, and what was then called plutonium-contaminated waste (now TRU waste), be studied further. The AEC promulgated regulations (10 CFR 50) that directed that high-level waste be solidified within five years, stored retrievably at all DOE facilities, and delivered to a federal repository within 10 years (AEC, 1970).

Several events created a situation in which it became important to define TRU waste. First, after planes carrying nuclear weapons had crashed in Spain in 1966 and Greenland, concerns were raised about how much contaminated soil and ice should be returned to the United States. Second, the International Atomic Energy Agency (IAEA) formed an advisory committee to develop categories of radioactive waste, one of which was "Alpha contaminated waste." Finally, the 1969 fire at the Rocky Flats Plant near Denver, Colorado, increased concern about this waste. Then the AEC's decision to accelerate the Lyons, Kansas, repository specifically for this waste became an impetus for defining this waste category.

The AEC defined TRU waste using a bounding definition: any contaminated material with an activity density of greater than 10 nCi/g (which is about the activity density of ^{226}Ra in the earth's crust). The AEC's definition of TRU waste purposely used mass as its basis to avoid the issue of dilution, in contrast to the volume basis used by the IAEA (Perge, 1982). Also, in March 1970, the AEC directed that TRU waste be stored so that it could be retrieved, rather than disposed of in trenches with low-level waste (Hollingsworth, 1970). Thereafter, it was stored on the surface in Idaho and elsewhere.

2.2 Preparation of Environmental Impact Statement

The National Environmental Policy Act of 1969 (NEPA, Public Law 91-190), signed in January 1970 by President Nixon, and its implementing regulations 40 CFR 1500-1508 required federal agencies to consider the environmental consequences of any major action through an EIS (environmental impact statement). NEPA was the first environmental statute applied to the WIPP.

Although NEPA required that federal agencies prepare an EIS, it did not provide specifics regarding content nor did it list criteria for making decisions on the acceptability of a project. Hence, during the 1970s, the courts, the executive branch, and federal agencies wrestled with the proper scope and extent of an EIS. For example, in response to NEPA, the AEC prepared an EIS on the Calvert Cliffs reactor that discussed only direct environmental impacts, but was quickly sued by citizen groups opposed to its construction (the Calvert Cliffs Coordinating Committee) because they, among other objections, considered its focus too narrow. The

1969 Congress passes National Environmental Policy Act (NEPA)

- 1969 - National Environmental Policy Act (NEPA)[B1]:
  - requires federal agencies to consider environmental consequences of any major action through environmental impact statement (EIS)
  - first environmental law to be applied to the WIPP

1970 Congress forms EPA

- 1970 - Congress forms Environmental Protection Agency (EPA) and transfers to it research, monitoring, standard setting, and enforcement activities.[B2 B3]

- 1974 - AEC promises Idaho that wastes will be shipped out of Idaho in the 1980's. **Oct:** Energy Reorganization Act[B4] splits AEC into Nuclear Regulatory Commission (NRC) and Energy Research and Development Agency (ERDA) effective January 1975.

1976 Bishop Lodge Conference to explore PRA for geologic disposal

1976 Ford orders demonstration of nuclear waste disposal

- 1976 - ERDA funds conference on modeling of geologic disposal systems to bring engineers and geologists together to explore predicting geological features, events, and processes (FEPs).[B5] **Oct:** President Ford orders expansion of ERDA program to demonstrate disposal for nuclear waste by 1985 and orders EPA to develop applicable standards.[B6, B7] Resource, Conservation and Recovery Act (RCRA)[B8] seeks to reduce hazardous waste generation. **Dec:** EPA announces intent to develop standards for HLW disposal.[B9]

1977 DOE created

- 1977 - **Oct:** DOE Organization Act[B10] creates Department of Energy (DOE) from ERDA. **Feb:** EPA conducts 1st public workshop to understand public concerns and technical issues of waste disposal.[B6, B11]

1978 Oversight by WIPP panel of NAS and New Mexico EEG begins

- 1978 - DOE Deputy Sec. Jerry O'Leary promises NM Congressional delegation "if NM did not wish to have the WIPP, then it could veto the plan." DOE General Counsels state O'Leary powerless to grant "state veto."[B12] **Jan:** EPA announces public forum to develop protection criteria for radioactive wastes.[B11] **Jun:** In response to DOE request to review scientific aspects of WIPP Project, WIPP Panel of BRWM of NAS holds first meeting.[B13] **Nov:** EPA publishes "Criteria for Radioactive Wastes" as guidance for federal agencies and seeks comments.[B14] DOE contracts with NM to establish Environmental Evaluation Group (EEG) to provide an independent state assessment of WIPP.

1979 Congress defines mission of WIPP

- 1979 - **Dec:** Congress defines mission[B15] of WIPP:
  - sets up WIPP as a research and development facility for disposal of only DOE TRU waste
  - exempts WIPP from NRC licensing
  - requires DOE to sign a "Consultation & Cooperation" (C&C) Agreement with NM.

- 1980 - NM and DOE begin negotiations on C&C Agreement to define procedures and process of cooperation

1981 Stipulated agreement between DOE and New Mexico signed

- 1981 - **Jan:** DOE decides to proceed with Site and Preliminary Design Validation (SPDV). In response:
  - **Mar:** Citizens for Alternatives to Radioactive Dumping (CARD) asks for preliminary injunction.[B16 B17]
  - **May:** NM sues DOE and Department of Interior (DOI) alleging violations of federal and state laws.[B17]
  - **Jul:** Southwest Research & Information Center (SWRIC) files lawsuit[B17, B18] and begins filing numerous interrogatories to which DOE must respond.

  In response to lawsuits,[B17] DOE Sec. Edwards visits NM to talk to Gov. King and accedes in a "Stipulated Agreement (SA)" for (1) geotechnical experiments, (2) SNL report on 17 technical issues, (3) state & public review of WIPP changes, and (4) creation of a state/federal task force to oversee transportation issues (e.g., emergency response and highway upgrades). C&C Agreement attached as Appendix A; "Working Agreement" as Appendix B.[B17] U.S. Dist. Judge Burciaga stays lawsuits. **Mar:** Developing generic disposal criteria for radioactive wastes difficult, thus EPA starts developing standards for each waste type.[B19]

- 1982 - EPA publishes working draft 20 as proposed 40 CFR 191.[B20] **Dec:** Supplemental SA signed committing DOE (1) to seek funds for upgrading highways in NM, (2) to more geotechnical studies, and (3) liability for WIPP-related accidents.[B21]

- 1983 - In response to questions by EEG, DOE concludes draft 40 CFR 191 applies to disposal phase but not test phase of WIPP. **May:** Oak Ridge complex admits releasing 2 x 10^6 lb of Hg from Y-12 plant between 1950 and 1977. Revelation prompts Natural Resources Defense Council (NRDC) and Legal Environmental Assistance Foundation (LEAF) to sue DOE.

- 1984 - **Feb:** EPA Scientific Advisory Board (SAB) endorses probabilistic approach of 40 CFR 191 but states criteria too restrictive and recommends changes.[B22] **Apr:** LEAF vs. Hodel[B23] subjects DOE to RCRA requirements though AEA exempted DOE from human health laws.

- 1984 - (con't) **Nov:** Hazardous & Solid Waste Amendments (HSWA)[B24] to RCRA ban land disposal of hazardous waste without treatment unless disposal site and generator demonstrate "no migration" of constituents for as long as waste remains hazardous. 1st modification to C&C Agreement limiting remote handled (RH) TRU waste amount to 5.1 x 10^6 Ci.

1985 EPA promulgates 40 CFR 191

- 1985 - **Feb:** NRDC sues EPA to issue 40 CFR 191 as mandated in NWPA.[B25] **Sep:** EPA promulgates 40 CFR 191 for disposal of SNF, HLW, and TRU[B6]:
  - probabilistic criteria indirectly based on population health risk
  - requires inclusion of all uncertainty

  Nov: DOE attempts to define "by-product material" to include mixed waste and thus exclude EPA regulation.

1986 EPA states mixed waste subject to RCRA (potentially ~60% of WIPP waste)

- 1986 - EPA states that mixed waste (radioactive waste with hazardous waste) is subject to RCRA regulations.[B26] **Mar:** NRDC and others sue EPA over groundwater and individual protection standards in 40 CFR 191.

- 1987 - **May:** DOE defines "by-product material" to exclude everything except radionuclides, and thereby TRU waste is subject to RCRA.[B27] **Jul:** In response to NRDC lawsuit, Court of Appeals for 1st Circuit in Boston vacates and remands all of 40 CFR 191.[B25] **Aug:** 2nd modification to C&C Agreement commits DOE to comply with all applicable laws; to use 40 CFR 191 as 1st issued for evaluating WIPP compliance until reissued by EPA; and apply NRC and Department of Transportation (DOT) regulations to TRU transport. **Sep:** Court reinstates Subpart A of 40 CFR 191 in response to EPA request.

- 1988 - **Oct:** ID Gov. Andrus bans shipments of radioactive waste into state because WIPP not open. With continued technical problems (e.g., TRUPACT-II not yet licensed), NM Congressional delegation cannot get consensus among themselves and WIPP Land Withdrawal Act dies. Congressman Richardson insisted upon full compliance of WIPP with 40 CFR 191 before receipt of any waste and funding for roads attached to bill).[B28] **Dec:** ID Gov. Andrus, CO Gov. Romer, and NM Gov. Carruthers meet in Salt Lake City to discuss WIPP and options to avert shutdown of Rocky Flats Plant from lack of storage imposed by CO, and inability to ship to ID; DOE agrees to pursue both administrative and legislative land withdrawal for WIPP.[B29]

- 1989 - Westinghouse completes "no-migration" petition for RCRA variance for WIPP pilot phase.[B30]

- 1990 - EPA issues no migration variance for test phase.[B31]

- 1991 - Westinghouse completes Parts A & B of RCRA permit application to state.[B32 B33]

1992 Congress passes Land Withdrawal Act

- 1992 - **Oct:** WIPP Land Withdrawal Act (LWA)[B34]:
  - transfers land from DOI to DOE
  - establishes EPA as regulator for WIPP
  - requires recertifying site every 5 yr
  - reinstates Subpart B of 40 CFR 191 except disputed individual and groundwater requirements
  - requires DOE cooperation with EEG
  - New Mexico given $600 million over 30 years

1993 EPA repromulgates 40 CFR 191

- 1993 - **Dec:** In response to court remand and WIPP LWA, EPA repromulgates 40 CFR 191 – no influential changes for WIPP.[B35]

- 1995 - **Jan:** EPA proposes compliance criteria for WIPP in 40 CFR 194.[B36]

1996 EPA states how to implement radioactive waste standard in 40 CFR 194

- 1996 - **Feb:** EPA promulgates final 40 CFR 194.[B37]
  - requires analysis and peer review of waste characterization, engineered barrier evaluation, and conceptual models
  - requires a monitoring system
  - specifies requirements on quality assurance (QA), peer review, and expert judgment
  - requires potash mining to be considered

  Sep: Congress amends WIPP LWA and relieves WIPP of need to comply with land disposal restrictions of RCRA, but other requirements of RCRA still apply.[B38]

- 1997 - **Oct:** EPA issues draft rule to approve WIPP with conditions: requires use of panel seals used in PA; design requires QA for waste generators; lists requirements for using process knowledge to characterize wastes; requires schedule for installing passive controls; denies any protective credit for passive controls; and 120-day public comment period begins.[B39]

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Figure 2-2. Performance goals were added or changed several times over the 25-yr history of the WIPP Project.

Appeals Court for the District of Columbia agreed, stating that all impacts—environmental, economic, and cultural—should be included in the EIS (CCCC, 1971). However, the level of detail required was still not clear. Consequently, preparing the first EIS for the WIPP was particularly difficult and was delayed several times as the nation's notion of such a document changed from an initial expectation of a relatively short, 10-page report to an extensive document of several thousand pages.

For the WIPP, three EISs were prepared: one in 1979-80 during deliberations on whether to proceed with the construction phase (DOE, 1979; 1980a), a supplement in 1989-90 to decide whether to proceed with a pilot phase (DOE, 1989b; 1990c), and a supplement in 1996-97 (DOE, 1996b; 1997) to decide whether to proceed with the disposal phase (Figure 2-2).

2.3 Establishment of Regulatory Risk Goals

2.3.1 Historical Events

In 1976, President Gerald Ford requested that the EPA accelerate development of applicable standards for proposed waste repositories (EPA, 1985a; Ford, 1976). In response, the EPA conducted several public meetings to develop societal consensus on regulatory criteria (EPA, 1978a, 1978b). In 1978, the EPA proposed generic criteria on all radioactive waste, but after receiving an unfavorable response, it withdrew the proposed regulations in March 1981 (EPA, 1985b) and began to develop standards for individual categories of radioactive waste.

In 1982, in response to a requirement in the Nuclear Waste Policy Act (NWPA) of 1982 (Public Law 97-425), the EPA officially published a draft of the disposal regulation for high-level nuclear waste, 40 CFR 191 (EPA, 1982), which had seen more than 20 revisions. One year later, the DOE stated in letters to the Environmental Evaluation Group (EEG), an oversight group for the WIPP established by the DOE in 1978, that the WIPP would comply with the requirements in 40 CFR 191 once the facility moved from a test phase to an operational phase but did not consider them applicable in the test phase; the latter interpretation contributed to a growing opposition against the test phase of the project. The EPA did not promulgate the final version of 40 CFR 191 until 1985, three years after submitting the proposed regulation and then only after drawing a lawsuit to hasten promulgation (EPA, 1985a). The 1985 Standard established criteria for the disposal system as a whole, and defined the term "performance assessment" (PA) as the type of calculations to be used to show compliance with this regulation.[3]

The 1985 final version of 40 CFR 191 involved some changes from the 1982 draft that included Individual and Groundwater Protection Requirements. These requirements led to a lawsuit by the same group, the Natural Resources Defense Council (NRDC), that had sued earlier to accelerate promulgation. The courts remanded the regulation shortly thereafter (NRDC vs. EPA et al., 1987), but the EPA repromulgated the Standard in 1993 without changes to the Containment Requirements (EPA, 1993b). These Containment Requirements strongly influenced the type of calculations necessary for the performance assessment.

In 1992, Congress defined the process by which WIPP compliance would be evaluated, transferred ownership of the WIPP site to the DOE, and designated the EPA as the regulator of the WIPP. The law officially marked the transition from the construction and system characterization phase to the compliance and testing phases, although the latter phases had already begun, informally, when the EPA issued 40 CFR 191 in 1985 and when Sandia first demonstrated the performance assessment process using the EPA standard in 1989 (Marietta et al., 1989; Bertram-Howery et al., 1989). In 1996, the EPA promulgated 40 CFR 194 (EPA, 1996a), a regulation to implement its 40 CFR 191 standard specifically for the WIPP, which imposed several new requirements and interpretations on modeling style. Basically, however, 40 CFR 194 adopted a process and methodology very similar to that used by Sandia for conducting performance assessments of the WIPP between 1989 and 1992 (Helton, 1993a; 1994; Helton et al., 1993a; 1997c).

Besides complying with 40 CFR 191, waste intended for the WIPP also came under a legal ruling in 1984 (LEAF v. Hodel, 1984) and subsequent changes in the EPA's and DOE's definitions of mixed waste (in 1986 and 1987 [EPA, 1986; DOE, 1987]), causing as much as 60% of the waste destined for the WIPP to be designated as chemically hazardous. Thus, the WIPP had to comply with a set of regulations for hazardous waste (40 CFR 260-270 and analogous New Mexico

3 During this period, the term performance assessment (PA) was adopted internationally to describe a general process of evaluating whether a geologic disposal system performed adequately and complied with regulatory criteria. Because the topic of this report is assessing compliance of WIPP with the EPA regulation, performance assessment, herein, refers to the specific EPA process.

regulations) promulgated in response to the Resource, Conservation, and Recovery Act (RCRA) (Public Law 98-616) and its major 1984 amendment (Public Law 94-580). In anticipation of the pilot phase for the WIPP, the DOE completed a No Migration Petition for a variance to RCRA regulations (DOE, 1989a), which the EPA granted for the test phase in 1990. In 1996, Congress eliminated the need for the WIPP to seek a no-migration variance from RCRA requirements for the disposal phase (Public Law 104-201), since the petition would have required calculations similar to those already provided for compliance with 40 CFR 191. However, other RCRA requirements as implemented in 40 CFR 260-270 still applied, such as specifying the types and amounts of waste in permits.

2.3.2 Containment Requirements of 40 CFR 191

Although dose calculations to humans were still of interest to the WIPP Panel of NAS and were used in supplemental EIS reports for the WIPP in 1989 (DOE, 1989b; 1989c; 1990c) and 1996 (DOE, 1996b; 1997), 40 CFR 191 required the evaluation of the cumulative release of radioisotopes (R) to the accessible environment. Furthermore, the definition of performance assessment stated that "these estimates [of releases] shall be incorporated into an overall probability distribution of cumulative release to the extent practicable." That is, uncertainty was treated quantitatively, which in turn indirectly required a stochastic model for the geologic disposal system. To elaborate, if $R(\bullet)$ is a function of parameters $\mathbf{x} = \{x_1, x_2, \ldots x_{nP}\}$, where nP is the total number of model parameters, and parameters $\mathbf{x}$ are uncertain or imprecisely known (i.e., the parameters cannot be assigned a single, universally accepted value, and thus are described by a probability model by means of a distribution with a range of values), then the consequences $R(\mathbf{x})$ will also have a range and distribution (Figure 2-3). Uncertainty analysis involves determining the shape of the distribution $R(\mathbf{x})$.

The risk measure for a performance assignment was not the expected value of the results, as in simple insurance calculations, nor the variance of the results, as in financial risk analysis of stock portfolios (Markowitz, 1952). Rather the entire distribution of the results was used. Hence, in the United States, performance assessment became a stochastic simulation of possible long-term behaviors of a real system by means of a computer-implemented mathematical model. In this respect, performance assessment remained similar to probabilistic risk assessments (PRAs) for nuclear reactors.

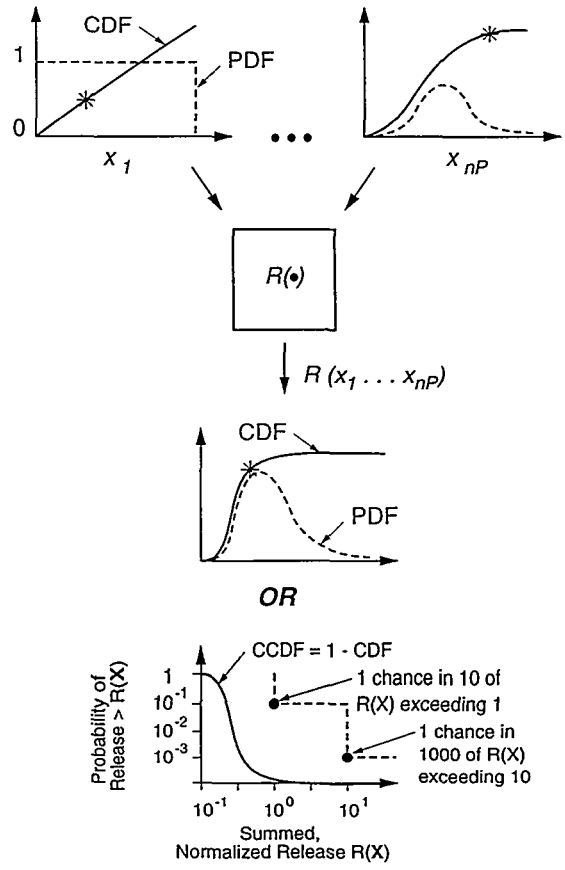

Figure 2-3. Uncertainty in model parameters p (expressed as a probability density function [PDF], or cumulative distribution function [CDF]) results in uncertainty in the results. This uncertainty (expressed as a complementary distribution function CCDF or 1-CDF) is compared to the probabilistic limits in 40 CFR 191.

The Containment Requirements of 40 CFR 191 established a 10,000-year regulatory period, during which system performance (such as actinide radioisotope transport in groundwater) had to be modeled. In 40 CFR 191, the Containment Requirements specified (a) limits (L_i) on the total activity (curies) that could be released from the disposal system for individual radioisotopes or (b) the chance that these limits could be exceeded (i.e., have less than 1 chance in 10 of exceeding L_i or less than 1 chance in 1000 of exceeding 10 L_i) (Figure 2-3). For a mix of radioisotopes, the release (R) for each radioisotope (i) was normalized with respect to its radioisotope limit (L_i).

$$R = \frac{1}{f_w}\left(\frac{Q_1}{L_1} + \frac{Q_2}{L_2} + \ldots \frac{Q_{nR}}{L_{nR}}\right)$$
$$= \frac{1}{f_w}\sum_{i=1}^{nR}\left(\frac{Q_i}{L_i}\right) \leq 1 \text{ (or 10)}$$

where

f_w = waste unit factor, various factors are defined (e.g., for transuranic waste $f_w = \sum A/10^6$ Ci where A is activity (Ci) of alpha-emitting transuranic radioisotopes in the repository with half-lives greater than 20 yr)

L_i = The release limit specified by the EPA for radioisotope i in 40 CFR 191

nR = number of radioisotopes contributing to release to the accessible boundary

R = total normalized release ("EPA sum")

Q_i = cumulative release of radioisotope i to accessible boundary

$$= \int_0^{10,000\ yr} q_i dt$$

The Containment Requirements specified the radioisotope limits (L_i); in other words, as customary, the regulatory agency had performed the dose-response assessment.[4] Because the Containment Requirements used cumulative releases of radioisotopes (Q_i), the EPA dose-response assessment, through crude calculations, converted from dose, which depends upon rate of release (rather than cumulative release), to obtain the EPA limits (L_i) (EPA, 1985b).

Bear in mind, however, that the calculations were intended to be illustrative rather than truly predictive. The EPA standards are an examination of the characteristics of geologic disposal system under a specified set of hypothetical circumstances (e.g., intrusion into the repository in the future, such as year 2500, by a drilling crew exploring for hydrocarbons with 500-yr-old technology). These circumstances are not the actual future expected by the EPA any more than the exam problem by a professor is the exact problem a student will encounter in his or her future. But the calculations in a performance assessment for a geologic repository were designed to demonstrate whether the disposal system had sufficient desirable characteristics given a normal evolution over geologic time to mitigate the release of radioisotopes.

4 This dose-response assessment depended on *bounding* type dose evaluations that sought to limit deaths to no more than would have occurred from an unmined uranium ore body, i.e., less than 1000 deaths over 10,000 yr. Thus, a performance assessment is not entirely probabilistic.

3.0 Site Characterization

To model a system, it is necessary first to define the system, regardless of whether the model's purpose is to gain insight or anticipate future behavior. System definition essentially describes the parameter space **x** and the relationship of those parameters through a model. The model may be only conceptual, but for the WIPP Project it was a mathematical model. In a mechanical system, the various parts of a system are *defined.* With a radioactive waste disposal system, however, the existing geologic parts must be *characterized.* Characterization of the subsystems of the WIPP was undertaken as soon as the site was selected, even before safety goals and a compliance process for waste disposal were established.

Before the late 1980s, site characterization activities were undertaken (1) to satisfy needs for the draft EIS in 1979 (DOE, 1979) and Supplemental EISs in 1989 (DOE, 1989c, 1990c), (2) to satisfy negotiated agreements with the State of New Mexico (State of NM, 1981; Documents, 1982), and (3) to develop a general understanding of natural phenomena significant to nuclear waste disposal. These site characterization studies investigated dissolution areas, breccia pipes, salt permeability, and radioisotope sorption.

3.1 Selection of Site

With the encouragement of Carlsbad's politicians and citizens, and the tacit approval of New Mexico's governor Bruce King, Oak Ridge National Laboratory (ORNL) examined for the AEC a portion of the Permian Basin called the Delaware Basin in southeastern New Mexico in the early 1970s for a suitable disposal site for nuclear waste. The citizens of Carlsbad were open to the idea of building a repository, because of the declining potash mining industry. During the 1960s, sylvite and langeinite had been extensively extracted from nearby potash mines for use in fertilizer production, but many had since closed because of foreign competition. The area was semiarid with little potable water and no significant, highly permeable aquifers near the surface. At that time, the area was being considered for disposal of both high-level waste and TRU waste generated during production of nuclear weapons. A potential site near the edge of the basin along the Capitan Reef was identified in 1973 (Figure 3-1).

ORNL drilled two wells, AEC-7 and AEC-8, in March 1974 near the northeastern and southwestern corners of the rectangular site for the first large-scale field test in the basin (Powers et al., eds., 1978). (Prior to this test, the USGS had conducted some tests for the AEC in 1961 for the Gnome Project and so some data on aquifers was already available [Gard, 1968; Cooper and Glanzman, 1971].) The cores from the two wells indicated fairly predictable stratigraphy. However, the work by ORNL was suspended two months later for several reasons. First, the AEC wished to emphasize the Retrievable Surface Storage Facility as its primary option. Also, because of the oil embargo, AEC Chairman Dixie Lee Ray would not withdraw the land around the site from oil exploration. Finally, Congress was considering a major reorganization of the AEC; legislation signed in October 1974 split the AEC into the Energy Research and Development Administration (ERDA) and the Nuclear Regulatory Commission (NRC), effective January 1975.

One of New Mexico's ERDA laboratories, Sandia National Laboratories, began work at the site selected by ORNL in January 1975. In late March of that year, Sandia received funding and officially became the lead laboratory to characterize the site (Powers et al., eds., 1978), develop the conceptual design (Sandia, 1977), initiate scientific studies on nuclear waste disposal in bedded salt, and draft an EIS (DOE, 1979), accompanied with instructions not to study "things to death" (Armstrong, 1975). In May 1975, Sandia drilled a combination geologic and exploratory well, ERDA-6, at the northwestern corner of the proposed site (Sandia and USGS, 1983). The well encountered up to 75° dipping beds near the planned lower level of the repository and, at a depth of 826 m, artesian brine and H_2S gas, causing genuine dismay on the part of ERDA staff who, as former AEC officials, had earlier struggled with difficulties at the Lyons, Kansas, site.

But the large Delaware Basin had other areas that met the various selection criteria, that is, a distance of at least a mile from boreholes, high purity salt at depths between 300 and 800 m, a lack of dissolution at the top of the Salado Formation, an avoidance of known oil and gas trends, and a minimal amount of state land, private land, and potash zones (Powers et al., eds., 1978). In late 1975, after examining the stratigraphy shown in the ERDA wells, along with confidential stratigraphic and geophysical seismic data from numerous private wells owned by oil companies, Sandia recommended locating the potential repository site nearer the basin center. The USGS independently suggested a similar location. The new site was about 11 km southeast of the first location with more predictable horizontal stratigraphy, i.e., away

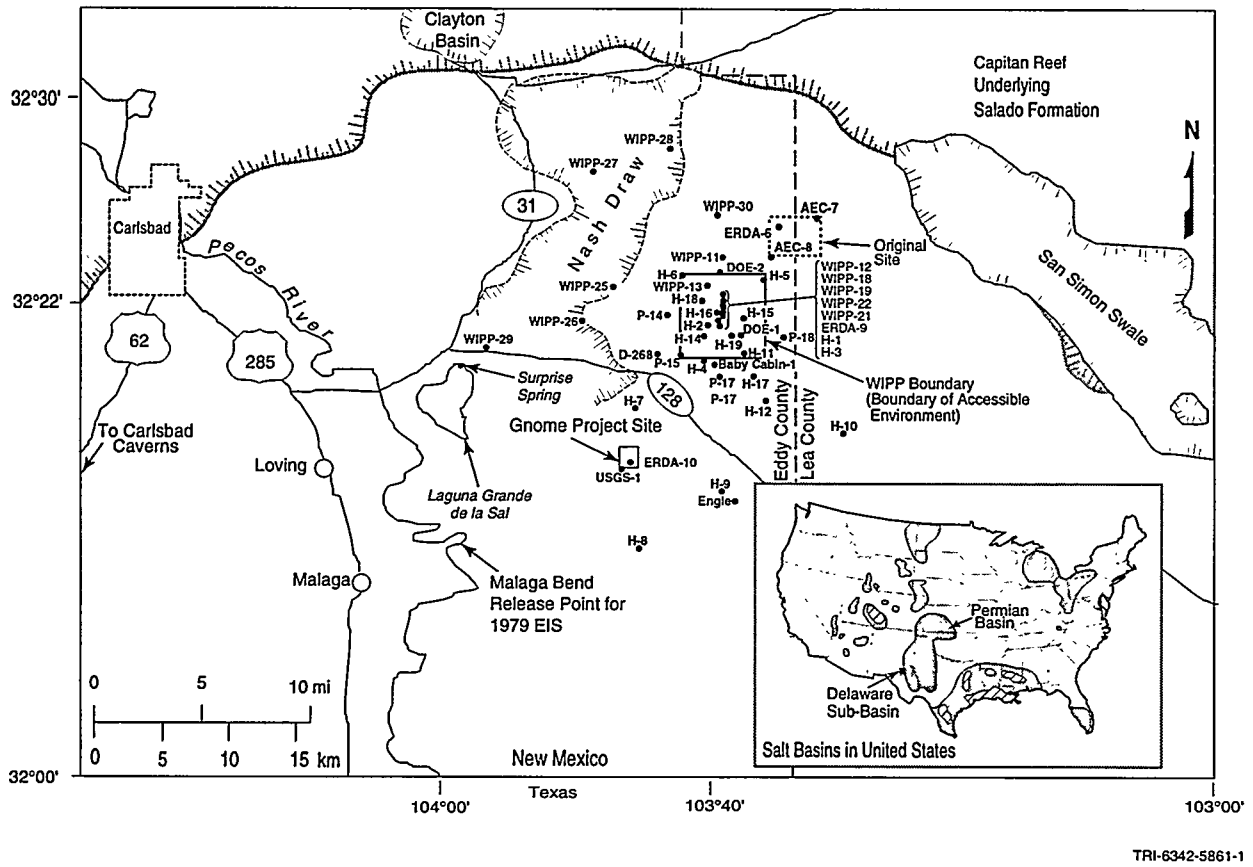

Figure 3-1. Physical setting of the WIPP in Delaware Basin (after Rechard, 1995, Figure 2.1-1).

from the ~10-km band around the Capitan Reef where deformation of the salt beds had occurred (Powers et al., eds., 1978). In April 1976, Sandia drilled ERDA-9 through the Salado Formation into the Castile Formation (hereafter shortened to Salado and Castile) at the center of the proposed site, which was 42 km from the town of Carlsbad, New Mexico. The stratigraphy was horizontal and no brine was encountered. This site became the Waste Isolation Pilot Plant (WIPP), which had been officially named four months earlier (NAS/NRC, 1984) (Figure 3-1).

3.2 Site Characterization Studies for EIS

The EIS process, as required by NEPA (Public Law 91-190), exerted its influence on the characterization process during the 1970s as the AEC (then the ERDA, in 1975, and ultimately the DOE) continued investigations on bedded salt in New Mexico. Because of ERDA's eagerness, the deadline for the EIS draft was originally October 1976, but a more immediate need was for Sandia to spend its resources locating a new site. Also, the nature of the EIS changed nearly as often as the official mission of the WIPP (as discussed in Section 4) and ultimately contributed to a 3-year delay in publishing the EIS. During this time Sandia collected data on the ecological, meteorological, and archaeological features of the region and its socioeconomic facets (Weart, 1979). Sandia's major description of the geology and hydrology (Powers et al., eds., 1978) drew upon information from Sandia's geologists (Griswold, 1977) and the USGS reports, which had been prepared when the USGS was searching for a regional site location (Pierce and Rich, 1962; Brokaw et al., 1972; Bachman et al., 1973; Jones et al., 1973; Bachman, 1973; Barnes, 1974).

During site selection, interest in fluid flow in water-bearing units of the area had focused on its effects on dissolution; after selection, interest shifted to the role of these units as potential pathways for radionuclide release. Also 75 new line miles of seismic reflection data and 9000 resistivity measurements were collected across the site (Powers et al., eds., 1978). In addition,

47 boreholes were completed under the direction of the USGS: 12 boreholes for evaluating geologic stratigraphy and cores, e.g., the 2 AEC wells, the 2 ERDA wells, and new wells designated as WIPP-#, such as WIPP-11; 21 boreholes for evaluating potash reserves, designated P1 through P21; and 14 boreholes for evaluating the permeability of various layers, usually the Rustler Formation, designated H#, e.g., H1. Four potash boreholes, P14 through P18, were also converted to hydrologic boreholes (Mercer and Orr, 1979). In the 1980s, other wells were also used to observe water drawdown from large-scale pumping tests (Figure 3-2).

USGS geohydrologists had suggested that the Magenta and Culebra Members of the Rustler Formation (hereafter shortened to Magenta, Culebra, or Rustler) and the Rustler/Salado contact zone were potential pathways for radionuclide release. At the time, the relative importance of these units was unknown, so the first tests targeted all three. The Rustler/Salado contact was confirmed to be transmissive in Nash Draw but did not yield significant quantities of water at the site and thus did not represent a significant pathway for fluid movement for either radionuclide transport or dissolution. The Culebra was more transmissive than the Magenta, and the transmissivities of these units varied by several orders of magnitude (Powers et al., eds., 1978; Mercer and Orr, 1979).

Experimental activities included determining the mineralogy of the bedded salts and the overlying formations (Powers et al., eds., 1978). Also, the sorptive properties of the clays in the salt and overlying dolomitic rocks in the Rustler were evaluated (Serne et al., 1977; Dosch and Lynch, 1978; Dosch, 1979), and the geochemical composition of the waters in the Rustler was determined (Powers et al., eds., 1978). The stability of the salt was also examined by determining whether Rb-Sr isotope ratios suggested any significant recystallization or brine flow through the formation since its deposition about 255 million years ago (Powers et al., eds., 1978; Weart, 1979) (Figure 3-3).

3.3 Long-Term Site Characterization Studies

3.3.1 Site Characterization Studies at Repository Horizon

Data needs for the EIS engendered several characterization studies. Although the results were not ready to be reported in the supporting documents for the EIS,

	Tracer Test	Drill Stem Test	Slug Test	Single-Well Pumping Test	Multi-Well Pumping Test	Responded to Large-Scale Pumping Test	Pumping Well for Large-Scale Pumping Test
AEC-7			●				
Cabin Baby-1			●			●	
D-268			●			●	
DOE-1				●		●	
DOE-2		●	●	●		●	
Engle					●		
ERDA-9			●			●	
H-1			●			●	
H-2	●		●		●	●	
H-3	●		●		●	●	●
H-4	●		●		●	●	
H-5			●		●		
H-6	●		●	●	●	●	
H-7				●	●		
H-8			●	●			
H-9			●	●	●		
H-10			●				
H-11	●	●	●		●	●	●
H-12			●	●		●	
H-14		●	●			●	
H-15		●	●			●	
H-16		●	●				
H-17		●	●			●	
H-18		●	●	●		●	
H-19	●			●	●		
P-14						●	●
P-15			●			●	
P-17			●			●	
P-18			●				
USGS-1					●		
WIPP-12			●			●	
WIPP-13							●
WIPP-18			●			●	
WIPP-19			●			●	
WIPP-21			●			●	
WIPP-22			●			●	
WIPP-25				●		●	
WIPP-26			●	●			
WIPP-27			●	●			
WIPP-28			●	●			
WIPP-29				●			
WIPP-30			●	●		●	

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Figure 3-2. Well tests conducted to characterize transmissivities of the Culebra Dolomite Member of the Rustler Formation (WIPP PA Division, 1991/1992, Vol. 3, Figure 2.6-3; Hill et al., 1997).

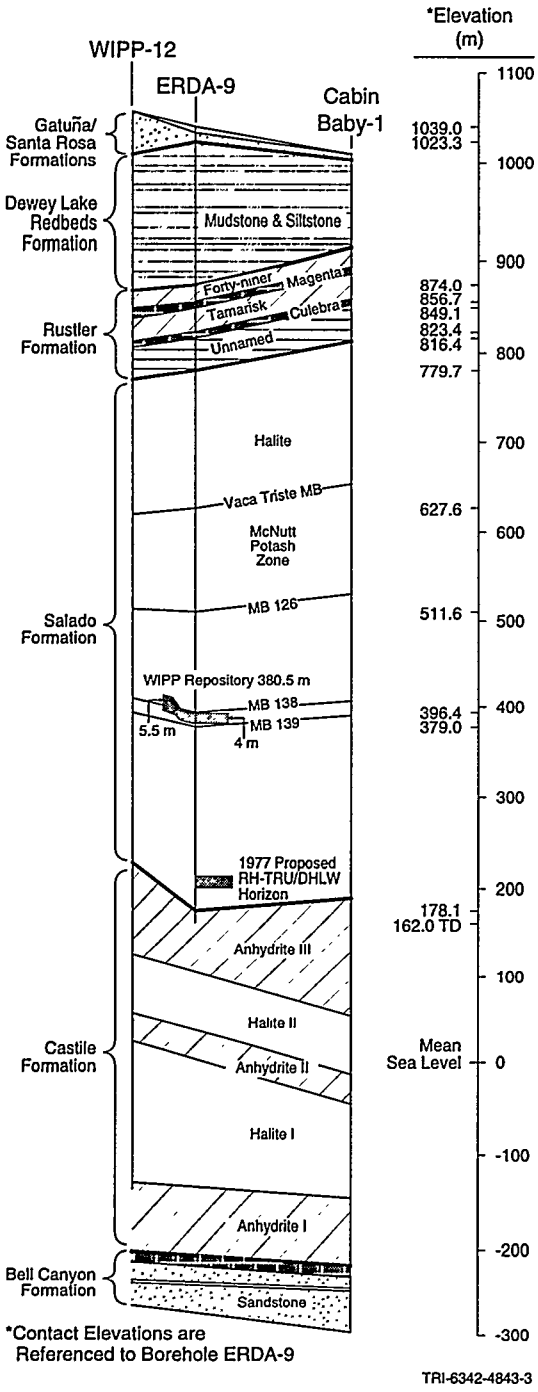


Figure 3-3. Stratigraphy above and below the WIPP repository (after Rechard, 1995, Figure 2.1-2)

some data were available soon afterwards. For example, with regard to waste acceptance criteria, the potential magnitude of the pressure from gas generated by waste from microbial degradation of organic material was not known; thus in 1979 Sandia tested the AEC-7 well to determine the permeability of the Salado in order to ascertain the ability of the repository to contain gas (Christensen et al., 1980). The test, using compressed air flow measurements, indicated a formation permeability over a 30-m test section between 5×10^{-19} and 2×10^{-17} m^2 (permeability at an elevation of 427 m [depth of 690 m] and at an elevation of 548 m, respectively) (Tyler et al., 1988).[1] The range of in situ permeability, in turn, suggested that if gas from microbial action (and anoxic corrosion) from brine migration to hot canisters were generated by TRU waste at less than 5 mole/drum/yr, the gas would dissipate into the rock without reaching lithostatic pressure; thus the TRU waste would be acceptable in the WIPP repository without incineration. Accordingly, a fledgling program to characterize gas generation, begun in 1978, was canceled after 1979 (see also Section 4).

Sandia did not originally intend to use the experimental region of the WIPP repository to study the permeability of the Salado. TRU waste generates relatively little heat, and so migration of brine because of a thermal gradient was of little concern after 1979 when high-level waste was excluded from the WIPP. Yet Sandia conducted three tests of fluid migration in a nearby potash mine in 1981 (Molecke and Torres, 1984) because of continued interest in high-level waste disposal in salt beds elsewhere and as part of the sealing and waste canister programs (see also Section 5).

Sandia first measured injected nitrogen flow around a WIPP drift in 1984 to determine the extent of the disturbed zone and corresponding permeability of the Salado. In 1986, Sandia conducted similar measurements using injected brine to evaluate the permeability of the intact salt. The predicted permeability ranged between 10^{-21} and 10^{-20} m^2, a factor of about 1000 less than the previously measured range in AEC-7 of 5×10^{-19} to 2×10^{-17} m^2. Several factors contributed to the early erroneous data, such as leakage around the packers through the disturbed rock zone, the short duration of the in situ tests, and the inability in laboratory experiments to measure permeabilities of less than 5×10^{-20} m^2 on the small ERDA-9 cores (Tyler et al., 1988) (Figure 3-4).

1 This range was higher than that encountered in laboratory tests using argon, nitrogen, and hydrogen gases which ranged between 1.5×10^{-17} and an experimental limit of 5×10^{-20} m^2.

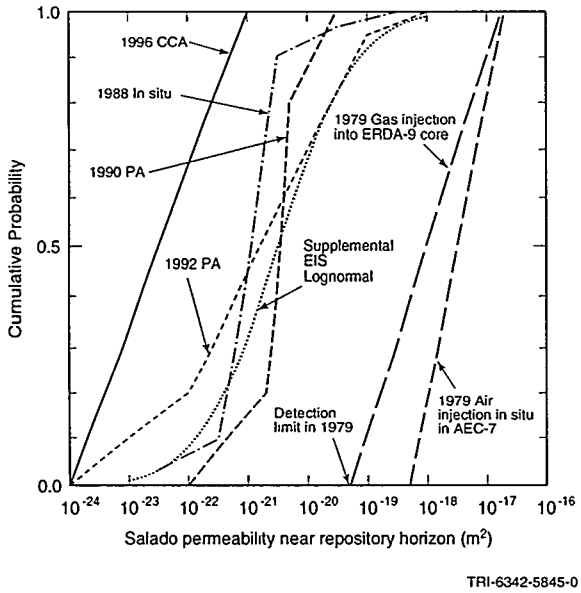


Figure 3-4. Estimated permeability of Salado Formation near repository horizon continued to decrease over the years.

In May 1987, Sandia reported that much more brine had migrated to the simulated high-level waste canisters than had been expected. Furthermore, a rough scoping calculation to evaluate repository performance identified inflow of intergranular Salado brine to the repository from a pressure gradient (rather than thermal gradient) as a concern for long-term performance if a human should inadvertently intrude into the repository with an exploratory well (Brush and Anderson, 1989). By December, the national press was reporting on the issue of brine flow into the repository (Begley and Miller, 1987). In January 1988, New Mexico's congressional delegation asked the full Board of Radioactive Waste Management (BRWM) of the NAS to study the brine inflow controversy.

However also in 1987, Sandia made a concerted effort to theoretically study thermoelastic behavior of the salt and found that the seemingly contradictory in situ measurements of lower permeability but higher brine flow to simulated high-level waste canisters could be explained. The new theoretical model predicted less than 43 m^3 (<0.5% of the original volume of a disposal room) would enter over the first 100 yr (Nowak et al., 1988). Given the resolution of measurements and the small amount of brine, the BRWM concluded that not enough brine would seep into the rooms to form a slurry of radioactive waste before the rooms had closed through salt creep. The NAS also proposed a verification test in a circular tunnel designated as Room Q. On the other hand, the lower permeability of the Salado had implications for gas generation, as discussed below.

3.3.2 Site Characterization Studies for Stipulated Agreement

Long-term characterization studies beyond the EIS were motivated by two requirements. First, Sandia was assigned to develop a scientific understanding of natural phenomena as deemed prudent by its scientists and members of the WIPP Panel of NAS. Second, 17 geotechnical experiments were undertaken during the 1980s to satisfy agreements with the State of New Mexico.

Although the political climate in New Mexico toward waste disposal was initially positive, the reevaluation of nuclear waste disposal by the Carter administration and the possibility of placement of commercial waste at the WIPP led the State of New Mexico to distrust DOE's intentions. To help with resolution of issues, the enabling legislation clearly stated the purpose of the WIPP and required a Consultation and Cooperation Agreement with the State of New Mexico. However, negotiations to reach such a cooperative agreement between New Mexico and the DOE were arduous. Thus, when DOE decided to proceed to preliminary design and construction in January 1981, the State of New Mexico sued along with several other parties. In response to the lawsuit, a Stipulated Agreement was negotiated between Governor King and Secretary of Energy Edwards in late summer of 1981, which included the Consultation and Cooperation Agreement. The new agreement defined the relationship of the WIPP Project with the State of New Mexico and itemized required geotechnical experiments based on state concerns from the final EIS (DOE, 1980a) (see Figure 2-2).

As part of the negotiated settlement with the State of New Mexico, the DOE deepened WIPP-12 into the Castile in November 1981 (Figure 3-5). The WIPP Project encountered a brine reservoir with pressure high enough that brine could flow to the surface. The discovery of the reservoir prompted the rotation of the waste panels from their planned location north of the experimental area to south of the shafts in 1982, thus moving the disposal region ~1800 m to the south, the current configuration. This well was extensively tested through 1983 (Lappin et al., eds., 1989). Also, geophysical studies indicated that a brine reservoir could also extend to the south of WIPP-12. DOE-1 was drilled in 1982 to obtain geologic data and evaluate the brine reservoirs in the Castile. No brine was found, and the well was later used to test hydraulic conditions in the Culebra. Both

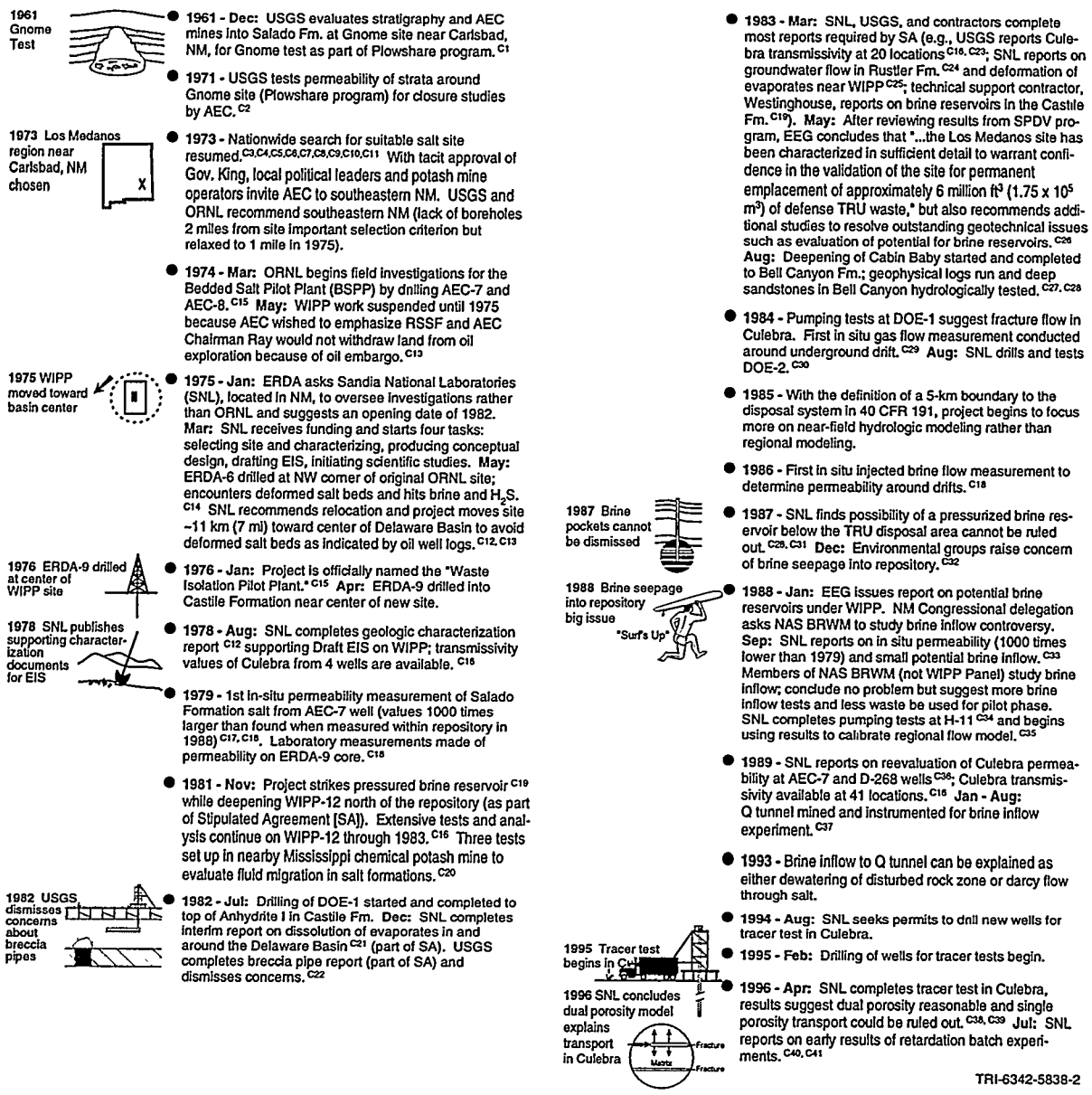


Figure 3-5. Site characterization studies at the WIPP.

the DOE and the EEG conducted consequence analyses of a drilling encounter with a brine reservoir similar to that found in WIPP-12, concluding that the health consequences were minor (Woolfolk, 1982; Channell, 1982). In addition, several new geophysical techniques were used to determine whether such brine reservoirs might exist under the waste panels, but by 1987, these studies were inconclusive. A zone of lower resistivity in the Castile existed under a portion of the waste disposal area and could be interpreted as brine; however, the zone was beneath the upper anhydrite layer in the Castile where brine had been encountered earlier (Lappin, 1988; Earth Technology Corp., 1988; WIPP PA Division, 1991/1992, Vol. 3).

In the 1980s, hydrologic characterization focused on the Culebra. Mercer (1983) had provided additional information on the transmissivities of the Culebra as part of the Stipulated Agreement. In 1984, pumping tests at DOE-1 suggested fracture flow in the Culebra. By 1987, Sandia had estimated the Culebra transmissivity at 15 new locations and re-estimated the transmissivity of 7 wells (Mercer, 1983). By 1989, Sandia had estimated Culebra transmissivity at 41 locations in a 860-km^2 area around the WIPP site (Lappin et al., eds., 1989; Lappin, 1988) (see Figure 3-2).

3.4 Site Characterization Studies for CCA

As summarized below, additional site characterization studies were conducted in the 1990s for PA calculations needed for the CCA (Compliance Certification Application) to the EPA.

3.4.1 Fluid Flow in Culebra

In the 1990s, tests were conducted to characterize the Culebra at two relatively high-permeability locations. These high-quality tests included the seven-well tracer test conducted at the newly drilled well, H-19, multiwell retesting at H-11, and single-well injection and withdrawal tests at both H-19 and H-11. The purpose of the tests was to evaluate the complex fracture flow (Meigs and McCord, 1996; Meigs et al., 1997).

In 1996, hydraulic tests and well logging at the H-19 hydropad suggested that the permeability of the upper portion of the Culebra was significantly lower than the permeability of the lower portion. In addition, the 1996 tracer tests at H-19 suggested that the upper portion of the Culebra did not substantially contribute to solute transport. These findings confirmed two previous observations. In 1979, Mercer and Orr (1979) reported that based on ^{131}I tracer tests at the H-3 hydropad, 100% of the flow came from the lower 3 m of the Culebra. In a description of the Culebra in the Air Intake Shaft in 1990, Holt and Powers (1990) noted that most of the fluid observed came from the lower portion of the Culebra. Therefore, the hydrologic effective thickness of Culebra in the 1996 PA for the CCA was taken as ~4 m (7 m less the 3 m for the upper portion of the Culebra) (Meigs and McCord, 1996) (Figure 3-6).

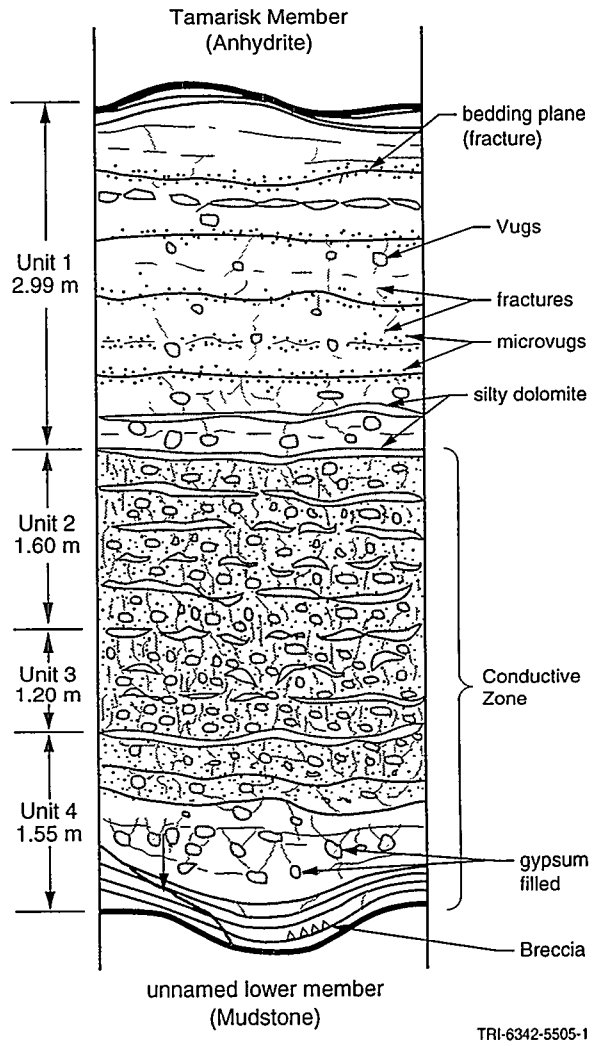


Figure 3-6. Stratigraphic subdivisions of Culebra Dolomite Member of Rustler Formation discerned after 1997 (after Holt, 1997, Figure 1-2).

3.4.2 Sorption Studies in Culebra

Sandia conducted several laboratory studies of sorption in the 1980s (Dosch, 1979; 1980; 1981; Lynch and Dosch, 1980; Lynch et al., 1981; Tien et al., 1983). These empirical studies used a variety of sorbents (dolomitic, anhydritic, and clay-rich rocks) and solutions (Salado, Castile, and Culebra brines), and in some cases included the effects of dissolved organics on sorption.

Because the State of New Mexico felt the early sorption studies were deficient, the DOE and the State of New Mexico modified the Consultation and Cooperation Agreement in 1988 to require New Mexico concurrence on any sorption distribution coefficients (K_ds) recommended for use in the final performance assessment of the WIPP. While experimental data were being obtained, Sandia convened a panel of staff members to estimate ranges and probability distributions of K_ds in support of two preliminary performance assessments (Trauth et al., 1992). In addition, an experimental program sought to develop a mechanistic surface-complexation model for the sorption of uranium by corrensite, a clay mineral found in the Culebra (Lappin et al., eds., 1989; Siegel et al., 1990; Park et al., 1995). However, by 1996, the only model available was of the sorption of U(VI). Thus, retardation of radioisotopes in the Culebra was more thoroughly studied in the laboratory. Early results from a batch experimental program using crushed dolomite were used in the CCA (Brush and Storz, 1996; Papenguth and Behl, 1996) (Figure 3-7).

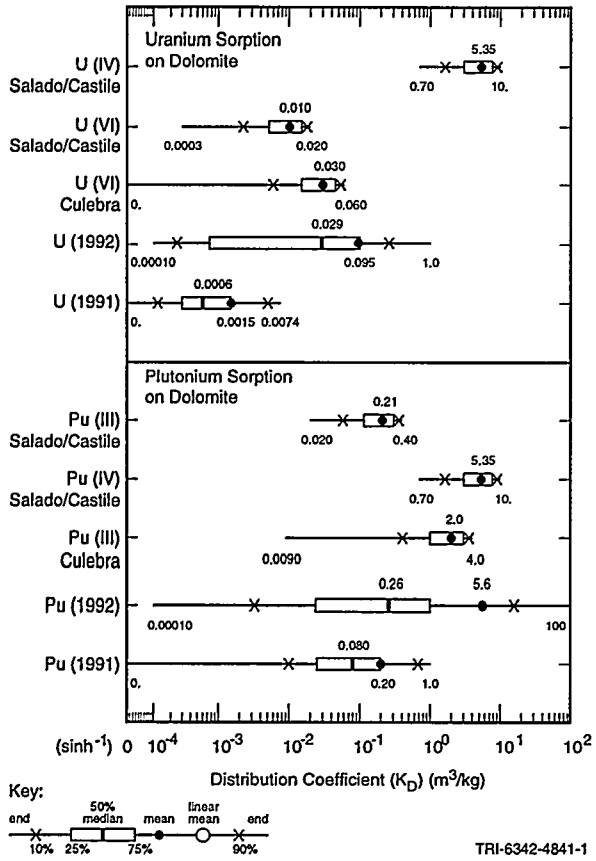


Figure 3-7. Ranges of K_ds (m^3/kg) for plutonium and uranium in 1991, 1992, and 1996 performance assessments (WIPP PA Division, 1991/1992, Vol. 3; WIPP PA Department, 1992/1993, Vol. 3; Brush and Storz, 1996).

4.0 Waste Properties

Two aspects of the TRU (transuranic) waste destined for the WIPP were important to characterize for the compliance calculations: the estimated inventory of radioisotopes in the waste,[1] and the mobility of these radioisotopes within the disposal system. Properties of the waste that influence the chemical environment within the disposal system, and thereby radioisotope mobility, include the amount of iron and microbial nutrients such as wood, plastic, rubber, nitrates, and sulfates. Only defense TRU waste was planned for disposal at the WIPP after December 1979, when the purpose of the WIPP Project was clearly defined. But up until then, other types of nuclear waste had also been considered.

4.1 Changes in Purpose of WIPP Project

Although the purpose of the WIPP as a general disposal site for radioactive waste was clear, decisions about what kind of waste would be sent there fluctuated throughout the 1970s. Under consideration were three kinds of waste—TRU waste, high-level waste from reprocessing spent nuclear fuel, and direct disposal of spent nuclear fuel—and whether the waste originated from defense or commercial activities. The initial focus of the AEC in the 1950s and 1960s with regard to disposal was on nuclear waste from reprocessing spent nuclear fuel (or high-level waste as it came to be called), because uranium was thought to be in such short supply that it would be necessary to recycle commercial fuel. The initial screening analysis of scenarios for the WIPP assumed 75,000 canisters of high-level waste, enough to accommodate the anticipated volume of high-level waste from all commercial reactors through the year 2000 (Claiborne and Gera, 1974). However by 1975, the emphasis of the WIPP was disposal of TRU defense waste, because of the prominence of the latter as a result of the Rocky Flats Plant fire, and the WIPP was officially removed from the commercial repository program within ERDA.

Yet the public expectation remained that the government should be responsible for disposal of wastes from commercial reactors, as prompted by the Atomic Energy Act of 1954 (Public Law 83-703). During the 1970s, the lack of a proven waste disposal scheme for high-level waste was presented by the public, through comments on the EIS and at licensing hearings, as an argument against construction of nuclear plants. Furthermore, California passed a law in 1976 banning construction of nuclear power plants until disposal of high-level waste was demonstrated (Carter, 1987, p. 86; Perge, 1982). Therefore, once the RSSF for high-level waste was officially abandoned in 1975, the DOE seriously considered disposal of commercial waste at the WIPP between 1977 and 1979, at least as a means of demonstrating the disposal concept. Conceptual drawings of the WIPP repository in 1977 showed two levels: one for the cooler contact-handled TRU (CH-TRU) waste, 640 m below the surface,[2] and the other for remote-handled TRU (RH-TRU) waste and the hotter high-level waste, 790 m below the surface (Weart, 1979; Sandia, 1977).

In March 1979, the Interagency Review Group formed by the Carter administration suggested that the WIPP be a candidate for commercial spent fuel, i.e., no distinction be made between commercial and defense radioactive waste. However, the then powerful House Armed Services Committee strongly opposed commercial waste disposal at the WIPP and regulation by the NRC. In response, Congress clearly defined the mission of the WIPP in December 1979 (Public Law 96-164) as a "...research and development facility to demonstrate the safe disposal of radioactive waste resulting from the defense activities and programs of the United States exempted from regulation by the Nuclear Regulatory Commission" (Figure 4-1).

4.2 Transuranic Nuclear Waste

4.2.1 Current Waste Description

Wastes destined for the WIPP consist of laboratory and production materials such as glassware, worn-out equipment and tools, scrap metal and wood, disposable

1 About 60 percent of the wastes may be co-contaminated with other hazardous constituents as defined under the *Resource Conservation and Recovery Act of 1976* (Public Law 94-580 and subsequent amendments). For compliance with EPA regulations, hazardous waste constituents are no longer important because Congress exempted the DOE from demonstrating "no migration" of hazardous components above health concentration standards. However, the types of hazardous constituents are important for disposal permits with the State of New Mexico.

2 In 1984, the single level of the WIPP repository was built 655 m below the surface.

- **1966 - Jan:** B52 collides with refueling tanker at 30,500 ft.; three nuclear weapons crash into Spanish soil; fourth weapon parachutes into ocean.[D1, D2] (Question raised as to how much plutonium contaminated soil to return to Savannah River Plant).

1970 AEC begins storing TRU waste above ground

- **1970 - Mar:** AEC directs TRU waste be stored retrievably at all DOE facilities rather than disposed with low level waste.[D3] AEC conservatively bounds TRU waste as waste contaminates with transuranic radioisoptopes with activity greater than 10 nCi/g.[D2]
- **1975 -** ERDA removes WIPP from commercial repository program.
- **1976 -** Laboratory tests on TRU waste behavior and HLW packages initiated.[D4, D5]
- **1977 - Apr:** Carter announces plan to defer indefinitely reprocessing of commercial SNF.[D6] **Nov:** Although role of NRC at WIPP unclear, DOE tells NRC it plans to seek license to build and operate WIPP based on policy from Carter administration.

1978 SNL tests gas generation potential of TRU waste

- **1978 -** SNL raises concern about gas generation and contracts with Los Alamos to do laboratory tests.[D7] **Oct:** Deutch (Massachusetts Institute of Technology [MIT] chem. prof.) report written for DOE recommends (1) disposing TRU waste at WIPP without planning for retrieval, and (2) demonstrating spent nuclear fuel (SNF), HLW, and TRU disposal at WIPP.[D8, D9] DOE Deputy Sec. J. O'Leary presses on with 2nd recommendation until 1979 enabling law for WIPP as a way to satisfy California law banning nuclear power plants until SNF disposal demonstrated.[D10] **Nov:** Lab studies focus on titanium alloys for HLW canisters.[D11]

1979 Draft EIS on WIPP has option for commercial SNF

1979 Congress passes bill to build WIPP for TRU waste only

- **1979 - Mar:** Interagency Review Group (IRG), formed by President Carter (in response to Deutch report), recommends disposal of SNF, HLW, and TRU in mined geologic repositories in final report.[D12] Report also suggests making WIPP candidate for commercial SNF repository. **Apr:** DOE defines project as a combination military/commercial repository in Draft EIS.[D13, D14] **May:** House Armed Services Committee cuts WIPP funding in response to DOE's expansion of the project to a repository for commercial SNF and thus requiring NRC licensing. **Dec:** Congress defines mission[D15] of WIPP:
  - sets up WIPP as a research and development facility for disposal of only TRU radioactive waste from DOE facilities
  - exempts WIPP from NRC licensing
  - requires DOE to sign a "Consultation & Cooperation" (C&C) Agreement with NM.

  Based on salt permeability tests in AEC-7 well, DOE cancels all gas generation and some backfill experiments. EPA defines TRU waste as waste with activity greater than 100 nCi/g.[D16]
- **1980 - Feb:** Carter orders SNF reprocessing to stop.[D17] **Mar:** Carter rescinds 1980 funds for WIPP because unlicensed repository for defense waste only and announces interim strategy to set aside money for other waste disposal projects at WIPP.
- **1980 - (con't) Jul:** House Armed Services Committee disagrees with Carter policy; therefore, rescinded funds are returned to WIPP in mid year. **Oct:** DOE issues final EIS eliminating SNF & HLW disposal and thereby reinstates WIPP mission defined by Congress in 1979.[D18]
- **1981 -** Tests begun in nearby potash mine Mississippi Chemical Mine Co., to evaluate corrosion of potential waste canisters and overpack alloys.[D19] **Jun:** WIPP Project Mgr. McGough rekindles disagreements between DOE and New Mexico by stating HLW could be placed by 1983 and remain during the operating phase of WIPP. Draft (but not final) Nuclear Waste Policy Act (NWPA) defines TRU waste as waste contaminated with transuranic radioisotopes with half-life greater than 20 yr and activity greater than 100 nCi/g.[D20]
- **1985 - Apr and Oct:** SNL turns on heat for simulated DHLW canister experiments. **Sep:** In 40 CFR 191, EPA defines TRU waste as waste with activity greater than 100 nCi/g and half-life greater than 20 yr.
- **1987 - Dec:** Nuclear Waste Policy Amendment Act (NWPAA)[D21] selects Yucca Mt., NV, to undergo site characterization for potential SNF and HLW disposal; (bedded salt not being considered for SNF and HLW).
- **1988 - Dec:** DOE abruptly cancels SNF and HLW experiments at WIPP because of NWPAA (no funds available to remove simulated disposal containers).

1989 Draft supplemental EIS identifies gas generation as issue

1989 Berlin wall falls

- **1989 - Mar:** SNL completes report to support Draft Supplemental EIS; report identifies generation of gases from container and waste corrosion as issue because salt permeability factor of 1000 lower than thought in 1979. Based on initial analysis results in February, DOE funds SNL to conduct new studies of gas generation.[D22, D23, D24, D25, D26, D27] **Nov:** Berlin Wall falls signaling the end of the Cold War and greatly changing future demands for nuclear weapon material and, thus, amount and composition of TRU waste going to WIPP.
- **1990 -** SNL and Westinghouse complete report on the pilot test phase of WIPP[D28] suggesting that 0.5% of waste capacity be brought to WIPP for gas generation experiments.

1992 NAS questions need for in situ gas generation tests

- **1992 -** SNL & Westinghouse complete work necessary to modify Test Phase Plan for gas generation tests.[D29] Westinghouse completes work necessary for modifying Waste Retrieval Plan.[D30] **Jun:** NAS WIPP Panel sends letter to DOE questioning scientific need for in-situ waste tests at WIPP.[D31]

1993 DOE decides not to test waste at WIPP

- **1993 - Oct:** DOE concurs with NAS and decides not to emplace waste in a pilot phase at WIPP – lab tests instead.[D32] DOE decides to make draft Compliance Certification Application (CCA) to EPA. Because actual waste not coming to WIPP, "bin tests" canceled.
- **1995 - Sep:** Gas generation studies completed and results used to establish rates for CCA.[D33, D34] **Dec:** DOE publishes updated revision of WIPP inventory.[D35]

1996 Tests on solubility reported

- **1996 -** Tests on solubility reported for use by CCA.

TRI-6342-5839-2

Figure 4-1. Development of project goals and characterization of waste properties.

laboratory clothing, cleaning rags, spent solvents that were sorbed or solidified, and solidified sludges, which have been contaminated with TRU radioisotopes. Any waste contaminated by alpha-emitting TRU radionuclides with half-lives greater than 20 yr and activities greater than 100 nCi/g, is considered TRU waste in the United States.[3] Most waste with less than this activity per mass is considered low-level waste.

The DOE classifies TRU waste as either contact-handled (CH) or remotely handled (RH) based on dose at the surface of the waste container. If this surface dose is less than or equal to 200 mrem/hr, the waste is defined as CH-TRU; if the dose is greater than 200 mrem/hr, the waste and its container are defined as RH-TRU (DOE, 1990c). The WIPP Land Withdrawal Act (LWA), passed in 1992 (Public Law 102-579), ratified previous agreements between the State of New Mexico and the DOE on the volume of CH-TRU in the original Consultation and Cooperation Agreement, signed in 1981, and was the first modification of this Agreement concerning the RH-TRU content as signed in 1984 (see Figure 2-2). That is, the total combined volumes of CH-TRU and RH-TRU waste were not to exceed 6.2 million cubic

3 TRU waste is also known as Alpha-Bearing or Intermediate Level Waste in other countries. The minimum defining activity varies between 0.3 and 1000 nCi/g.

feet, and the emplaced RH-TRU waste was not to exceed a total activity of 5.1×10^6 curies or a total activity concentration of 23 curies per liter (averaged over the volume of the canister). Only RH-TRU waste producing a dose of less than or equal to 1000 rem/hr is eligible for disposal at the WIPP (DOE, 1996a). Furthermore, no more than 5% of the emplaced RH-TRU waste was to exhibit a dose in excess of 100 rem/hr. While the capacity for CH-TRU waste is adequate for all of the DOE's TRU waste produced after 1970, the legal limit is too small to include the CH-TRU waste buried prior to 1970 (Weart, 1979; DOE, 1979; 1980a). The final amount of RH-TRU in the DOE complex is not known at this time because it depends on how some future waste will be treated and packaged.

4.2.2 Modification of TRU Waste Definition

As noted in Section 2, the definition of TRU waste as a material contaminated with TRU radioisotopes with an activity of 10 nCi/g was quickly proposed and used as a temporary but safe bound by the AEC in 1970 (Perge, 1982). In its early attempts at nuclear waste standards, the EPA, through simple calculations on resuspended plutonium in soil, concluded in 1979 that a limit of 100 nCi/g would keep doses below 500 mrem/yr and thus increased the bound by an order of magnitude (Sjoblom, 1982). In 1980 and 1981, the early legislative drafts of the Nuclear Waste Policy Act (NWPA) of 1982 adopted this activity threshold for defining TRU waste but also added a half-life threshold of 5 years. In 40 CFR 191, promulgated in 1985 by the EPA, this latter threshold was increased to a 20-year half-life (EPA, 1985a). The change in the definition by the EPA required that the DOE reclassify ~20% of the TRU waste that existed between 10 and 100 nCi/g. The activity of most waste was around 10^4 nCi/g (Smith, 1982; WIPP PA Division, 1991/1992, Vol. 3).

4.2.3 TRU Waste Inventory

For preliminary performance assessments between 1989 and 1992, waste activities and mass were estimated from the Integrated Data Base (e.g., IDB, 1990), produced annually by ORNL for the DOE, and from requests for supplemental information by Westinghouse directly to the waste generator sites (see for example, WIPP PA Division, 1991/1992, Vol. 3). For the CCA, the DOE assembled a baseline inventory, referred to as the Transuranic Waste Baseline Inventory Report (TWBIR) (DOE, 1995b). This baseline included estimates of the radioisotope inventory for 569 stored or to-be-generated waste streams for CH-TRU waste and over 400 waste streams for RH-TRU waste. Waste-related input parameters for the CCA analysis were then developed from this data.

4.3 Waste Characterization Studies

4.3.1 Initial EIS Waste Studies

Simultaneous with site characterization, Sandia began tests in early 1977 on the behavior of TRU waste forms and high-level waste (Molecke, 1978). These tests were to be used by Westinghouse, the technical support contractor for the WIPP, to develop waste acceptance criteria for the various forms of the waste (Sandia, 1979). To ensure waste acceptability, in 1976 the WIPP Project had sent guidance to the generator/storage sites that the waste be incinerated to remove combustible and organic material. In the late 1970s and early 1980s, the major generators and storage facilities that were expected to ship directly to the WIPP were Idaho National Engineering and Environmental Laboratory; Hanford Reservation, Washington; Rocky Flats Plant, Colorado; Los Alamos National Laboratory, New Mexico; Savannah River Plant, South Carolina; Oak Ridge National Laboratory, Tennessee; and the Nevada Test Site. Other generators or storage sites such as Lawrence Livermore National Laboratory, California; Mound Laboratories, Ohio; Argonne National Laboratory, Illinois; Bettis Atomic Power Laboratory, Pennsylvania; and other smaller sites would either continue to ship their waste to the major storage sites for certification or begin their own certification programs and ship directly to the WIPP. By 1979, results from initial tests indicated that most characteristics of the TRU waste were acceptable and so incineration was no longer recommended (Sandia, 1979; Weart, 1979). The conclusions reached included the following (Hunter, 1979):

1. Gas generated through microbial degradation of organic material or anoxic corrosion of waste containers was possible but the gas would readily dissipate as a result of the anticipated permeability of the salt, even when the highest gas-generation rates were used, as summarized by Molecke (1979).

2. Combustibility was not an issue because spontaneous internal ignition was not possible. Thus the only concern was from fires from other sources during operations.

3. Immobilization was unnecessary because most characteristics that would be affected did not present a difficulty with regard to long-term disposal. Specifically, (a) leachability was not an issue (analysis assumed a waste leachability equivalent to salt dissolution with no adverse effects), (b) organic ligands that might reduce sorptive capacity of the host salt and Culebra dolomite were not an issue (analysis showed peak releases occurred at 10,000 yr rather than 100,000 yr but otherwise no effect), (c) use of sorbing materials might be desirable but was not necessary (analysis indicated improvement if a further safety margin was necessary); and (d) only sludge wastes were considered unacceptable (analysis had not been done to evaluate whether sludge waste with 60% to 80% moisture by weight would be acceptable in a dry repository).

4. Criticality was not an issue, because analysis by Los Alamos indicated criticality was not possible without dissolution and reconcentration of fissile isotopes in the Culebra and reconcentration appeared unlikely based on analogy with the natural reactors at Oklo (qualitative argument). (Criticality was formally examined and screened out for the 1996 PA; see Section 6.6.)

4.3.2 Container Corrosion Studies

Another conclusion reached (Hunter, 1979) was that corrosion of waste was not an issue, because long-term integrity was not considered necessary in a salt repository. Waste drums were thought to be adequate to facilitate retrieval in the first 25 years. Gas generation potential was not important in a dry repository and was of the same order of magnitude as microbial generation in a wet, anoxic environment from interstitial brine migration because of hot, high-level waste.

Later, Sandia would conduct a few tests with waste drums to evaluate their expected life, but the first corrosion studies were on candidate materials for canisters and disposal containers for high-level waste, where long-term containment might be desired. In 1981, Sandia set up a field test at a nearby potash mine owned by Mississippi Chemical Mine Co. Besides testing a few materials, the test provided experience with instrumentation and sampling techniques. In 1986, Sandia agreed to help Savannah River Laboratory test the behavior of its high-level waste canisters in a salt repository. Seven other countries asked to join, resulting in the Materials Interface Interaction Test (MIIT) project, which tested over 50 combinations of materials in Room J of the experimental region and continued for the next 5 years (Wicks et al., 1993).

4.3.3 Characterization for Performance Assessment Studies

Based on the characterization studies for the EIS, the DOE decided in 1980 that further studies of waste characteristics such as gas generation were unnecessary. Furthermore, after passage of the 1987 amendments to the Nuclear Waste Policy Act (Public Law 100-203), most experiments on high-level waste, degradation of container materials, and influence of heat were stopped. However, the decision not to perform waste characterization studies was reconsidered after new data were presented in 1989. The new data included estimates from Brush and Anderson (1989) that anoxic corrosion of steel in the waste, if brine were available in the repository, could produce significant quantities of hydrogen gas in addition to any gas produced from microbial degradation of cellulosic material, e.g., wood, in the waste. Second, and as noted previously, in situ measurements of permeability in the Salado, taken while resolving the issue of brine flow into the repository, were found to be 1000 times smaller than those measured earlier (Nowak et al., 1988). An analysis conducted by Sandia (Lappin et al., eds., 1989), using Brush and Anderson's assumptions and the new permeability measurements, demonstrated that gas could potentially influence the overall performance of the repository.

Based on initial analysis, the DOE funded Sandia to conduct laboratory studies of gas generation in February 1989. These studies predominately comprised single-process experiments on anoxic corrosion, microbial activity, and radiolysis, most with nonradioactive simulated CH-TRU waste (Brush, 1990; Brush et al., 1991a, 1991b; 1993; Reed et al., 1993; Telander and Westerman, 1993; Francis and Gillow, 1994). The experiments were completed by September 1995, and the observed gas generation rates were used in the performance assessment for the CCA (Francis et al., 1997; Telander and Westerman, 1997).

Initially, a full-scale pilot phase for the repository had been envisioned by the DOE and Congress. But by the 1980s, the EEG and groups opposed to the WIPP Project saw the pilot phase as a way for the DOE to open the WIPP without certifying that it complied with regulations. They argued that institutional inertia would be so great that the WIPP would remain open, even if it

could not comply with all EPA regulations during the transition from the pilot phase to the operational phase. To justify a pilot phase, the DOE proposed tests to reevaluate gas generation using actual TRU waste in the repository. In 1990, Sandia and Westinghouse presented operational and technical arguments for placing 0.5% of the waste capacity of the WIPP in the repository (DOE, 1990b). To resolve the issue between the DOE and opposition groups, Congress, in the 1992 WIPP Land Withdrawal Act, required the WIPP Panel of the NAS to certify a scientific need—not just operational usefulness—for the tests. The WIPP Panel did not see a scientific need for in situ tests, because the data could be obtained by laboratory tests, and so recommended in June 1992 that tests with actual waste be conducted in laboratories instead (NAS/NRC, 1992). The DOE concurred in October 1993 (Anonymous, 1993). Without a pilot phase, the DOE decided to accelerate into the compliance phase for the WIPP in October 1993, halted most of the in situ experiments, and then completely closed the in situ experimental area in October 1995.

As described in Section 8.4, an area that remained active was related to the solubility of actinides, primarily plutonium and americium, which had been shown to be important in the preliminary performance assessments (Helton et al., 1993a; 1996) along with shear strength of the waste, which influences releases after direct intrusion by exploratory drilling. The solubility data for the radioisotopes modeled in the 1991 and 1992 PAs were developed by an external expert panel, which had been convened in early 1991 (Trauth et al., 1991; 1992). At that time, experts from several disciplines were used to determine radioisotope solubility in the WIPP repository environment, particularly because disparate data from various studies had not yet been synthesized in the literature and direct experiments had only just started. The panel examined existing data and developed probability distributions of solubilities to express the uncertainty thought to pertain at the WIPP repository.

The solubility experiments initiated used one actinide (either Am, Np, Th, U, or Pu) for each of four possible oxidation states (+III, +IV, +V, +VI) at various pH and brine compositions. Behavior of other actinides with an equivalent oxidation state was assumed to be similar, e.g., the measured solubility of Pu(III) was assumed to be similar to the solubility of Am(III). The influence of organic ligands on solubility was also studied experimentally. However, the solubility of actinides in high ionic strength brines was modeled at Sandia using FMT, as described in Section 7.4.3. The experimental and modeling results were then combined to produce a distribution of solubility from which to sample for the 1996 PA calculations in support of the CCA. Then, the sampled dissolved concentration was combined with estimates of the amount of humic, microbial, mineral, and intrinsic colloidal concentration of actinide species to produce an "effective solubility" (Figure 4-2).

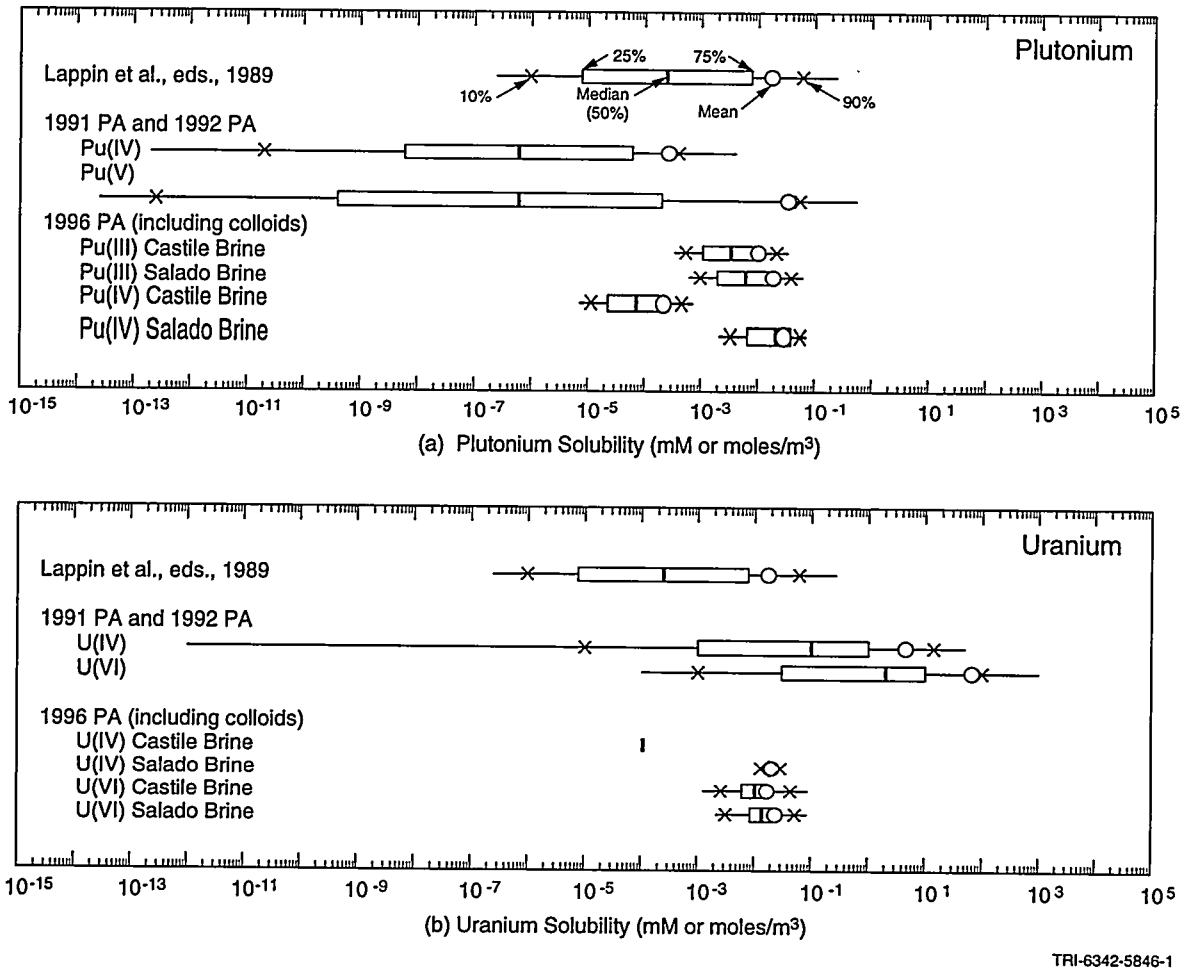


Figure 4-2. Solubility of radioisotopes used in calculations between 1989 and 1996 for (a) plutonium and (b) uranium.

5.0 Design, Construction, and Behavior of Repository

The repository (or underground facility) of the WIPP disposal system is 655 m beneath the surface at an elevation of 384 m above mean sea level. The room layout is essentially unchanged from the conceptual design Sandia proposed in 1977 (Sandia, 1977), although room dimensions were reduced and space between rooms was increased in 1980. In situ experiments also helped to refine constitutive equations in numerical models in the mid 1980s and to evaluate concepts such as backfilling the repository and estimating potential amounts of brine inflow to the repository in the late 1980s and early 1990s (Tyler et al., 1988).

5.1 Design of Repository

5.1.1 Early Conceptual Design

Concurrent with site selection and writing the EIS, Sandia prepared a conceptual design of the WIPP surface and underground facilities. The conceptual design was completed in 1977 (Sandia, 1977). The basic design drew on information gained from experience with nearby potash mines in the Delaware Basin. However, the extraction ratio was dramatically reduced to ~33% overall (~40% in any one disposal panel) to increase room stability and mine safety. The design for the transport containers and other hardware was based on Sandia's experience with equipment design at the Nevada Test Site.

The disposal area for the CH-TRU waste was set at 373 m elevation (666 m depth) and consisted of eight closely spaced panels 28 m apart with eight rooms in each panel, 34 m apart. Each room was 5 m high, 14 m wide, and 137 m long (Figure 5-1a). In the rooms, the CH-TRU waste was to be stacked four barrels high. The experimental area and disposal area for the high-level waste (RH-TRU waste, high-level waste, and spent nuclear fuel) was set at a 200 m elevation (823 m depth), below the Infra-Cowden anhydrite strata near the base of the Salado. The repository had five shafts.

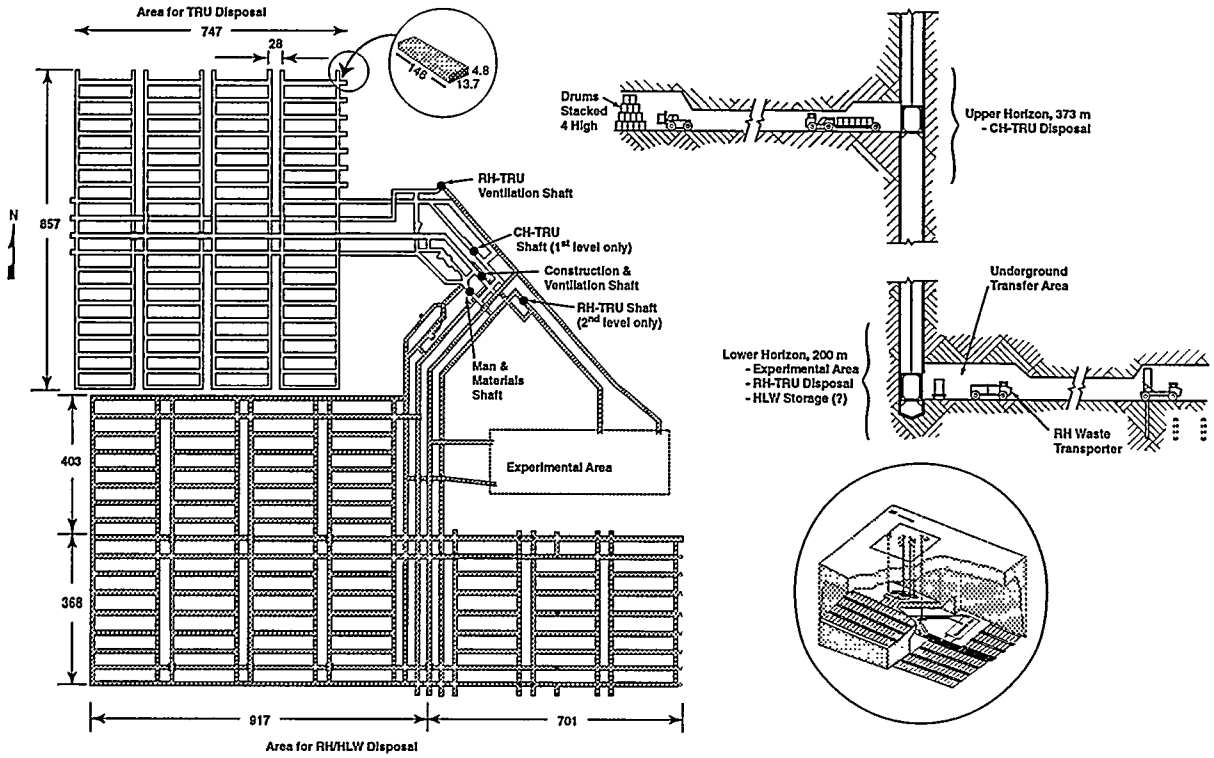
The initial conceptual design anticipated disposal of 105,000 m^3 of CH-TRU waste retrievably stored since 1970 and accommodations for about 7000 m^3 of CH-TRU waste produced annually. The design also anticipated disposal of 2500 m^3 of RH-TRU retrievably stored since 1970 and accommodations for about 200 m^3 of RH-TRU waste produced annually. The volume of high-level waste was sufficient for about 1000 assemblies. Although expansion was contemplated, space for CH- and RH-TRU waste buried prior to 1970 and TRU waste produced from future decontamination and decommission of facilities was not included in the initial design. In December 1979, Congressional legislation limited the WIPP to the disposal of defense TRU waste. The second level, containing the experimental area and the RH-TRU waste disposal area, was abandoned and the repository was reconfigured for the final EIS as a single-level facility with four shafts (DOE, 1980a) (Figure 5-1b). RH-TRU canisters were to be placed in the repository walls, and CH-TRU drums would be placed in the large repository rooms. This concept was tested in Rooms J and T of the experimental region in the mid-1980s (Figure 5-1b) (Matalucci, 1988).

5.1.2 Detailed Design

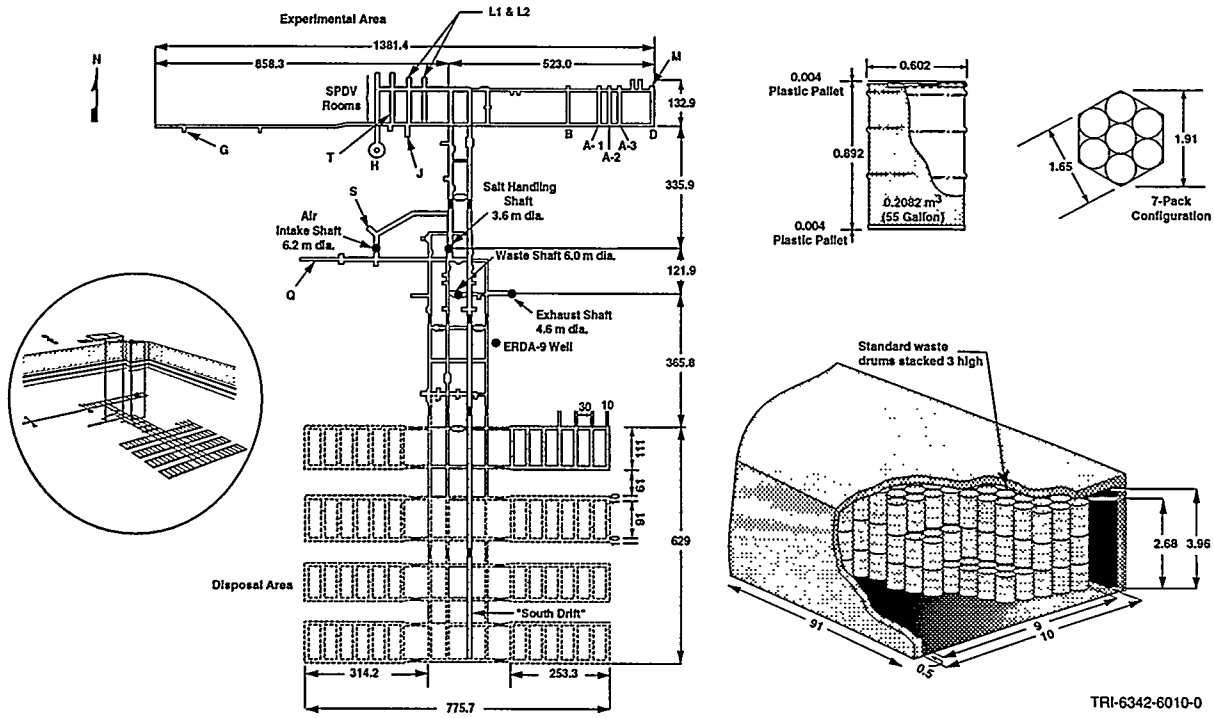
In 1978, the DOE contracted with Bechtel National, Inc., to be the architect/engineer for the WIPP Project (and with Westinghouse Electric Corporation, for technical support to the DOE and Sandia) (Figure 5-2). The preliminary design of the WIPP (or "Title I design" in the language of the DOE orders) began in July 1979 and moved to a detailed design ("Title II") phase by February 1981. The detailed repository design included a waste disposal area, an experimental area, and an operations area around the shafts.

The underground operations area was designed to service and maintain underground equipment for mining and disposal operations, monitor for radioactive contamination, and allow limited decontamination of personnel and equipment. The experimental area consisted of a series of rooms north of the operations area and shafts. Bechtel's initial design placed the disposal area for the CH-TRU waste north of the experimental rooms.

The disposal area provided enough space for the disposal of the projected 176,000 m^3 of waste and consisted of eight panels spaced 61 m apart with eight rooms in each panel, 30.5 m apart. Each room was 4 m high, 10 m wide, and 91 m long. The detailed design envisioned that waste would be emplaced in all drifts, including those that connected rooms, and not just in the rooms as shown in the conceptual design. Also plans



(a) 1977 conceptual design



(b) Final construction plans for WIPP repository

Figure 5-1. Designs for the WIPP repository, including (a) 1977 conceptual design (Sandia, 1977) and (b) final construction plans for WIPP repository (Rechard, 1995).

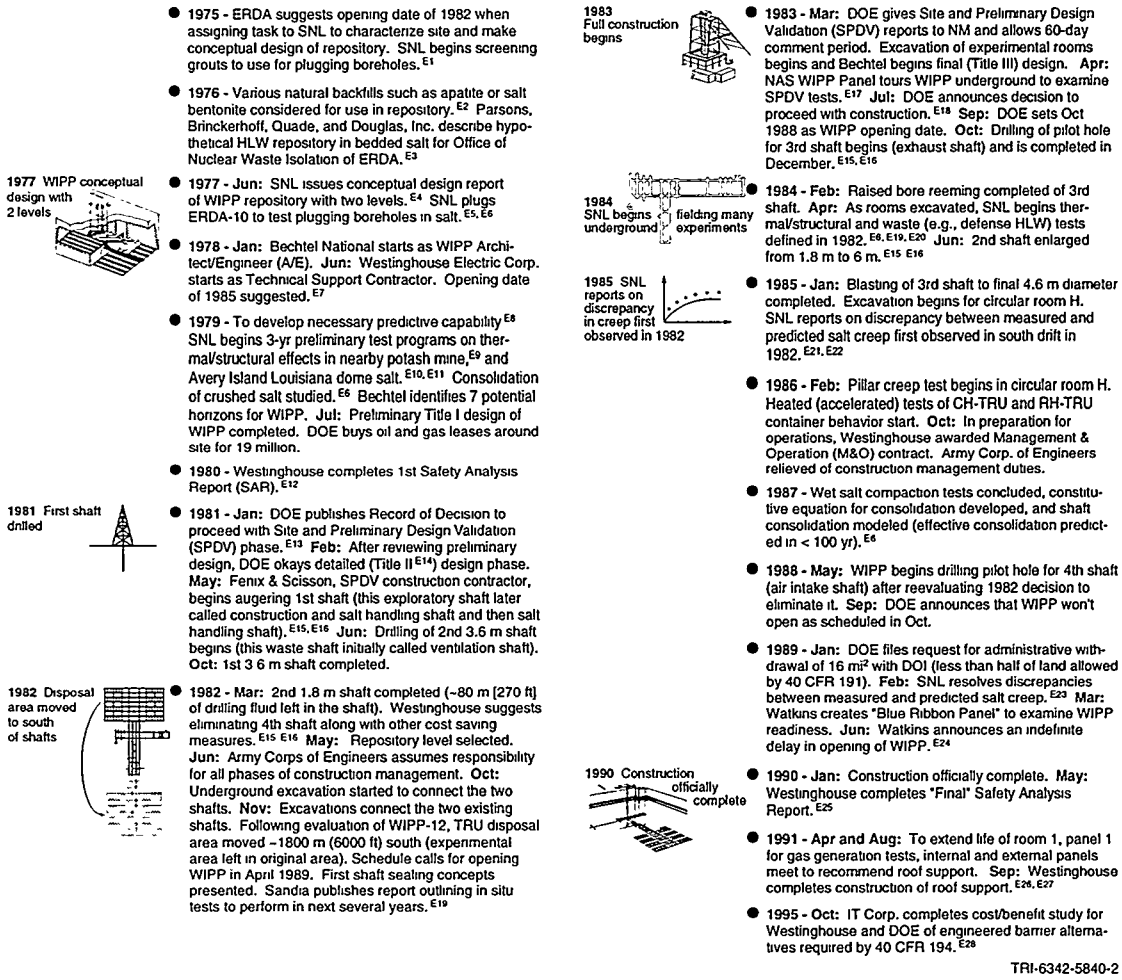


Figure 5-2. Events associated with design, construction, and modeling of the WIPP repository.

indicated that waste would fill the center section, where the panels joined the main drifts; this section was divided into southern and northern equivalent panels, bringing the panel total to 10. The extraction ratio for a panel was similar to the conceptual design (38%) but the overall extraction ratio of the disposal area was decreased further to 22%.

In March, Bechtel began final design ("Title III") of the repository, and a design report was published in 1986 (Bechtel, 1986). Some modifications were necessary after the final report was published concerning the backfill (discussed in Section 5.4 and drum emplacement. In the design submitted for the CCA, the TRU waste is to be placed in 55-gallon drums, grouped as 7-packs, 3 drums high (Figure 5-1b). The change from a 6-pack drum grouping to a 7-pack was in accordance with container modifications as they evolved from the initially proposed TRUPACT I transportation container, whose design was rejected by the EEG in 1985 (Channell et al., 1986), to the TRUPACT II container certified by the NRC in August 1989. (According to the 1987 modification to the Consultation and Cooperative Agreement between the DOE and the State of New Mexico, the NRC had jurisdiction over transportation of nuclear waste to the WIPP.)

5.2 Construction of Repository

After publication of the final EIS in 1980 and a record of decision in January 1981, the DOE began the Site and Preliminary Design Validation (SPDV) program. The program further characterized stratigraphy near the repository and validated that rooms and tunnels, which were using dimensions specified in the preliminary design, would remain stable during the emplacement period. As part of the SPDV program, Fenix & Scisson, Inc., Sandia's contractor for the conceptual design and DOE's SPDV contractor, drilled two shafts,

the 3.6-m salt-handling shaft and the 2-m waste shaft, in May and June, 1981. The first shaft, 3.6-m in diameter, was initially called the exploratory shaft, then renamed the construction and salt handling shaft, and finally referred to as the salt handling shaft. The second shaft drilled, the 1.8-m ventilation shaft, was later renamed the waste shaft (DOE, 1996a).

In June 1982, the Army Corps of Engineers assumed responsibility for managing the SPVD and subsequent construction. Once the second shaft was completed in March 1982 and the repository level selected, excavations were begun in October to connect the two shafts, which were completed by the end of November. In the second half of the year, four full-sized disposal rooms were excavated. Although Bechtel's initial repository design placed the waste disposal area north of the experimental area, the disposal area was moved south of the shafts at the end of 1982 after the discovery of a brine reservoir in WIPP-12. Shortly thereafter, a drift to the south end of the disposal area was excavated to confirm the stratigraphy and manual measurements were taken frequently to evaluate deformation of the drift.

In March 1983, the results from in situ experiments in the SPDV rooms were reported (Tyler et al., 1988). Sandia helped analyze data from these and other geologic field activities and from geomechanical instrumentation to determine whether the design criteria were suitable and to confirm the reference design for the underground opening (Tyler et al., 1988). Also that month, excavation began on rooms for Sandia's geotechnical experiments.

The WIPP Panel of NAS toured the underground excavation for the first time in April 1983 to examine the SPVD rooms. Four months later, DOE announced its decision to proceed with full construction of surface facilities and continued excavation of the underground facility. The pilot hole for the third shaft, the exhaust shaft, was drilled in September 1983. Reeming of the pilot hole to 3.6 m from the underground to the surface, and then blasting to the final 4.6 m diameter was finished by January 1985 (DOE, 1996a). Between April and June 1984, the waste shaft diameter was enlarged from 1.8 to 6 m. In 1988, a fourth shaft was added for increased air circulation (Figure 5-2). A pilot hole was first drilled from the surface and then reemed to a final 6.2 m diameter from the underground to the surface.

By the end of 1988 the surface and underground facilities were essentially complete; they were declared officially complete in January 1990. Hence, as now built, the underground is connected to the surface by four vertical shafts: the waste shaft, the salt handling shaft, the exhaust shaft, and the air intake shaft (Figure 5-1b). All shafts except the exhaust shaft have permanent hoists capable of moving personnel, equipment, and materials between the surface and the repository. Every shaft will eventually be backfilled as described below.

5.3 Experiments and Model Development of Repository Behavior

Early in the history of the WIPP Project, the DOE, with Sandia as the lead national laboratory, viewed the WIPP as a research and development facility. The project was to conduct experiments that supported model development of salt creep and experiments on backfilling and sealing the repository, brine movement, and areas of interest to high-level waste disposal concepts such as canister material behavior (discussed in Section 4.3.2).

5.3.1 Experiments with Salt Creep Behavior

Sandia began to build a salt creep laboratory in 1974. Testing on specimens from mines and salt domes was in progress by 1975, and creep in salt from ERDA-9 cores was studied in 1977. Sandia initiated a 3-year program in 1979 to evaluate, through in situ and laboratory experiments, salt deformations around mine openings and the effects of heat on acceleration of salt creep. The in situ experiments were conducted in a nearby potash mine owned by the Mississippi Chemical Company and at the Avery Island salt dome in Louisiana. Sandia proposed that data from these and future in situ experiments be compared with predicted behavior of the underground openings, thus partially validating the constitutive salt creep models and the numerical methods, which had been developed with model parameters obtained solely from laboratory experiments.

Laboratory creep tests were started on larger specimens from the underground workings of the WIPP in 1982; in situ salt creep manual measurements at the WIPP also began in 1982 as the main drift through the disposal area, the "south drift," was excavated. Sandia began fielding more extensive in situ salt creep experiments in 1984 as experimental rooms were completed. To measure accelerated salt creep, heat in simulated high-level waste canisters was turned on in these rooms in 1985. However, when Congress decided in 1987 to

characterize only the Yucca Mountain site in Nevada as a commercial spent-fuel and high-level waste repository (Public Law 100-203), the DOE canceled the simulated high-level waste experiments in Rooms A1, A2, A3, and B at the WIPP (Figure 5-1b) (Matalucci, 1988), which had been supporting a potential commercial repository elsewhere in bedded salt.

5.3.2 Modeling of Repository Behavior

The first modeling efforts in 1975 reviewed empirical constitutive creep laws developed during Project Salt Vault and numerical modeling capabilities available in the mining industry. Some of the first calculations, completed in 1978, evaluated a potential concern that hot canisters would become buoyant in the plastic salt and move significant vertical distances (Dawson and Tillerson, 1978). Sandia modeled the repository, with Sandia-developed codes (SANCHO, Stone et al., 1985) and constitutive laws (Tyler et al., 1988), and used the results in 1980 (Krieg et al., 1981) to examine the reasonableness of various modifications to the room dimensions proposed in the 1977 conceptual design, which had been based on experience with the area's potash mines. Predictions from other numerical codes (e.g., SPECTROM-32, Callahan et al., 1989) were compared extensively to test data in 1980, and more calculations on predicted room deformation were conducted in 1982 and 1985 (Morgan et al., 1985).

Although, from a practical standpoint, the predicted and measured values of salt creep were close, the manually measured salt creep in the south drift in 1982 and the automated measurements of the SPDV rooms in 1982 and 1983 and various other experimental rooms (Rooms A1, A2, A3, B1, G, and H [Matalucci, 1988]) (Figure 5-1b) were nevertheless about three times greater than predicted values (Morgan et al., 1985; 1986). Thus between 1985 and 1989 an alternate conceptual model and mathematical expression were incorporated into codes and tested. Among other adjustments, the newer expression specified the combination of vertical, horizontal and lateral stresses at which the salt yielded and began to creep, using a Tresca yield surface rather than a von Mises yield surface and steady-state analysis. Thereafter, agreement of predictions with in situ measurements was excellent (Figure 5-3) (Munson et al., 1989). The partially validated codes, SANCHO and its improved version, SANTOS (Stone 1997), were used to evaluate the reduction in porosity as the room closed by means of creep over time. The results were then used for the performance assessments conducted in the 1990s.

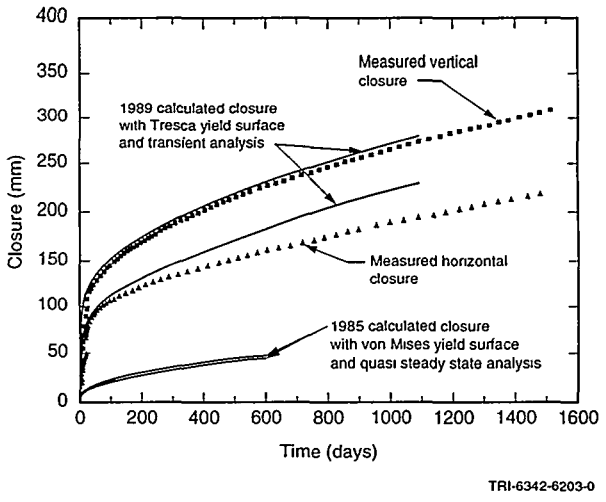


Figure 5-3. Comparison of measured and calculated creep closure in Room D (after Munson et al., 1989, Figures 3-5 and 3-6).

5.4 Repository Backfill

5.4.1 Use of Backfill in Disposal Area

The DOE had considered the use of backfill in the disposal area from the time of the initial conceptual design (Sandia, 1977). In 1976, some thought was given to placing sorptive minerals such as apatite or bentonite around the drums to sorb radioisotopes if enough brine were present to corrode the drums (Barr and O'Brien, 1976). More importantly, it was assumed that backfill would be emplaced in the repository to help fill the void space and reduce the magnitude of subsidence in overlying units, in addition to mitigating any potential risk of underground fire propagation (DOE, 1980a; 1980b). Backfill experiments were conducted in Room J of the experimental region (Matalucci, 1988) (Figure 5-1b).

Although backfill in the disposal area was considered part of the baseline design for the repository (Sandia, 1977; DOE, 1980a), as reported in the supporting documents for the Supplemental EIS (Lappin et al., eds., 1989) and the four preliminary performance assessments through 1992, the need for the backfill to mitigate subsidence and fire propagation diminished during the 1980s. These findings were formally reported in 1990 in the Safety Analysis Report (DOE, 1990a), which concluded that fire propagation in the waste disposal region was unlikely even without backfill, and in a 1994 Westinghouse study (WEC, 1994), which indicated that the addition of backfill would have a negligible impact on the subsidence of overlying units.

For the 1996 PA in support of the CCA, however, backfill was reconsidered and again included in the design. A chemical backfill of MgO was proposed that would combine with any microbially produced CO_2 so that brine present in the repository would not become acidic, thereby increasing the solubility of the actinide radioisotopes. The backfill was not essential to show compliance but was an engineering measure that assured compliance and thereby met the assurance requirements of 40 CFR 191.

5.4.2 Shaft Backfill and Sealing Methods

Because of the problem presented by the presence of boreholes in the vicinity of the abandoned Lyons, Kansas, site, experiments on borehole plugs were immediately pursued by the WIPP Project in 1975. By 1977, three grouts had been selected and tested by plugging ERDA-10 drilled south of the WIPP site (near the Gnome site) (see Figure 3-1). In 1979, an experiment in AEC-7 tested the ability of a plug to withstand the 12.7-MPa pressure of the Ramsey Sands aquifer in the Bell Canyon Formation (Tyler et al., 1988).

Initial concepts for backfilling shafts were described and the first laboratory tests on compacting crushed salt conducted in 1982. Sandia presented the first conceptual design for shaft backfill in 1984, which continued to evolve during the late 1980s. Crushed salt, the primary backfill (usually referred to as "seals" in the WIPP Project) for the shafts through the Salado, was expected to limit the creation of a preferred pathway for radioisotope migration. Sandia developed a machine to build salt bricks from crushed salt in 1986 such that portions of the salt backfill in the drifts and shafts would be compacted to ensure adequate densities. Studies were conducted and reported in 1987 to estimate the density of reconsolidated salt under the lithostatic pressure from creep closure of the shaft (Nowak and Stormont, 1987). This study suggested 95% of intact salt densities could be obtained in the lower portions of the shaft in less than 100 yr, provided some brine (~2.5 %wt) was added to the crushed salt. Compaction of the salt through tamping was proposed to obtain a highly dense backfill during emplacement.

Stopping brine flow to the salt backfill from aquifers in the Rustler and upper units was also thought necessary because significant volumes of brine might delay or even prevent consolidation of the crushed salt. The first shaft sealing concepts near the Rustler envisioned using large seals of concrete, concrete-grout, or possibly other mixtures directly below the Rustler and halfway down the salt column to protect the lower crushed salt component prior to consolidation (Stormont, 1988; Lappin et al., eds., 1989, Figure 4-10). In 1990, bentonite clay, a swelling clay shown to be stable and with low permeability in brines in 1979 and 1984 studies, was added as a separate long-term component to the seals in the Rustler and at other seal locations in the Salado. Thus, the 1990 reference design for the shaft backfill included concrete plugs as a short-term component and bentonite clay and compacted salt as long-term components of the seals (Nowak et al., 1990). Details of the various options were developed in 1993 (Van Sambeek et al., 1993). As mentioned in Section 3, an important aspect of the backfill program between 1984 and 1988 was evaluating permeability and brine flow around the repository, but small in situ tests of seals in Rooms L1, L2, and M of the experimental region (Matalucci, 1988) (Figure 5-1b) were also conducted. In the 1990s, a more complete testing program was begun to demonstrate and develop confidence in the sealing concepts of the backfill, and asphalt was added as a long-term component.

5.4.3 Panel Sealing Methods

Initially, the repository design did not include constructed barriers to separate the waste into modules. But when the WIPP Panel of the NAS expressed concern that fire in combustible portions of the waste could pose a hazard to mine workers, barriers throughout the disposal area were proposed to enhance mine safety. By 1987, barriers between each panel (Figure 5-1b) were considered the best of several options for balancing concerns about safety, cost, and mine operations (constructing closures is both expensive and time-consuming) (Argüello and Torres, 1988). Shortly thereafter, plans called for isolating individual panels over the long term, with 30- to 40-m-long seals composed of preconsolidated salt and large concrete plugs at each end (Lappin et al., eds., 1989; Nowak et al., 1990). However, during the 1990s, the value of panel seals was questioned because of their limited ability to contain gas, given the presence of the disturbed rock zone that would develop in the anhydrite interbeds directly above and below the repository. Some consideration was given in the 1990s to grouting the anhydrite interbeds or removing the anhydrite beds entirely to improve performance. Even though long-term isolation of radioisotopes could not be fully assured, in 1997, the EPA stipulated that the DOE was to use panel seals in its design as one of several conditions for certifying compliance of the WIPP (EPA, 1997; EPA, 1998).

6.0 Hazard Identification and Scenario Development

An important step in a performance assessment is to identify potential hazards that might disrupt the geologic disposal system. The first list of hazards for consideration at the WIPP was published in 1974 about the same time as site selection (Claiborne and Gera, 1974). The list was updated in 1979 for the EIS (Bingham and Barr, 1979, 1980), in 1989 for the preliminary performance assessments (Hunter, 1989), and in 1995-96 for the final performance assessment (Galson and Swift, 1995). Although the process of identifying hazards and then selecting specific hazards for modeling was relatively informal initially, with each iteration the process of identifying hazards became more rigorous. Engineering analysis can be conducted without hazard identification as an explicit step, but in performance assessment it is not only useful for the step to be formal but is also one aspect that sets it apart from small-scale analysis.

6.1 Description of Hazard Identification

6.1.1 Categories of Hazardous Agents

By the late 1980s, agents of hazards were typically categorized as features, events (i.e., short-term phenomena), and processes (i.e., continuous phenomena) or "FEPs" that act upon the system whereby a hazard might occur. The event category had already been used in the *Reactor Safety Study* of 1975 (Rasmussen, 1975), which inaugurated large probabilistic risk assessments. When this work was applied to a geologic disposal system in 1976, the definition of agents was broadened to include processes. Then in 1981, the IAEA formally considered "undetected features" (IAEA, 1981). Hence for assessing performance at the WIPP disposal system, hazard identification entails selecting *features* (e.g., a brine reservoir under the repository), *events* (e.g., humans drilling into the repository), and *processes* (e.g., generation of gas in the repository after waste disposal) relevant to repository functions. Many aspects of the characterization of the site and waste, the design of engineered components such as seals, the experimental programs, and the computer modeling capabilities were based on concerns about specific hazards.

6.1.2 Scenarios

In the WIPP Project, a *future* (or "elementary event" in the terminology of probability theory) is a hypothetical chain of physical events and processes, including particular features. A *scenario* (or scenario class) is a group of futures; it is sometimes represented by a key future with specific common attribute(s), e.g., human intrusion. Futures were grouped into various scenarios in the WIPP Project to focus attention on certain hazards, such as human intrusion. Grouping is feasible and practical if the probabilities of a FEP residing in the scenario class are easily calculated and if behavior within the scenario class is similar enough that a single, key future can characterize and represent the whole scenario class.

Given that a mathematical model of the disposal system exists, $R(\bullet)$, "future" and "scenario" can be precisely defined in terms of parameter space, $\mathbf{x}$, as follows. The parameter space $\mathbf{x}$ can be arbitrarily divided into two subsets: a subset defining conditions of scenarios, $\mathbf{x}^s$, and a subset of remaining model parameters, $\mathbf{x}^p$, i.e., $\mathbf{x} = [\mathbf{x}^s, \mathbf{x}^p]$. A future is one point in parameter space for scenarios, $\mathbf{x}^s$, and a scenario is a grouping of similar futures or subsets of $\mathbf{x}^s$, i.e., $S_j \subset \mathbf{x}^s$, where S_j is a scenario of what can occur to produce an unwanted outcome. For example, a scenario might be described as $S_j = \left\{x_1^s < a \text{ and } x_2^s > b\right\}$, where x_1^s and x_2^s defined conditions for human intrusion and location of a brine reservoir, respectively (Tierney, 1995). Characterization of scenarios and their uncertainty came to be known as stochastic uncertainty, $\mathbf{x}^s$, within the WIPP Project to distinguish it from the more general epistemic (or subjective) uncertainty associated with model parameters, $\mathbf{x}^p$.

6.1.3 Steps of Hazard Identification and Scenario Development

Hazard identification is a heuristic process, especially because analysts must identify potential hazards before they have developed any extensive experience with the system. This situation, in turn, suggests that continued reevaluation is necessary as the general inquiry about the disposal system continues, which has indeed occurred at the WIPP. For discussion, hazard identification and scenario development is assumed to consist of the following steps:

1. Identifying and listing the full scope of hazardous agents (FEPs) relevant to disposal system functions. The list is called the universe.

2. Selecting from this list, based on well-defined screening criteria, those FEPs that might reasonably contribute to contaminant releases as defined in the regulations.

3. If desired, grouping the selected FEPs into scenarios that are of particular interest (or required by the regulator).

4. Choosing scenarios, again based on well-defined criteria, for consequence analysis. Scenarios with similar consequences can grouped together.

Criteria for omitting scenarios include (a) exceedingly low probabilities of occurrence, (b) exceedingly low consequences, or (c) no role in accepted specifications of the calculation (for the WIPP Project, these criteria are based on guidance from Appendix C of 40 CFR 191).

6.2 Hazard Identification After Site Selection

6.2.1 Features, Events, and Processes Considered

After site selection in 1973-74, ORNL identified hazards not eliminated by the selection process for the WIPP (Claiborne and Gera, 1974). Natural events considered were volcanism, faulting, erosion, and meteorite impact (the latter not site specific). Climatic change and nuclear criticality were only mentioned. Anthropogenic events were drilling, sabotage, and nuclear warfare (the latter two events were not site specific). The primary process considered was groundwater transport of radioisotopes following faulting or meteorite impact. No undetected features were considered.

6.2.2 Screening Calculations

Claiborne and Gera (1974) also qualitatively evaluated the consequences and/or probabilities of the FEPs, although a few FEPs were analyzed deterministically. Readily dismissed were effects after closure from sabotage (which would require an occupying army to drill into the repository) and a crater from a surface blast of a nuclear weapon (which could not reach the repository). Exposure from drilling was assumed to be limited to the drilling crew. Failure of borehole plugs after a borehole had been abandoned was considered. However, potentiometric heads at the site were believed to favor flow to aquifers below the repository, and it was argued that it was unlikely that an individual would drill so deeply to obtain salty water (Culebra brine flowing through the Salado into the deep aquifers would become even more salty). Measurable consequences from a meteorite impact or faulting that moved the repository near the Culebra aquifer were estimated but the probabilities of the events were thought to be very low. The WIPP site was in one of the most tectonically stable parts of the United States and the probability of a catastrophic meteoritic impact was believed to be about 1.6×10^{-13} per year.

6.3 Hazard Identification and Scenario Development for EIS

In 1976, Sandia first began to develop descriptions of potential mechanisms through which radioactive waste could be released for the safety analysis to be included in the EIS. The purpose was to enumerate which processes might need further experimental study, which should be included in computer modeling, and which could be eliminated. This activity became the foundation for scenario development conducted for the first performance assessments in the 1990s.

6.3.1 Features, Events, and Processes Considered

For the EIS, Sandia considered events similar to ORNL's list, such as drilling, although in a more formal manner. Sabotage, nuclear warfare, erosion, and meteroite impact were quickly dismissed as single events, although retained as initiating events for other events that would affect release. A new event added was potash mining and a new feature added was an undetected brine pocket. Hence, by the time of the EIS, four potential sources of brine had been identified for scenarios: (1) brine pockets, (2) brine aquifers, (3) interstatiscal grain brine, and (4) brine inclusions. In addition to groundwater transport, new processes included climatic change, subsidence, continued dissolution of salt directly over the site, diapirism (i.e., formation of salt domes in the strata overlying the salt beds because of repository heat), and buoyancy of canisters (i.e., vertical movement of canisters because of repository heat).

Nuclear criticality, diapirism and buoyancy of canisters had been added because of public attention, but were dismissed after evaluation (Bingham and Barr, 1979; 1980; Dawson and Tillerson, 1978).

6.3.2 Construction of Scenarios

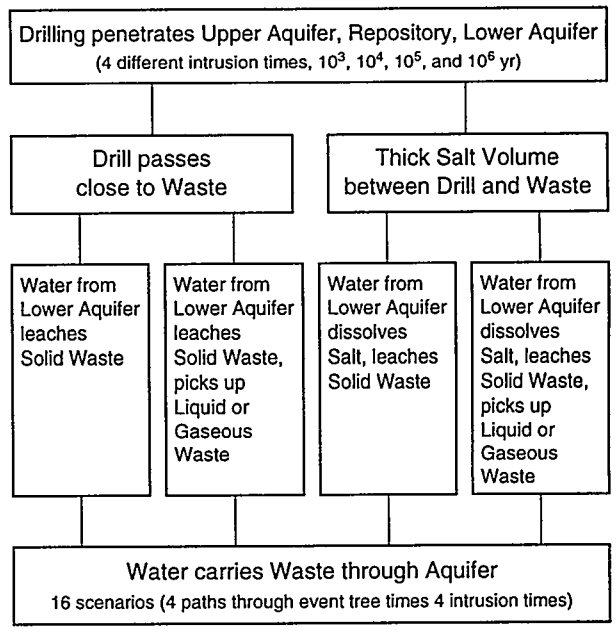
To link FEPs into futures, the WIPP Project first attempted to use fault trees but found the method untenable for three main reasons: the fault tree quickly became large and unmanageable; it was difficult to determine whether futures were mutually exclusive; and it was difficult to include time dependent events. Instead, Sandia developed a method based on event trees that began with a system that was not disrupted and subsequent events that provided mechanisms for moving radioisotopes through the disposal system (Figure 6-1a). The WIPP histories fell into three scenario classes (Bingham and Barr, 1979; 1980): (1) those that exposed waste directly to the biosphere, e.g., drilling through the repository, (2) those in which water flowed between two aquifers after borehole plugs had failed in exploratory drill holes, and (3) those in which water flowed from only one aquifer, requiring diffusion and convection to move radioisotopes through the salt. These three categories of scenarios and two other worst-case scenarios were analyzed early in 1978 for inclusion in the Draft EIS (DOE, 1979).

6.3.3 Hazards Related to Waste Acceptance Criteria

As described in Section 4, potential hazards from the waste were evaluated for the EIS while attempting to establish waste acceptance criteria. These hazards included combustibility, microbial gas generation and enhanced mobilization of actinides related to organic content, nuclear criticality, and excessive mobilization, corrosion, and gas generation related to volume of liquids in sludge wastes. The repository was found capable of mitigating all existing hazards, except for sludge waste with high liquid content. For sludge waste, insufficient tests had been run to estimate consequences and so free liquids in the waste were limited instead (Sandia, 1979).

6.4 General Scenario Development

While the DOE was examining hazards at the WIPP, the NRC funded another group within Sandia (separate from the WIPP Project) at the end of 1976 to



(a) Event tree approach used for scenarios in WIPP EIS in 1979 (after Bingham and Barr, 1980).

Events

TS = Subsidence resulting from solution mining of potash

E1 = One or more boreholes pass through waste and into a brine pocket

E2 = One or more boreholes pass through waste without penetrating a brine pocket

Combinations and nomenclature

$S_1 = \overline{TS}\,\overline{E1}\,\overline{E2}$ = "E0" (base case - undisturbed performance)

$S_2 = \overline{TS}\,\overline{E1}\,E2$ = "E2" (assumed bound by E1)

$S_3 = \overline{TS}\,E1\,\overline{E2}$ = "E1"

$S_4 = \overline{TS}\,E1\,E2$ = "E1E2"

$S_5 = TS\,\overline{E1}\,\overline{E2}$ = "TS"

$S_6 = TS\,\overline{E1}\,E2$ = "TSE2"

$S_7 = TS\,E1\,\overline{E2}$ = "TSE1"

$S_8 = TS\,E1\,E2$ = "TSE1E2"

$$\sum_{j=1}^{nS} P(S_j) = 1$$

(b) Combination (order unimportant) of events for scenarios in 1990 - 1992 performance calculations (after Rechard, 1995).

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Figure 6-1. Technique to construct scenarios from various features, events, and processes changed between WIPP Environmental Impact Statement in 1979 and performance assessment calculations after 1989.

pioneer work on a probabilistic risk assessment on geologic disposal of commercial high-level waste and spent nuclear fuel (Campbell et al., 1978). Their work benefited from the early work at the WIPP and the commercial disposal program conducted at ORNL until 1976 and then at Battelle when ORNL declined further involvement. Later the WIPP Project was to benefit from the NRC's efforts.

6.4.1 Defining Universe (FEPs List)

In 1976, the NRC funded two conferences that brought together a panel of earth scientists to generate a generic list of FEPs as a starting point in assessing the performance of a geologic repository (Cranwell et al., 1990). In a related international effort in 1981, the IAEA recommended a list of FEPs for initial consideration, along with a suggested procedure for performing an assessment (IAEA, 1981).

6.4.2 Screening Universe and Forming Scenarios

At first, the NRC requested that the Sandia group pursue a scenario development process similar to the *Reactor Safety Study* completed in 1975 (Rasmussen, 1975). However, like the Sandia WIPP Project, they found discretization of a highly coupled geologic disposal system by means of fault trees was not useful. In a draft report in 1981 (final report published in 1990) (Cranwell et al., 1990), Sandia proposed to the NRC a method to screen out unreasonable FEPs and form a limited number of scenarios. Based on its own and others' experience, the recommended procedure was a process in which the FEPs were essentially screened twice. The procedure included (1) generating an initial comprehensive list of relevant FEPs, (2) classifying the FEPs to aid in completeness arguments, (3) screening the FEPs based on well-defined criteria, (4) forming scenarios by combining specific remaining FEPs to form a scenario class, (5) screening the scenarios, and (6) selecting a final scenario set. Combinations of all events were used to construct scenarios; the time (or order) of occurrence was not considered (Figure 6-1). Hence, fewer scenarios were generated. Instead, broad classes of scenarios were developed that were convenient for modeling and construction of complementary cumulative distribution functions (CCDFs); time and order of occurrence of FEPs were to be included during modeling.

6.4.3 Including the Human Intrusion Event

Because injection boreholes and salt dissolution at the proposed repository at Lyons, Kansas, were discovered after site selection, human intrusion, both intentional and inadvertent, was included in the initial repository analysis for the WIPP. However, the inclusion of human intrusion eventually became controversial. For example, during the late 1970s, each site with previously buried TRU waste and low-level radioactive waste, such as the national laboratories in Idaho or Los Alamos, prepared an EIS that evaluated disposal alternatives, e.g., leaving the waste as is, improving the trench covers, or retrieving and shipping the waste to the WIPP. The analyses examined costs and risks. Usually human intrusion of some type provided a significant portion of the calculated consequences but attaching a probability to the event was difficult. Also, no one knew how the human intrusion event related to the robustness of the disposal system (Smith, 1982), except that more varied types of intrusion would likely occur for disposal sites at shallow depths and for sites near unique mineral resources.

By 1982, in the draft of 40 CFR 191, the EPA reduced the scope of human intrusion to an inadvertent activity, specifically from exploratory drilling primarily for oil and gas deposits (EPA, 1982). Thus, the risk from human intrusion became a measure of the type of media selected, the depth of the repository, and its association with economic minerals. A natural extension, although unstated until 1996 in 40 CFR 194 (EPA, 1996a), was to assume the exploratory drilling used technology currently operating in the region. This assumption was conservative, given acceptance of the proposition that the repository presented the greatest hazard to a society with technical capabilities comparable to our own rather than to a society with less or greater technical prowess. In 1995, for the future regulation of the disposal of commercial spent nuclear fuel at Yucca Mountain, the NAS recommended that the calculation of risks from human intrusion be distinctly separate from those from natural migration of disposed waste (NAS/NRC, 1995).

6.5 Hazard Identification for Preliminary PAs

The preliminary performance assessments at the WIPP (Marietta et al., 1989; Bertram-Howery et al., 1990; WIPP PA Division, 1991/1992; WIPP PA Dept.,

1992/1993; Rechard, 1991; Rechard et al., 1989; 1990) used an initial screening of FEPs performed in 1989 (Hunter, 1989). The scenarios retained for consequence analysis belonged to two principal categories: undisturbed (base-case) performance and disturbed (human-intrusion) performance.

6.5.1 Hazards Identified in 1980s

Several natural processes that might compromise the disposal system had been postulated during the early 1980s and mandated investigations by agreements with the State of New Mexico. The possibilities of dissolution of the Salado (to form "breccia pipes") and dissolution at great depths in the Culebra at the site (causing "karst" hydrologic flow) were examined by the USGS (Snyder et al., 1982) and Sandia (Lambert, 1983). They were resolved as either not likely to occur at all or not in a manner that would impair WIPP performance, respectively. However by 1987, the presence of a brine reservoir under the repository in the Castile could not be unequivocally dismissed and so became a potential undetected feature. In addition, the Supplemental EIS of 1989 identified gas generation as an important process to reexamine.

6.5.2 Features, Events, and Processes Retained

The FEPs for the WIPP were defined in 1989 (Hunter, 1989) in conjunction with a demonstration of the performance assessment methodology. The basic features retained included an undetected brine reservoir under a portion of the repository, seals in the shafts to limit downward movement of Culebra brine or upward movement of contaminated gas and brine from the repository (see Section 5, Design of Facility), an overlying, fractured brine aquifer in the Culebra, spatial variability of transmissivity fields (zones were analyzed in 1989-1990, and random fields analyzed thereafter), and fractured anhydrite beds slightly above or below the repository horizon (see Section 3, Characterization of Site) (Figure 6-2).

The primary basic event considered was human intrusion from exploratory drilling. However, disturbance of the stratigraphy from potash mining above the repository and nuclear criticality were also identified, although not included as events until the 1996 analysis for the CCA. No disruptive natural events with probabilities greater than 10^{-4} per 10^{4} yr for any of the WIPP PAs were identified (Hunter, 1989; Galson and Swift, 1995).

Natural processes retained for the preliminary PAs included climate variability (analyzed separately in 1989 and included in performance assessments after 1990), hydrologic transport in fractures of the Culebra (alternative models explored in 1992), generation of gas from container corrosion or microbial degradation of organic material such as cellulose in the waste, two-phase (brine and gas) Darcy flow in and around the repository in the Salado, pressure-dependent creep of salt around the waste in 1992, and fracturing of anhydrite layers in 1994.

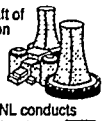
6.5.3 Undisturbed Scenario

The undisturbed scenario, E0, represented the performance of the disposal system from the time of disposal through the 10,000-yr regulatory period and incorporated all expected changes in the system and associated uncertainties (Figure 6-1b). In the demonstration PA of 1989, two potential pathways for migration of contaminants were considered. In the first path, the pressure gradient between the waste disposal panels and the Culebra was assumed to cause brine and radioisotopes to migrate either through drifts or anhydrite interbeds to the base of the shafts and then upward to the Culebra. Transport was then assumed to occur laterally in the Culebra toward the subsurface boundary of the accessible environment. In the second path, brine and radioisotopes were assumed to migrate laterally from the undisturbed repository through thin anhydrite interbeds toward the subsurface boundary of the accessible environment within the Salado (Figure 6-3). In all performance assessments conducted for the WIPP, no radioisotope releases occurred for the undisturbed scenario. In November 1996, the NAS echoed the findings of WIPP analyses that showed that the excellent isolating properties of bedded salt at the WIPP could be compromised only by human intrusion (NAS/NRC, 1996).

6.5.4 Human Intrusion Scenarios

In the performance assessments, the only disruptive event for scenario construction was inadvertent human intrusion. After 1990 the future inadvertent drilling events were assumed to occur randomly in time and space, that is, each drilling event was independent of every other drilling event, and mathematically described as a Poisson process (Helton, 1993c).

1974 Draft of 1st PRA on nuclear reactors



1974 ORNL conducts scenario development and consequence analysis of WIPP

- **1974 - Aug:** Draft of first major Probabilistic Risk Assessment (PRA) published on two reactors by 60 member team for Nuclear Regulatory Commission (NRC); method uses fault trees to synthesize probability of total system failure.[F1] **Oct:** ORNL conducts first scenario development and deterministic analysis for WIPP.[F2] Probability of meteorite impact, probability of fault (and volcanism) and exploratory drilling intersecting disposal area estimated.
- **1976 - Dec:** NRC funds panel of earth scientists to identify events and processes that could disrupt a generic repository.[F3]
- **1977 - Apr:** Second meeting of NRC panel of earth scientists occurs to identify events and process.[F3] INEEL begins risk analysis of alternatives for TRU waste stored and buried at Radioactive Waste Management Complex (RWMC) over next 4 years.[F4] Los Alamos, Savannah River, and Harford begin similar studies as well. Human intrusion event significant contributor to consequences in these studies.

1978 SNL adapts PRA for reactors to nuclear waste repositories

- **1978 -** Hydrologic and radioisotope transport modeling for draft EIS is primarily regional and extends for 250,000 yr (10 half-lives of ^{239}Pu) using large, 3-D Swift flow model.[F5] NRC funds SNL to work on probabilistic PA and apply to hypothetical bedded salt repository.[F6, F7]

1979 Draft EIS on WIPP conducts transport analysis of Pu out to 250,000 yr

- **1979 - Apr:** Draft EIS on WIPP published.[F5] SNL completes development of scenarios for release radioisotopes from WIPP (part of EIS; process resulting method abandons fault trees and uses simple event trees).[F8, F9] 3 major classes of scenarios identified (connection between Culebra aquifers above and below repository, Bell Canyon aquifers, U-tube connection to Culebra, and stagnant pool connection to Culebra) plus drilling intrusion. (Later U-tube split into catastrophic connection and standard U-tube connection.)[F5] Probabilities of scenarios assigned based on qualitative reasoned arguments.

1981 SNL reports on PA of hypothetical salt site

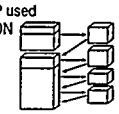
- **1981 -** Draft of final report to NRC on performance assessment (PA) of hypothetical bedded salt repository readily available[F7, F10] – uses a set of loosely connected codes, precursors to SWIFT_II[F10] (fluid flow code), and NEFTRAN[F12] (network transport code). IAEA recommends procedure for PA and potential list of events and processes for scenarios.[F13] **Jul:** Stipulated Agreement (SA) between New Mexico and DOE describes disruptive scenarios (e.g., breccia pipe, salt dissolution, and salt deformation) that are to be dismissed through further site characterization.[F14]
- **1986 - Aug:** DOE asks SNL to assess performance of WIPP against 40 CFR 191 criteria;[F15] SNL accepts PA task.[F16]
- **1987 - Mar:** SNL finds that porous-media flow assumption adequately models flow in Culebra at H-3 but that transport is best modeled as a dual porosity media[F17] (though roughly approximated as equivalent porous media).[F18, F19, F20] Modeling with variable brine densities suggest Culebra acting as leaky confined aquifer;[F17] subsequent models ignored suggestion until 1997. Also model suggests highly transmissive zone in the Culebra to the south of H-11 and DOE-1.[F17] SNL finds possibility of a pressurized brine reservoir below the TRU disposal area cannot be ruled out.[F18, F21]
- **1988 -** SNL begins work of CAMCON to link detailed consequence models in probabilistic PA.[F22] SNL also simultaneously begins work on prototype of CAMCON to meet Dec. 1989 deadline. SNL completes pumping tests at H-11[F23] and begins using results to calibrate regional flow model.[F24, F25]

1989 Draft supplemental EIS identifies gas generation as issue

1989 Demo for WIPP PA

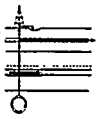
- **1989 - Mar:** SNL completes report to support Draft Supplemental EIS; report identifies generation of gases from container and waste corrosion as issue because salt permeability lower than thought in 1979. Also, different flow direction in past during wet climate hypothesized to explain discrepancy between geochemical analysis and current hydrologic flow in Culebra.[F26] DOE issues Draft Supplemental EIS.[F27] **Dec:** SNL reevaluates release scenarios[F28] and issues WIPP PA demonstration outlining process for future PAs.[F29, F30] No release without human intrusion; out of 26 parameters, solubility, intrusion time, and borehole permeability most important; cuttings from direct drilling set at 3 drums.

1990 First full PA of WIPP used CAMCON



- **1990 - Jan:** DOE issues Final Supplemental EIS.[F31] **Jun:** DOE issues "Record of Decision" on WIPP Final Supplemental EIS stating construction is officially complete, testing phase (~5 yr) should proceed, and then another Supplemental EIS should be prepared before going to full operation.[F32] SNL refines FEP screening and analyzes four scenarios (E0, E1, E2, E1E2).[F33] **Dec:** SNL issues 1st full PA highlighting use of CAMCON modeling system.[F34, F35, F36] (e.g. secondary parameter database completed); coupling of code demonstrated, which allowed better evaluation such as sensitivity analysis. PA includes both scenario and parameter uncertainty: out of 3 parameters, solubility, intrusion time, and borehole permeability important; cuttings from direct drilling important release pathway.

1991 Major models linked in WIPP PA



- **1991 - Dec:** SNL issues 2nd PA highlighting major components of the PA process and documents.[F37] (e.g., rigorous use of scenarios and geostatistics for transmissivity fields); 46 parameters sampled; cuttings most important release pathway.

1992 Refinements to models (e.g., transmissivity fields) in WIPP PA

- **1992 - Dec:** SNL issues 3rd PA refining models and data used in the PA, uncertainty in transmissivity fields refined, 49 parameters sampled, evaluated time dependent λ parameter in Poisson intrusion model; direct cuttings most important pathway.[F38]
- **1994 - Mar:** SNL explores possibility of linking PA with decision analysis in System Prioritization Methodology (SPM); results form basis of Draft Compliance Certification Application (DCCA).
- **1995 - Mar:** DOE submits DCCA to EPA for review.[F39] Computer specialists hired to modify CAMCON implementation to enforce software configuration management and control runs for PA calculations. NAS provides guidance on new regulation for potential Yucca Mt. repository; suggests reporting risk from human intrusion separately. Second attempt at SPM.[F40]

1996 SNL completes PA for WIPP certification; moving van required to send copies to EPA

- **1996 - Oct:** SNL completes PA for Compliance Certification Application (CCA) of WIPP that includes MgO backfill, potash mining and subsidence scenario, and greater intrusion rate; except for few vectors, drill cuttings only release pathway; 57 parameters sampled.[F41] Calculation was run 3 times with 100 samples each and took 37,000 CPU hrs on 40 DEC alpha processors and retained 100 GBytes of data in 97,000 files. DOE sends 80,000 page, 400 lb. CCA to EPA.[F42] **Nov:** NAS reports that WIPP site "excellent choice" geologically.[F33, F34] DOE issues 84,000-page 2nd Supplemental Draft EIS.[F45, F46] **Dec:** EPA begins detailed evaluation of CCA and supporting information at SNL and elsewhere including PA conceptual models, computer codes, model parameters, QA records, and specific technical issues (e.g. MgO backfill and passive institutional control).[F47]
- **1997 - Jan:** Conceptual Model Peer Review Group (formed in response to 40 CFR 194) concludes 22 of 24 conceptual models adequate. Spallings model must be redone and MgO backfill description improved. **Mar:** SNL conducts mini PA for EPA to do parametric sensitivity analysis of PA model parameters lacking "ironclad" defense. **Apr:** Conceptual Model Peer Review Group reports that with additional information provided by SNL, they are satisfied that the new model of spallings[F48] and the model of the MgO backfill are adequate.[F49] **May:** SNL explains apparent discrepancy between geohydrology and geochemistry by viewing flow in Culebra as a 3D regional system.[F50] As part of EPA evaluation of CCA, SNL runs EPA-mandated PA calculations using EPA-selected values for 26 parameters and EPA-selected model assumptions, based on results from parameter review team comments in December and sensitivity analysis in March. In letter to DOE secretary, EPA Administrator Browner decrees DOE application "complete"; starting the 1yr clock for review of CCA.

1998 EPA certifies WIPP

- **1998 - May:** EPA certifies WIPP.[F51] **Jul:** NM AG sues EPA alleging insufficient time to comment on CCA. CARD and SWRIC also file law suits.
- **1998 - Mar:** Judge Penn lifts injunction: four days later WIPP receives first shipment of non-RCRA waste from LANL.[F52, F53]

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Figure 6-2. Engineering analysis of the illustrated behavior of the disposal system.

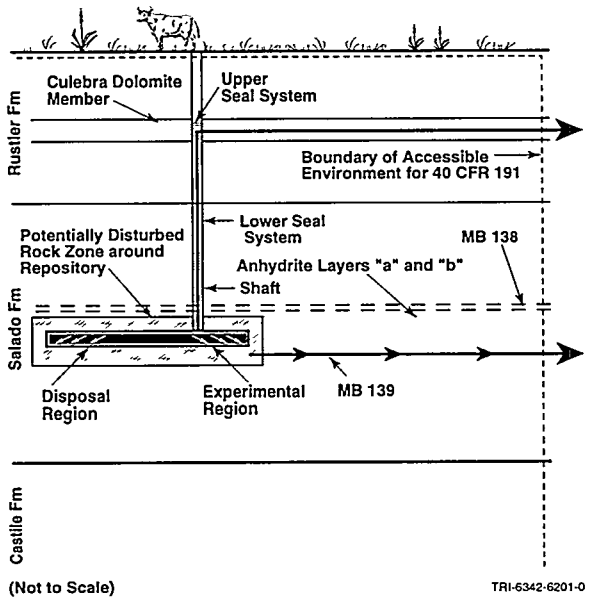


Figure 6-3. Undisturbed scenario used since 1989 (Rechard, 1995, Figure 3.2-3).

In the human intrusion scenario, it was assumed that if the disposal area of the repository was penetrated by an exploratory borehole, radioisotopes could be released in two different ways over two different time scales. First, an immediate release could occur during the drilling process, because the drill bit was assumed to bore vertically through a stack of CH-TRU waste containers or through a single RH-TRU waste container (refer to Section 4.2 and the description of TRU categories). Material within the containers could be ground up by the drill bit (called cuttings) and transported to the surface by the circulating drilling fluid. Additional material might be eroded from the walls of the borehole by the swirling action of the drilling fluid (called cavings) or the spalling of solid material into the hole as the panel depressurizes. Second, although it was assumed the boreholes would be plugged upon abandonment according to current industry standards, selective degradation of these plugs, accompanied by an eventual shift to a permeability similar to that of sand (as suggested in Appendix C of 40 CFR 191), could lead to the possibility of long-term releases by means of transport through the repository, up the boreholes to the aquifer in the Culebra, and then laterally through the Culebra toward the boundary of the accessible environment. In subsequent discussion of the models, transport through the repository and boreholes was evaluated by repository modeling, and flow and transport through the Culebra by Culebra flow and transport modeling (see Section 7).

In a refinement of FEP screening in 1990 (Guzowski, 1990), the presence of the brine reservoir in the underlying Castile was combined with exploratory drilling to produce three representative intrusion scenarios (Figure 6-1b): E1, a borehole drilled through the repository and the brine reservoir; E2, a borehole drilled through the repository only; and E1E2, a combination of the two. In the scenarios, the borehole plugs were assumed to degrade so that contact was maximized between the pressurized Castile brine and the panel of waste. For example, for the E1E2 scenario, the borehole that penetrates the Castile brine reservoir (E1 borehole) was assumed to remain plugged just above the level of the waste panel. The E2 borehole was assumed to remain plugged just above the level of the Culebra aquifer. Thus, the pressure-driven brine flows through the panel before flowing up the E2 borehole to the Culebra aquifer. These plug configurations were chosen to facilitate examination of the specific scenarios and did not reflect the most realistic conditions expected. Any brine entering through the boreholes was assumed to access all waste within one panel (Figure 6-4). For improved computational resolution in 1991 and 1992, the three scenarios were divided further into computational scenarios on the basis of time of intrusion and radioactivity of the intersected wastes starting in 1991 (Helton and Iuzzolino, 1993). In addition, E2-type intrusions were

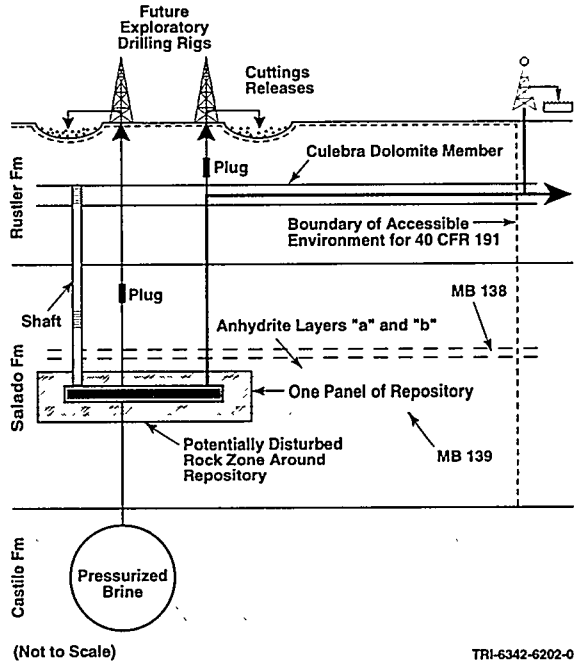


Figure 6-4. E1E2 human intrusion scenario used since 1990. Intrusion borehole plugs are placed to maximize brine flow through the repository (Rechard, 1995, Figure 3.2-4).

not analyzed explicitly but rather assumed to have the same consequences as E1-type intrusions (WIPP PA Division, 1991/1992).

6.6 Hazard Identification for 1996 PA

For the 1996 PA in support of the CCA, a formal hazard identification and screening process was conducted, as described in Galson and Swift (1995). 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 et al., 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 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 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. However, during its 1997 review of the CCA, the EPA (and the EEG) closely examined the justifications for eliminating various FEPs. In particular, the removal of the water flood event was scrutinized and the EPA requested additional analysis.

7.0 Consequence Analysis

Although various components of the disposal system had been analyzed independently since project startup, the first major consequence analysis of the disposal system as a whole was conducted in 1979 for the EIS (DOE, 1979; 1980a). Thereafter, the whole system was analyzed for the Supplemental EIS in 1989 (Lappin et al., eds., 1989; DOE, 1989c, 1990c), during the four preliminary performance assessments between 1989 and 1992 (Marietta et al., 1989; WIPP PA Division, 1991/1992; WIPP PA Department, 1992/1993), and for the 1996 PA in support of the CCA (DOE, 1996a). Currently, component and system analysis continues in preparation for recertification, which is required every five years during operation of the WIPP.

7.1 Description of Consequence Analysis for the WIPP

In general, a consequence analysis consists of (1) a toxicity assessment, which evaluates the response of a "receptor" (e.g., a human) to a hazard (or "stressor"), and (2) an exposure pathway assessment, which evaluates the exposure intensity of a hazard that reaches a receptor[1]. For the original EIS, the dose conversion factors developed by the DOE were used (DOE, 1979; 1980a). Later, the EPA performed the toxicity assessment when it established its release criteria in 40 CFR 191, as promulgated in 1985 and 1993 (EPA, 1985a; 1993b). Evaluating the exposure pathway for the WIPP involved development of a model that is actually a mathematical function, designated herein as $q(\bullet)$ to predict exposures.

For a geologic disposal system, a challenge in consequence analysis has been understanding long-term behavior of system components, e.g., waste containers and their interaction with the host rock environment. The various physical scales in a geologic disposal system made one detailed exposure pathway model impractical. Instead $q(\bullet)$ was further divided into many component models $M_\ell(\bullet)$, that transferred variables produced by one model, $M_{\ell-1}(\bullet)$, to variables used by the next model, $M_{\ell+1}(\bullet)$. The models $M_\ell(\bullet)$ were dependent on the scenario S_j (stochastic model parameters) under consideration and a subset of epistemic model parameters $\mathbf{x}^p$ related to that scenario. As discussed in the following sections, the component models used for the WIPP were the direct release model, repository model, source term model, Culebra flow model, and Culebra transport model.

7.2 Analysis for Site Selection and EIS

7.2.1 Site Selection Analysis

Although the consequence analysis performed during site selection by ORNL (Claiborne and Gera, 1974) is closer to a screening analysis, it influenced the assumptions and modeling techniques for the original EIS on the WIPP and so is described here. The analysis included assumptions about two component models $M_\ell(\bullet)$: the source term model and the flow and transport model in the Culebra.

Source Term Modeling. In 1974, the radioisotope source was assumed to be high-level waste in 75,000 borosilicate glass canisters that were placed in the repository floor. Water from over- or underlying aquifers was assumed to have access to only a portion of the inventory—either 98 canisters in a short row or 765 canisters in a long row. The leach rate was set at 10^{-7} g/cm^2 per day, based on measured leach rates in borosilicate glass, and resulted in release rates of between 130 Ci/yr after 10^3 yr and 5 Ci/yr after 10^6 yr.

Culebra Flow and Transport Modeling. In general, flow through the repository was assumed to be from the Culebra aquifer to deeper aquifers. However, transport of radioisotopes after catastrophic faulting placed the repository at the same level as the Culebra aquifer was analyzed. The radioisotope inventory was simply diluted by the volume of flow through the Culebra and by the minimum annual flow of the Pecos River at Malaga Bend. The dilution from the Culebra, as estimated from data from the Gnome site (Gard, 1968), was 1.1×10^5 m^3/yr based on a 2.4-km-wide repository, 9-m-thick Culebra, and a Culebra porosity of 0.10.

1 In essence, the consequence model is divided into two model categories: (1) a model for the human receptor, which currently is a simple linear bounding response curve, and (2) a model of the environment around the receptor, which consists of numerous complex exposure pathway models.

7.2.2 EIS Analysis

In support of the EIS on the WIPP (DOE, 1979; 1980), the consequences of release were evaluated in terms of human exposure to radioisotopes that had reached the biosphere by some mechanism (Powers et al., eds., 1978). As described in Section 6, three pathways to humans were of concern: direct release and two groundwater pathways. The consequences of release by these three pathways were treated deterministically and individually. The period considered for evaluating the environmental impact of the WIPP was a quarter of a million years, roughly ten half-lives of ^{239}Pu. The consequence analysis used four component models $M_\ell(\bullet)$: the direct release model, the source term model, the Culebra flow model, and the Culebra transport model.

Direct Release Modeling. In the EIS, consequences measured as dose were evaluated from direct release to either a geologist on the drill rig examining the core or a member of a single-family farm living 500 m downwind of the abandoned mud pits. Direct release to the surface from an intrusion into the repository was estimated simply as the amount of radioisotopes that would be removed by a 25-cm (10-in) rotary drill for oil and gas exploration and deposited in the mud pit or a 7.6-cm (3-in) core drill for mineral exploration and deposited in a small pit. Exposure of a geologist examining the 7.6-cm core was calculated to be 1 mrem. Exposure to a member of a single-family farm, who breathed the contaminated air and ate the farm's leafy vegetables, dairy products, and beef cattle, was calculated to be a maximum of 0.036 mrem, primarily from inhalation.

Source Term Modeling. Release of radioisotopes from the WIPP repository via the groundwater pathway was assumed to be congruent with dissolution of salt encapsulating the waste, regardless of the solubility of the radioisotope (DOE, 1980a, Appendix K). Disruption of the repository was assumed to occur after 1000 yr for all scenarios, except the bounding case scenario, with catastrophic flow of the Rustler aquifer through the repository, which was assumed to occur at 50,000 yr.

Culebra Flow and Transport Modeling. In the EIS, flow through the repository was assumed to be from the deep aquifers into the overlying Rustler. The Culebra and Magenta were combined and modeled by INTERA, Inc., for Sandia as a 12-m-thick layer, referred to as the Rustler aquifers, using a finite difference code, SWIFT (Sandia Waste Isolation Flow and Transport code [Dillon et al., 1978]), whose development was being funded by the NRC (see following section). The three-dimensional regional model for flow encompassed the Pecos River past Malaga Bend to the south, the town of Carlsbad to the west, Clayton Basin to the north, and the west edge of San Simon Swale to the east (Figure 7-1). Each layer was assumed to be an isotropic porous medium with several hydraulic conductivity zones. The Rustler aquifer layer was assumed to have a uniform porosity of 0.10, a hydraulic conductivity of 0.3 m/s above the repository and to the north, a hydraulic conductivity of 1.2 m/s in Nash Draw and to the south of the WIPP site, and higher conductivity along the Pecos River (DOE, 1980a, Table K-2, Figure K-7). At this time, no regulatory exclusion zone or boundary had been defined, so regional flow, assumed to be toward the southwest, was discharged at Malaga Bend on the Pecos River. Transport calculations used a one-dimensional model along several flow paths to the Pecos River and included retardation.

7.2.3 NRC Analysis

For the NRC, Sandia initially described a consequence analysis method in 1978 (Campbell et al., 1978). In 1981 that process was applied to a hypothetical bedded salt repository (Cranwell et al., 1987) along with the scenario development procedure discussed in the previous section. The analysis process was similar to that used in the WIPP EIS and proposed at Pacific Northwest Laboratory in 1977 (Bartlett et al., 1977). The exposure pathway model, $q(\bullet)$, comprised a series of loosely connected individual codes specifically developed for the task. The study simulated a steady-state groundwater flow field using the finite-difference flow code, SWIFT (Dillon et al., 1978), evaluated a particle pathway, and then calculated radioisotope transport along this pathway using a network model, NWFT/DVM (Campbell et al., 1981). The groundwater releases to the surface were then input to a lumped parameter (compartment) model to evaluate radioisotope concentrations in surface water, sediments, and soil. These concentrations were then propagated through various food chains that eventually led to humans (Iman et al., 1978; Campbell et al., 1978; Cranwell et al., 1987).

In the mid-1980s, Sandia applied SWIFT II (Reeves et al., 1986) and NEFTRAN (Longsine et al., 1987) to a hypothetical repository in basalt similar to the geology found near Hanford, Washington (Bonano et al., 1988) for the NRC. Although the application of the numerical solution for the partial differential equations describing radioisotope transport had been implemented in SWIFT II, it remained difficult in practice so NEFTRAN, the next generation of NWFT/DVM, was used.

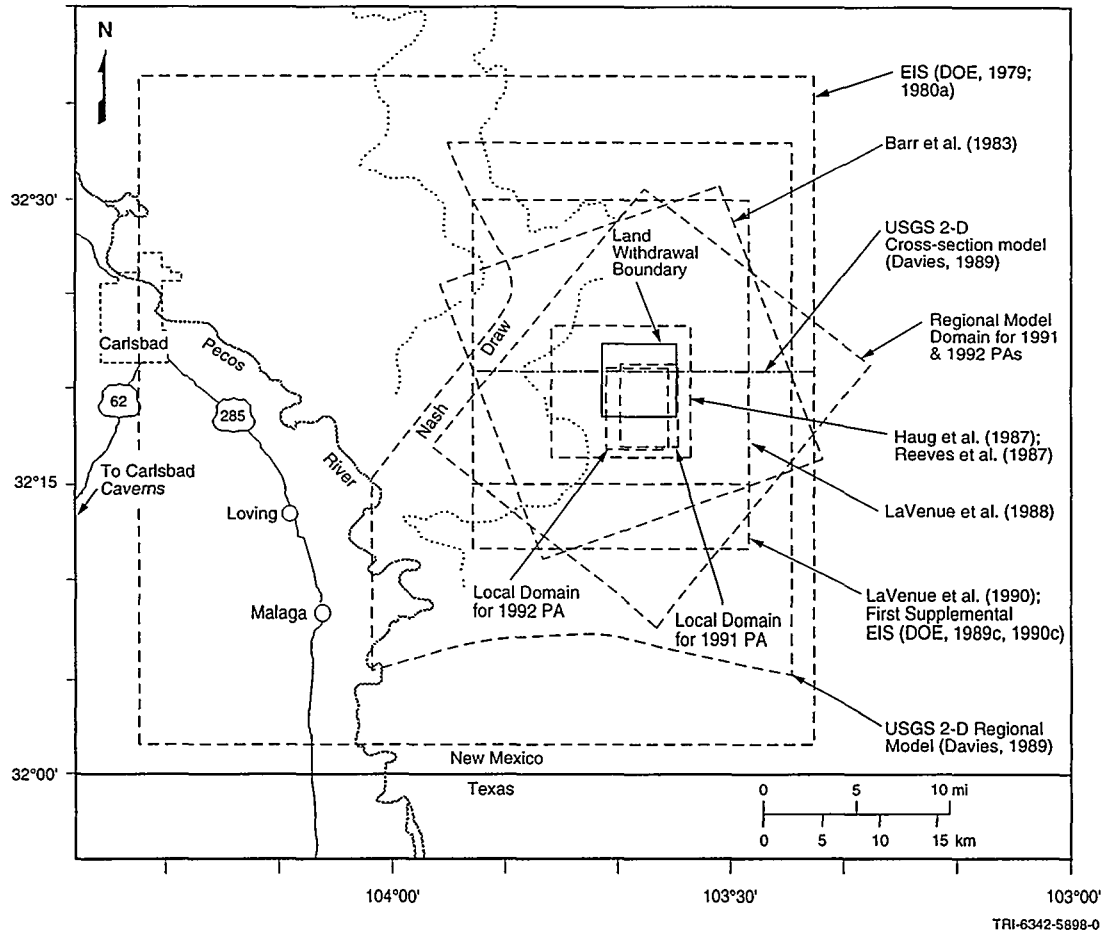


Figure 7-1. Boundaries of models constructed for analysis of groundwater flow and transport of radioisotopes in the Culebra at the WIPP (after Lappin et al., eds., 1989, Figure 3-25).

7.2.4 Supplemental EIS Analysis

The Draft and Final Supplemental EISs for the WIPP, completed in 1989 and 1990 (Lappin et al., eds., 1989; DOE, 1989c; 1990c), modeled release in a manner similar to that used for the original EIS, but incorporated the pressurized brine reservoir feature. As before, four major components $M_\ell(\bullet)$ of the exposure pathway model, $q(\bullet)$, were constructed: the direct release model, the Culebra flow model, the Culebra transport model, and the source term model. However, the latter model included some aspects of the repository and brine reservoir.

Direct Release Modeling. For direct release in the Supplemental EIS, an estimate was made of the volume of waste removed through erosion by the circulating drilling mud and then added to the volume removed by the drill bit, resulting in an upper bound of three full drums of CH-TRU waste. For dose, the same pathways were used as in the EIS. The geologist dose was about the same as that calculated in the 1980 EIS, ~0.08 mrem, but the farm family dose, a maximum committed dose equivalent over 50 yr from inhalation, was more than the 1980 EIS calculation, ~0.77 mrem.

Source Term/Repository Modeling. A distinct source term/repository model was developed for the Supplemental EIS, using SWIFT II (Reeves et al, 1986), to evaluate the concentration of radioisotopes injected into the Culebra. Four deterministic cases for the human-intrusion scenario were run, Cases IIa-d, using best and degraded values for parameters. The inventory was limited to either one panel (Cases IIa, IIb, IIc) or one room (Case IId) based on assumptions of the compaction and permeability of the salt backfill. Unlike the original EIS, the source term concentration was limited by a general actinide solubility in addition to the inventory. A range was established for the solubility limit for all actinide radioisotopes but in the calculations was set at either 10^{-3} mM or 10^{-1} mM (Cases IIb and IIc) (see Figure 4-2).

The use of a solubility limit to determine concentration required that the amount of brine available to dissolve the radioisotopes be estimated. A constant flow from the Salado of either 1.3 m^3/yr (Case IIb and IIc) or 0.1 m^3/yr (Case IId) was used, based on calculations completed in 1988 in conjunction with the brine inflow controversy (Nowak et al., 1988). The amount of brine flow from the Castile brine reservoir was estimated using a well bore submodel in SWIFT II and a numerical mesh of the brine reservoir. The degraded borehole permeability was assumed to be either 10^{-12} or 10^{-11} m^2. Other parameters of the Castile brine reservoir such as initial pressures, thickness, and diameter were varied. Given the assumption that the borehole plugs failed at 75 yr, the amount of brine from the Castile reservoir was initially either 9.9 m^3/yr (Case IIa) or 98 m^3/yr (Cases IIb, IIc, and IId).

Characterization Study of Culebra in 1987. In 1987, Haug et al. (1987) of INTERA, Inc., calibrated a two-dimensional flow model to the H-3 pumping test (Beauheim, 1987) and the effects from the excavation of the shafts for Sandia. Data from several boreholes that had been drilled and tested by Sandia since 1980 were included in this model. The boundaries of the model were slightly larger than the WIPP site (Figure 7-1). As a secondary calibration target, measured brine densities were used; they were assigned at the boundaries and subsequently modified to match the observed fluid densities. Vertical leakage was included in an effort to calibrate the brine densities, which led to the recommendation that future modeling studies treat the Culebra as a leaky-confined aquifer. The transmissivity field was estimated by kriging and modified by the addition of artificial transmissivity well measurements ("pilot points"), which were positioned manually by trial and error. Haug et al. (1987) found that to match the low levels at wells H-11 and DOE-1 required placing a highly transmissive zone south of these wells. Based on a comparison of a model at the regional scale, Haug et al. (1987) also concluded a single-porosity (matrix-only) conceptual model adequately simulated the *fluid flow* field.

Modeling of Culebra Fluid Flow in 1989. The 1987 study was followed by two more modeling studies by INTERA, Inc., for Sandia in 1988 and 1989 (LaVenue et al., 1988, 1990) in support of the Supplemental EIS. The differences in the models were that vertical leakage was not included and brine density varied spatially but was held constant over time. Also, the boundaries of the 1988 and 1989 studies were larger than those of the 1987 study. The 1989 study extended approximately 30 km north and south and 20 km east and west, with the WIPP site at the center (Figure 7-1). These boundaries were selected to include the region for which head data were available and to minimize the boundary effects during simulation of the H-3, WIPP-13, and H-11 pumping tests. Fixed heads, based upon the regional head values, were assigned around all four boundaries. Transmissivities were estimated by kriging from measurements at 41 well locations (Figure 3-2, all wells except H-19); the transmissivities varied over seven orders of magnitude over the model domain and three orders of magnitude within the WIPP site. As before, pilot points were added to modify the transmissivity field during steady-state and transient calibration; however, pilot point locations were selected using an adjoint sensitivity analysis technique rather than manual trial and error. Fluid flow was calculated on the basis of a fully confined Culebra with an effective thickness equal to the total average thickness of 7 m. Flow in the Culebra was predominantly north to south at the WIPP site, and strongly affected by the high-transmissivity zone in the southeastern portion of the site first proposed by Haug et al. (1987) in 1987.

Culebra Transport Modeling. For contaminant transport, a one-dimensional model was used with a dual-porosity formulation (i.e., fracture transport with matrix diffusion) along a selected flow path (Lappin et al., eds., 1989). However, the effect of lateral dispersion was estimated. Some transport parameters such as fracture block length (0.25 to 7 m) and fracture porosity (0.0015) were based on best estimates from nonsorbing tracer tests at H-3 and H-11 wells. Other transport parameters such as matrix porosity, matrix tortuosity, and grain density were evaluated from 73 core samples taken from 15 different wells. Longitudinal dispersivity was set at a maximum of 100 m.

7.3 Analysis Logistics for PA

Although the task had initially been assigned to Westinghouse, the DOE asked Sandia to assess the performance of the WIPP in 1986 in order to compare it with the criteria in 40 CFR 191 (Krenz, 1986; Beckner, 1986). The practical aspects of performing the exposure pathway calculations in a performance assessment were daunting for a system comprised of several complex model components, such as the WIPP disposal system. An important practical problem in the WIPP Project was how to link the component models together so that they were sufficiently comprehensible, traceable, and repeatable for regulatory review. Another important consideration was determining the appropriate level of detail for the individual models that comprised the exposure model. The manner in which these issues were addressed for the WIPP is discussed below.

7.3.1 Iteration of Calculations

In 1989, the WIPP PA analysts adopted the idea of conducting sequential performance assessments, that is, conducting an initial performance assessment with simple or incomplete models and preliminary data, followed by other performance assessments with better data and/or more detailed models (Rechard, 1989). The idea had been used before, e.g., repeated NAS studies of ozone depletion in 1975 through 1982 (NAS/NRC, 1982; Morgan et al., 1990) or the 1975 *Reactor Safety Study* and its 1990 update (Breeding et al., 1992). The value of repeating the PA process was that engineers and scientists could gain an understanding about the disposal system and how best to model it and also replace weak links in the simulation chain as improved models and data became available, as discussed in Section 7.4.

In addition, multiple performance assessment iterations achieved other benefits. First, a long, multiyear project could be divided into annual tasks, with more easily agreed-upon goals and schedules. Second, iterations allowed annual peer reviews so that the project received feedback that not only provided insights on the models and engineering analysis but also facilitated communication about controversial waste disposal issues and fostered interactions among members of the multidisciplinary teams. For instance, the PA group at Sandia formed a special external review group in 1987 that met through 1992 to review the preliminary performance assessments. In addition, the WIPP Panel of the NAS and the EEG, though not set up in 1978 exclusively to review performance-assessment-like calculations (or evaluate compliance), received quarterly presentations and made comments on performance assessment calculations.

Third, later iterations based on more advanced models or newly collected data could sometimes answer critical questions posed in earlier iterations. For example, the choice of the most appropriate conceptual model (i.e., whether to use single porosity or dual porosity to model radioisotope 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 (Section 8), iterative performance assessments 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 performance assessments from 1989 through 1992, with each building upon the others (Marietta et al., 1989; Rechard et al., 1990; Bertram-Howery et al., 1990; WIPP PA Division, 1991/1992; WIPP PA Department, 1992/1993).

7.3.2 Detailed Modeling Style

The analysis that Sandia conducted for the 1979 EIS relied heavily on detailed, phenomenological mathematical modeling to evaluate potential exposures (DOE, 1979; DOE, 1980a), particularly because public expectations, expressed as comments on early nuclear reactor EISs or the promulgation of regulations, suggested a preference for "realistic" analysis. By the 1990 PA, PA analysts had also chosen to emphasize the detailed modeling style. Comments received from the EPA (Bertram-Howery et al., 1990) and the WIPP Panel of the NAS on the 1989 demonstration encouraged Sandia to move from the simplified NEFTRAN models to more detailed modeling. Another reason for using a detailed modeling style was the general acceptance in the United States of its use in probabilistic risk assessments (Rasmussen, 1975; Breeding et al., 1992). The detailed style included phenomenological details and often multiple dimensions in the model and avoided simplified or conservative models or parameter values unless required data or knowledge was unavailable (Rechard, 1995). Also, when exploring the feasibility and desirability of subseabed disposal of radioactive waste, Sandia used detailed modeling of some system components such as ocean circulation (Marietta and Simmons, 1988). Some models, such as PANEL, remained simplified, but in general phenomenological models were used extensively in 1992 and 1996. However, the phenomenological models often used fairly coarse numerical descriptions, and in a few instances, the results of some models (e.g., SANTOS and FMT mentioned in Section 7.4) were abstracted into simplified descriptions rather than used directly.

7.3.3 CAMCON Development for PA

The major role of modeling in a performance assessment makes computer software fundamental to the process. Modeling a detailed complex system meant that models must be linked together reliably throughout a large number of repetitive computer simulations, as in a Monte Carlo analysis. Also, results must be properly identified for traceability. In response to these needs, the WIPP Project built the computer system, CAMCON, to aid in linking software and identifying results (Rechard, 1989, 1991; Rechard et al., 1989).

Although Sandia had developed codes that were loosely connected for the NRC in the late 1970s and early 1980s, the Canadians developed the first integrated system, SYVAC (Dormuth and Sherman, 1981; Lyon, 1982) in 1981. By the time Sandia was assigned the task of assessing the WIPP in August 1986, several other software systems had been built to meet the general requirements of performance assessment. One approach was to build one code with numerous submodels (e.g., SYVAC, VANDAL [Thompson, 1987], and LISA [NRC, 1983]), and another was to place one analysis code into a package that included data preparation, Monte Carlo sampling, and results display (e.g., NEFTRAN-S [Campbell et al., 1991]). Flexibility and quality assurance features, however, were especially important for radioactive waste disposal because the calculations were to be under intense scrutiny by the regulator. For the WIPP, serious work on developing a fully operational procedure that incorporated these characteristics began about mid-1987, with the CAMCON system developed primarily between 1988 and 1990. During the first year, a prototype was rapidly developed for the 1989 PA demonstration (Rechard et al., 1989). Simultaneously, a more carefully constructed version was developed for the first complete PA in 1990. The CAMCON system adopted several of the concepts that had been put forward for the NRC program, but discarded specific tools.

The original concept for CAMCON was to provide an analysis "toolbox" (more than one tool) whereby any number of either complicated numerical or simple analytical codes could be linked together (Rechard, 1989; Rechard et al., 1989). With this toolbox, any one of several interchangeable but not identical codes could be used for a model component, i.e., $M_\ell^1, M_\ell^2, \ldots M_\ell^{nM}$ where nM designates codes that perform a similar function, and ℓ designates a specific model component such as the Culebra transport model mentioned earlier. Section 7.4 discusses the different codes selected for the model components. The selected model components could then be linked with other model components to form the exposure pathway model, $q(\bullet)$. The toolbox also included tools such as MATSET (Rechard, ed., 1992), ALGEBRA (Gilkey, 1988), and RELATE (Rechard, ed., 1992) to extract data from a parameter data base, to algebraically manipulate output to evaluate new parameters or results (e.g., evaluation of a line integral to calculate release across a boundary), and to interpolate results across different meshes in order to make linkages between codes practical. The toolbox also included tools to help implement software quality assurance procedures.

The early version of the CAMCON system consisted of six components (Rechard et al., 1989; Rechard, 1991; Rechard, ed., 1992) (Figure 7-2): (1) code modules (or "grouping" of 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 (Rechard et al., 1993a); (5) a series of help files containing instructions on use and history of updates; and (6) two data bases—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 (discussed in Section 8). CAMDAT, which was based on a neutral file format that had evolved between 1980 and 1988 in Sandia's Engineering Analysis Department (Taylor et al., 1986; Mills-Curran et al., 1988), was the link between the computer modules.

The concept for the calculational system for the 1996 PA in support of the CCA was essentially the same, although details were changed. By the time of the 1996 PA, the costs of the stringent QA 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, and thus code modules were not necessary. Second, software specifically designed for configuration management was used rather than an ad hoc directory structure. Finally, 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. Efficiency of computer use increased as the result of these changes, although the driving force behind them was to provide the EPA auditors with objective evidence that the PA process was truly traceable, understandable, and repeatable by others.

7.4 Consequence Models in Performance Assessments

The major categories of the WIPP consequence model components for the performance assessments were the direct release models, the repository fluid flow model, the source term model, the Culebra transmissivity model, and the Culebra flow and transport model (Figure 7-3).

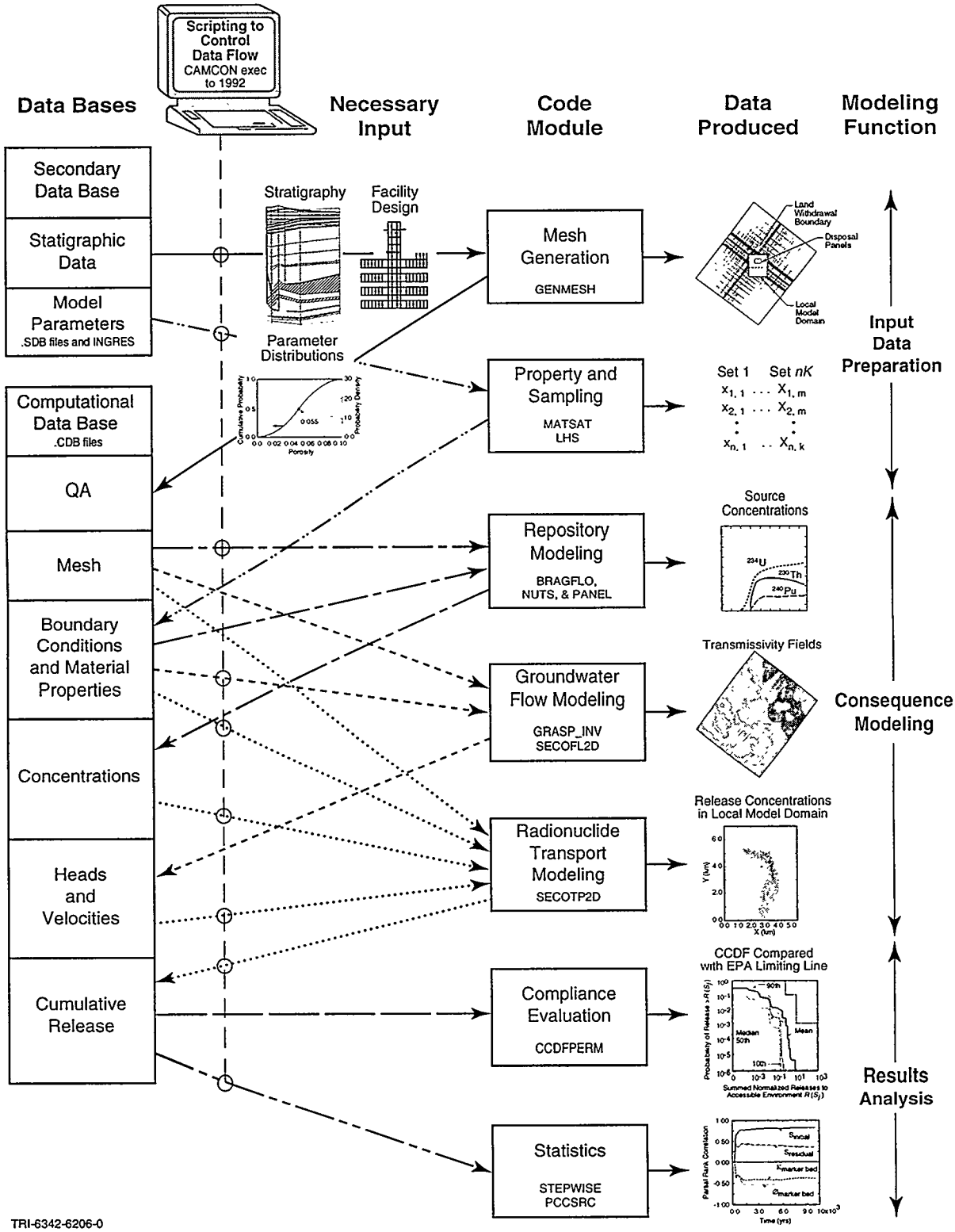


Figure 7-2. Schematic of CAMCON model system completed in 1991. By 1996, CAMCONexec was replaced by computer specialists for run control and management of files (Rechard, 1995, Figure 3.3-4).

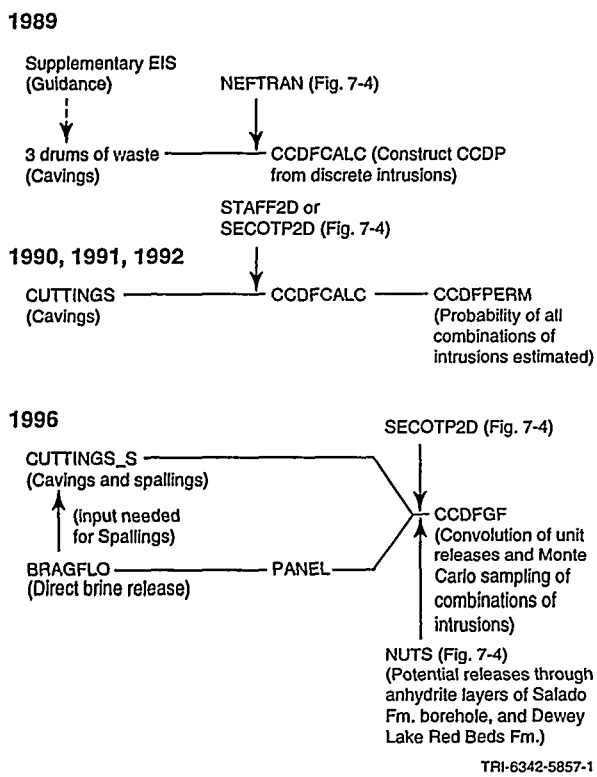


Figure 7-3. Changes in components of the exposure pathway ("consequence") model for immediate releases after exploratory drilling for evaluating compliance of the WIPP.

7.4.1 Direct Cutting and Brine Release Modeling

Direct Cuttings Modeling. For the 1989 PA, the direct release was the same as that estimated for the Supplemental EIS (i.e., three drums of waste). However by 1990, the CUTTINGS model (Rechard, ed., 1992) had been developed to calculate releases caused by erosion of material from the sides of the borehole from standard rotary drilling and was used through 1992. CUTTINGS could estimate erosional removal assuming either laminar or turbulent flow in the annulus between the drill stem and borehole, based on the Reynold's number. CUTTINGS also decayed the radioisotopes based on the time of intrusion into the repository.

Spalling Modeling. Concurrent with the 1992 PA (but not part of the 1992 PA), the quantity of waste particulates released through the spalling of the borehole wall was estimated, i.e., waste released due to the movement of high-pressure gas generated from degradation of containers and organic material to the borehole when the repository was penetrated (Berglund, 1992). Several types of calculations were made and an experimental program was begun to more thoroughly understand the phenomenon. A conservative estimate was made initially and included in the first version of the PA for the CCA submitted in October 1996. However, in reviewing the CCA, the EPA found that the spallings model, though conservative, lacked sufficient realism to be easily defended scientifically; hence, they requested redevelopment in the supplemental analysis for the CCA. The mathematical model for spallings was included in the 1996 PA as CUTTINGS_S.

Direct Brine Release Modeling. In response to requests since 1989 by the EEG, Sandia evaluated in the 1996 PA the potential release of contaminated brine to the surface during drilling. The WIPP two-phase flow code BRAGFLO was used to simulate the direct brine releases. However, to more accurately capture the flow patterns associated with direct releases of short duration, a conceptual model different from the repository model described below was constructed to represent the excavated rooms, drift passageways, and salt pillars. The actinide source term model, PANEL, described later, was used to estimate the activity of radioisotopes in the brine released (Figure 7-3).

7.4.2 Repository Fluid Flow Modeling

Unlike the EIS analysis, the intrusion borehole(s) and repository in the PAs were represented as a model component, $M_\ell(\bullet)$ separate from the source term model. The primary purpose of the repository fluid flow model was to estimate the movement of fluids, both brine and gas, into and out of the repository. In the 1989 PA, the repository, shafts, and intrusion boreholes were represented as distinct legs in the network model NEFTRAN (Figure 7-4). In 1990, the movement of brine into the repository from the Salado was estimated using the single fluid phase finite-element code, SUTRA (Voss, 1984). Another code, PANEL (Rechard, ed., 1992), estimated, through solution of ordinary differential equations, the time history of flow from the hypothesized underlying brine reservoir in the Castile into the Culebra.

Prior to and during 1990, Sandia explored the behavior of fluid flow and radioisotope migration in and around the repository, including (a) gas flow from the disposal area to the shaft using the two-phase code BOAST (Fanchi et al., 1987), (b) Salado brine flow through a panel to a borehole using SUTRA, (c) effects of anhydrite layers on Salado brine flow through a panel

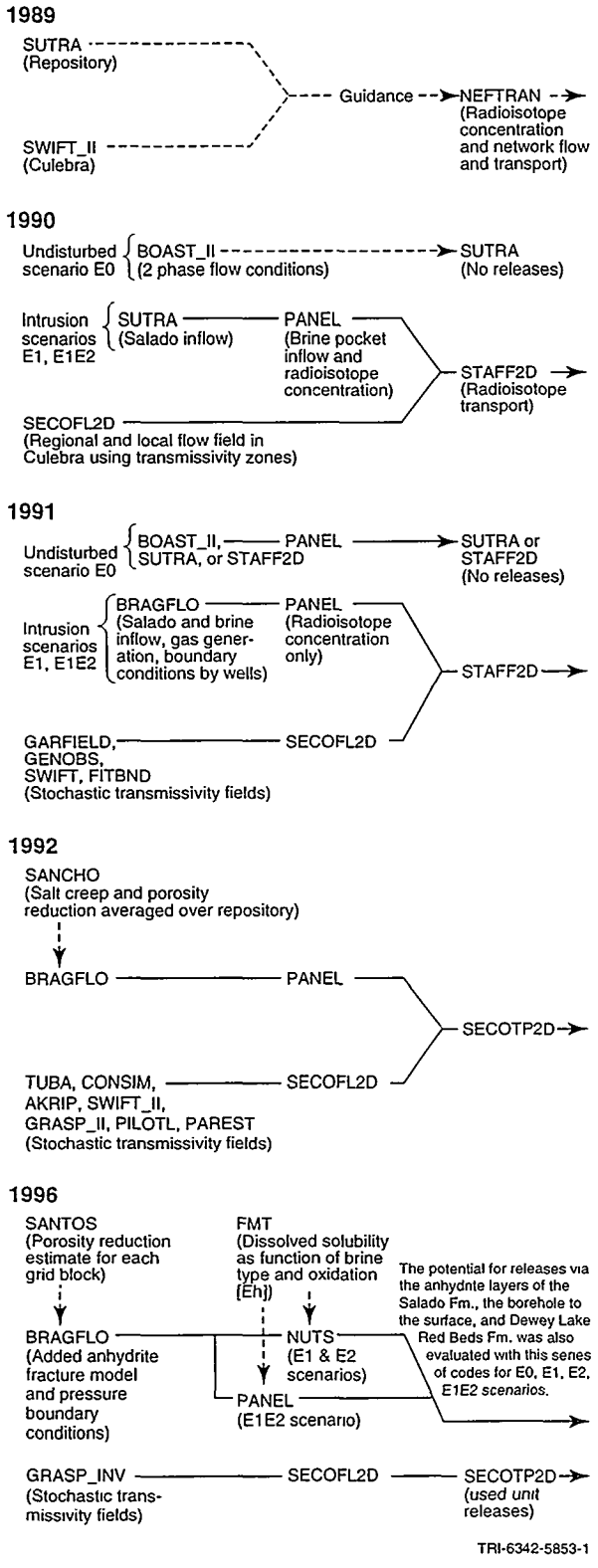


Figure 7-4. Changes in components of the exposure pathway model for long-term releases via leaking borehole and Culebra brine aquifer.

using SUTRA, and (d) flow from a brine reservoir through a disposal room using PANEL (Rechard et al., 1990). A useful concept from this exploration was the practice of varying the size of the gridblocks in the third dimension, i.e., the direction perpendicular to the plane of the grid, in an otherwise two-dimensional numerical mesh. This "element flaring" allowed the analyst to approximately account for volume changes and thereby approximate the very small borehole or shaft, moderate repository volume, or the much larger volumes beyond the repository. The technique had evolved into a fine art by 1992 and was carried forward to the 1996 PA.

Although SUTRA was again used in 1991 for the undisturbed scenario to estimate flow of brine in the repository, all estimates of brine inflow for the human-intrusion scenarios were calculated by the newly developed two-phase fluid flow code, BRAGFLO (i.e., PANEL and BOAST were replaced). The roots of BRAGFLO formulation are in TSRS, a multiphase multicompositional thermal reservoir simulator developed for the DOE for modeling in situ processing of tar sand (Vaughn, 1986). BRAGFLO was developed with a fully implicit numerical formulation because no other code in the public domain, including BOAST, was then available for simulating the convergent flow of gas and brine to the intrusion borehole. Also, a gas generation submodel was incorporated into the 1991 version of BRAGFLO to account for gas generated by the anoxic corrosion of metals and the degradation of organic material in the TRU waste.

In 1992, alternative models of capillary pressure and relative permeability of the salt (Brooks-Corey and van Genuchten) were included through sampling (WIPP PA Department, 1992-1993, Vol. 3). Also, the effects of salt creep, which reduced porosity in the repository, were incorporated by using a generalized porosity reduction surface abstracted from numerous simulations using the salt creep code, SANCHO (Stone et al., 1985), previously developed for characterizing the WIPP facility (see Section 5.3). This surface was refined using SANTOS (Stone, 1997) and used for each grid block in the disposal area in 1996. Also in 1996, a submodel was added to account for brittle fracture of the anhydrite layers caused by pressure buildup from gas generated in the repository.

7.4.3 Source Term Modeling

The 1989 PA used a source-term submodel in the transport code NEFTRAN, as in the Supplemental EIS (Lappin et al., eds., 1989), to evaluate the radioisotope

concentrations released into the Culebra. However, the source term model quickly evolved from a submodel within NEFTRAN to a separate, lumped parameter ("mixing cell") model, PANEL, in 1990 (Rechard, ed., 1992). PANEL determined radioisotope concentrations based on solubility limits and decayed inventory values based on brine passing through a specified volume (e.g., disposal panel). PANEL had the capability to either internally estimate the brine flow or read in external estimates. As briefly mentioned in the previous section, in 1990, PANEL internally estimated the brine flow from the hypothesized underlying brine reservoir, but used an external estimate by SUTRA for the brine flow from the Salado. In the 1991 PA and thereafter, however, only the features in PANEL (WIPP PA Department, 1996) for solubility limits and decayed inventory were used; all brine flow estimates into the repository were calculated by BRAGFLO.

The 1996 PA used the code FMT (Babb and Novak, 1995) to evaluate radioisotope solubility for the E1 and E2 scenarios as a function of oxidation state of the radioisotopes, based on the oxidation capability (Eh) of the repository and the type of brine dominating the water chemistry (i.e., ionic strength and dominate constituents) (Figure 7-4). The dissolved concentrations of radioisotopes as evaluated by FMT were combined with the concentration estimates of four categories of colloids (mineral, intrinsic, microbial, and humic) for input to the finite difference code, NUTS (Stockman et al., 1996), in its evaluation of radioisotope transport within the repository.

7.4.4 Culebra Transmissivity Modeling

The transmissivity parameter of the Culebra, i.e., hydraulic conductivity times strata thickness, varies spatially across the region surrounding the WIPP site. Incorporating the uncertainty of this continuously distributed parameter was necessary to properly evaluate the uncertainty in the PA results. However, the tools to incorporate this type of parameter uncertainty did not exist within the WIPP Project in 1989 and had to be developed.

Zonation of Culebra in 1989 and 1990. Although a calibrated spatially distributed transmissivity distribution had been developed for the Supplemental EIS (Lappin et al., eds., 1989), it represented only one of several possibilities. To propagate the uncertainty represented in the transmissivity field required developing numerous calibrated fields, which would have had to have been done manually in 1989 or 1990 and so was not feasible. Instead, the 1989 PA used the one-dimensional network code, NEFTRAN, and divided the Culebra into different legs, each with a different transmissivity distribution. Similarly, the 1990 PA used a two-dimensional finite difference code specifically developed for the WIPP, SECOFL2D (Roache, 1993; Rechard, ed., 1992), and divided the Culebra into either 8 or 13 fixed zones. Uncertainty ranges of transmissivity were developed solely from well measurements from each zone in the first case, or well measurements and pilot points of the calibrated fields from the Supplemental EIS (LaVenue et al., 1990; Lappin et al., eds., 1989). In both years, the ranges of transmissivity distributions did not overlap between zones and the distributions of each zone were not correlated.

Culebra Transmissivity Fields in 1991. In 1991, the PA group at Sandia devised a relatively simple process to generate numerous transmissivity fields that agreed with estimated transmissivity measurements in wells and, when used as input to a fluid flow code, would generate aquifer pressures that reasonably matched known pressures (or "heads") in wells around the WIPP, i.e., the fields were "conditioned" or "made coherent" with measured transmissivity and well pressure data. First, transmissivity fields were generated (with a code, GARFIELD) (Rechard, ed., 1992). Next, randomly measured transmissivity fields were conditioned with actual measurements of transmissivity. The fields were further indirectly conditioned with the measured head data by evaluating the sensitivity of changes in the specified heads at the model boundary (with GENOBS and SWIFT II) (Rechard, ed., 1992), and appropriate fixed boundary heads were assigned. The transmissivity fields were then ranked by estimated travel time from a point directly above the disposal panels to the 2.4-km boundary of the accessible environment and then randomly selected as input for the fluid flow calculations using SECOFL2D (Figure 7-4).

Culebra Transmissivity Fields in 1992. Sandia convened an expert working group that met in 1991 and 1992 to provide advice on various ways to propagate the uncertainty represented in the transmissivity fields of the Culebra (Zimmerman and Gallegos, 1993). Based on discussion within this group, the original method of Haug et al. (1987) and LaVenue et al. (1990) was automated by 1992, which made the procedure feasible for use in a PA. First, multiple transmissivity fields were generated (using TUBA [LaVenue and RamaRao, 1992]) and conditioned on transmissivity data as in 1991 (but using CONSIM [LaVenue and RamaRao, 1992]). The fields were then conditioned directly on steady state and transient head data by the technique originally used

for the Supplemental EIS (i.e., pilot points) (LaVenue et al., 1990). Pilot points were automatically located (PILOTL) and assigned transmissivity values (PAREST) using an optimization routine (GRASP II) (LaVenue and RamaRao, 1992). By 1996, this series of codes was tightly coupled and referred to as GRASP-INV (Figure 7-4).

7.4.5 Culebra Flow and Transport

For flow and transport in the Culebra, all of the WIPP PAs calculated the fluid flow field assuming a single-porosity Culebra aquifer, but then estimated radioisotope migration through this flow field, assuming advective transport in fractures and diffusion into the surrounding matrix. The flow and transport models changed from the two-dimensional flow evaluation and one-dimensional transport evaluation for the Supplemental EIS to a two-dimensional flow and transport evaluation in 1992.

Culebra Flow and Transport in 1989 and 1990. In 1989, the analysis of fluid flow and transport with SWIFT II and NEFTRAN was similar to that used in the Supplemental EIS although uncertainty was evaluated for the performance assessment. In 1990, SECOFL2D was used to evaluate numerous flow fields based on sampled values for various parameters, e.g., parameter values for each transmissivity zone as mentioned earlier. The two-dimensional, finite-element code, STAFF2D (Huyakorn et al., 1991), was used to evaluate radioisotope transport within the Culebra to the WIPP site boundary at ~2.3 km.

Culebra Flow and Transport in 1991. The 1991 PA rotated the model mesh 38° from a north-south orientation to align one boundary of the mesh with the axis of Nash Draw such that a no-flow boundary could be specified along a portion of that boundary. In addition, the northeastern corner of the model was treated as a no–flow boundary because of the low transmissivities in the area and the lack of any nearby wells to provide head estimates. SECOFL2D was used for fluid flow and STAFF2D for radioisotope transport. Three conceptual models were compared (*ceteris paribus* rather than sampling model weights): matrix transport only, fracture transport only, and fracture transport with matrix diffusion.

Culebra Flow and Transport in 1992. The 1992 PA used SECOFL2D to evaluate fluid flow in the Culebra, but radioisotope transport was evaluated with the newly developed, two-dimensional, finite-difference code, SECOTP2D (Roache, 1993; Ramsey et al., 1996). SECOTP2D easily read the flow fields calculated by SECOFL2D. As in 1991 analyses, the 1992 PA considered both single-porosity (fracture-flow only) and dual-porosity (fracture flow with matrix diffusion). Although fracture spacing was sampled in each simulation, only a single spacing was assigned to the entire aquifer. The distribution of fracture spacing was weighted heavily toward large values, and the calculations assumed an effective thickness of the Culebra equal to its total thickness (7 m).

Culebra Flow and Transport in 1996. The same codes (SECOFL2D and SECOTP2D) were used in the 1996 PA, but between 1992 and 1996, the calculational procedure was modified (unit releases were evaluated for transport and convoluted with actual releases) and the hydrologic and transport parameters of the Culebra were refined and used in calibrating flow fields. The refined parameter values were based on information from the tracer test at the new H-19 well, additional measurements at H-11, reevaluation of transmissivity and tracer measurements at H-3 and H-6 (see Section 3.4), and measurements from the DOE's Water Quality Sampling Program conducted annually around the site. Also, the effective thickness for the Culebra in the 1996 PA was set at 4 m (Figure 3-6). Currently, Sandia has concluded that the Culebra is adequately represented by a dual-porosity continuum model on the scale of PA calculations.

8.0 Probability Evaluation and Sensitivity Analysis

In general, three elements are required for a stochastic model simulation in a performance assessment (Rechard, 1995; Tierney and Rechard, 1997): a consequence model, $R(\bullet)$, which was discussed in the previous section; a space of model parameters, $\mathbf{x} = \{x_1, x_2, \ldots x_{nP}\}$, which was conceptually developed during system characterization and hazard identification; and a joint cumulative distribution function of model parameters, $F(\mathbf{x})$. However, given that the parameter space was divided into two disjoint parts (those parameters associated with scenarios and the remaining parameters, i.e., $\mathbf{x} = [\mathbf{x}^s, \mathbf{x}^p]$), two types of probability evaluations were necessary for the WIPP PAs. Using scenarios for the model parameters, or stochastic uncertainty, was first attempted for the initial WIPP EIS in 1979 (Bingham and Barr, 1979, 1980). Probabilistic descriptions for the model parameters, or subjective uncertainty, was not attempted until the 1989 PA (Marietta et al., 1989).

8.1 Scenario Probabilities and Parameter Selection for the EIS

8.1.1 Scenario Probabilities

The probability of a scenario occurring was evaluated to screen out those with low probabilities. The use of fault trees to develop scenarios or calculate probabilities was found to be impractical during preparations for the EIS and therefore abandoned (Bartlett et al., 1977; Bingham and Barr, 1979, 1980). Hence, the probability models for screening scenarios for the EIS were mostly subjective judgments. The remaining scenarios were grouped into three scenario classes, which were evaluated by means of consequence models. Typically, the probabilities were estimated for three or four time periods: 10^3, 10^4, 10^5, and sometimes 10^6 yr (Bingham and Barr, 1979, 1980). However, some probability models were created from measured failure rate data. For example, historical "failure rates" based on estimates of meteorites striking the earth, extreme erosion rates of land masses, geometrical arguments on probability of striking buried canisters, or faults intersecting the repository were all used.

8.1.2 Parameter Selection

Like the *Reactor Safety Study* (Rasmussen, 1975) conducted only four years earlier, there was no attempt to evaluate how the epistemic uncertainty in the model parameters contributed to the uncertainty in results for the original EIS. Uncertainty was evaluated only using scenarios. Furthermore, the EIS did not attempt to combine the various conditional consequences into an overall distribution. Model parameters were selected for the EIS for each scenario independent of other scenarios. The overall philosophy was to present conservative results when possible. Thus, different values for the same parameter sets might be used for separate hazards in order to maximize the consequences.

8.2 Scenario Probabilities for Performance Assessments

As described earlier, PA analysts continued to define a few scenarios to simplify modeling and call attention to human intrusion through exploratory drilling as specifically identified in 40 CFR 191. Therefore, the probability of the scenarios, $P_r(S_j)$, had to be calculated by some method. In concept, the probability model for a scenario evaluates the probability that parameters lie in a subset of the parameter space that defines the scenario. Hence in theory, the distribution of all the parameters, $F(p)$, of the scenario can be used to define the scenario probability, $P_r(S_j)$. However, in the 1989 PA, the probabilities of various scenarios were based on subjective judgment, with no ranges of uncertainty as in the 1979 EIS.

After 1989, the inadvertent human intrusion event was assumed to be a Poisson process and so the probability of various numbers and combinations of intrusions was analytically calculated through the Poisson probability density function (Helton, 1993c). Usually the Poisson process was assumed to have a constant expected rate of intrusion, λ, over the 10,000-yr regulatory period. The intrusion rate was constant throughout any one simulation in the preliminary performance assessments between 1989 and 1992. However, a different value between 0 and the maximum value λ_{max} of 30 boreholes/km^2 per 10,000 yr was selected for each of the many simulations, and thereby accounted for uncertainty in scenario probabilities. The probability of all permutations of intrusion geometry (e.g., one intrusion only, two intrusions into one panel of the repository, two intrusions into two different panels, etc.) and permutations of fixed intrusion times (e.g., at 2000-yr intervals for a groundwater pathway as in 1991 or only one at

1000 yr as in 1992) was evaluated directly with the code CCDFPERM (see Figure 7-1 in Section 7).

In 1990, as an alternative, the WIPP Project conceptually examined the influence on results when the rate of intrusion was assumed to vary with time, $\lambda(t)$ (Tierney, 1991). For the 1992 PA, an actual function $\lambda(t)$ was constructed based on input from an expert panel that had considered future societies (Hora et al., 1991) and the effectiveness of markers at the site to convey the existence of hazards (Trauth et al., 1993). As a result, the overall number of intrusions decreased dramatically in comparison to a companion 1992 analysis with a constant λ (Helton et al., 1996). In addition, the probability of all permutations of the intrusion geometry and of the intrusion times was no longer evaluated analytically but rather estimated through Monte Carlo sampling procedures (Helton and Shiver, 1996).

In 1996, λ varied with time but used the same function for all simulations. The functions were as follows: (1) $\lambda = 0$ while active institution controls, such as land control, were present, $t < 100$ yr; (2) $\lambda = 0.01\ \lambda_{max}$ while passive institutional controls, such as markers about the WIPP site, were present, 100 yr $< t <$ 700 yr; and (3) $\lambda = \lambda_{max}$ thereafter where the maximum rate of intrusion, λ_{max}, was increased to 48.5 boreholes/km^2 per 10,000 yr based on guidance in 40 CFR 194 (EPA, 1995; 1996a).

8.3 Parameter Uncertainty in Performance Assessments

In 1985, when 40 CFR 191 requested the DOE applicant to "assemble all of the results of the performance assessments to determine compliance with §191.13 into a 'complementary cumulative distribution function'," an important goal for performance assessment became a consistent evaluation of system consequences such that individual consequences and the uncertainty from each could be combined in an overall distribution of the consequences. To address these issues (data consistency, uncertainty description, and uncertainty propagation), the WIPP Project had to develop a traceable system for regulatory review in which distributions for the uncertain parameters could be developed and values for fixed parameters selected.

8.3.1 Data Bases for Model Parameters and Results

In early 1989, the WIPP Project conceptually described three categories of data bases (Rechard, 1989): the primary, secondary, and computational data bases. The several primary data bases held measured field and laboratory data gathered by investigators from experiments during characterization of the WIPP disposal system (e.g., Munson et al., 1990). In general, the information stored in the primary data base was to be controlled by the investigators. The secondary data base contained distributions of parameters that had been derived from the primary data bases specifically for the various component models of the exposure pathway model $q(\bullet)$. The computational data base, generated during each performance assessment, comprised the calculated results. By 1990, the WIPP Project used the latter two data bases directly in the performance assessment calculations; however, the computational database existed only as a collection of catalogued files rather than as a relational database.

8.3.2 Quality Assurance Procedures

For the 1991 and 1992 PAs, the WIPP Project developed rudimentary quality assurance procedures. The purpose was to provide a reasonable degree of assurance to those outside the PA community that the results from the performance assessment process presented a logically consistent view of WIPP performance, based on current knowledge and explicitly identified sources of uncertainty.

The early procedures specified requirements in three primary areas of the analysis process (Rechard, 1995): Parameter Selection (Rechard et al., 1992a), Software (Rechard et al., 1991), and Analysis (Rechard et al., 1992b). In addition, procedures were prepared to ensure quality in two other secondary areas: Report Review and Expert Judgment Panels. These areas were related to the primary areas because, for example, all three primary QA areas required reports and review.

The Parameter QA procedures sought to provide the PA analyst with consistent computational model parameters. The fundamental requirement was the development of a secondary data base managed by a Task Leader responsible for selecting appropriate data in consultation with investigators and PA analysts. Transferring data from investigative or experimental groups to the secondary data base was an important method by

which the PA analysts interacted with WIPP Project investigative groups.

The Software QA procedures were designed to ensure that the software met the expectations of the PA analyst. The fundamental requirement was the development of a Software Management System (the CAMCON Modeling System; see Section 7).

The procedures initially presented in the early 1990s were developed into a full suite of quality assurance procedures for the 1996 PA supporting the CCA.

8.3.3 Parameter Selection

The use of a consistent set of parameters was initiated for the 1989 PA and had become an important aspect by the 1990-1992 calculations. The general procedure used to acquire parameter distributions in the calculations from 1990 through 1992 was as follows (Rechard et al., 1992a; WIPP PA Division, 1991/1992):

1. **Identify Necessary Parameters.** The PA analysts identified parameter sets ($\mathbf{x} = x_1, \ldots, x_{nP}$) that were necessary for PA calculations.

2. **Gather Necessary Underlying Data.** The PA analysts formally requested observational data from appropriate WIPP Project investigators. The investigator may have supplemented these data with additional data and general information from various sources to bridge any data gaps. Occasionally the PA personnel also informally compiled data for preliminary calculations and documented the status of the data.

3. **Construct Parameter Distributions.** Probability distributions were developed to describe uncertain parameters. Based on the information gathered, the PA analysts either constructed parameter distributions or used distributions provided by investigators, as described more fully below.

4. **Update Secondary Data Base.** The endorsed or elicited information on the model parameters was updated or entered in the secondary data base. The model parameters in the database were described formally beginning in 1990 (Rechard et al., 1990; WIPP PA Division, 1991/1992; WIPP PA Department, 1992/1993) and contained 191 parameters. By 1996, the database contained 1561 parameters.

5. **Select Parameters To Be Sampled.** Specific model parameters were chosen for sampling in each performance assessment. All other parameters were kept at their median values, unless specifically noted. An important practical problem for parameter uncertainty analysis was determining the number of uncertain parameters to propagate. As the computational ability increased, the number of uncertain parameters also grew with each assessment: 28 in 1989 (Marietta et al., 1989), 39 in 1990 (Rechard et al., 1990), 46 in 1991 (WIPP PA Division, 1991/1992), 49 in 1992 (WIPP PA Department, 1992/1993), and 57 in 1996 (DOE, 1996a).

8.3.4 Describing Parameter Uncertainty

To evaluate the epistemic (subjective) uncertainty of PA results, a joint cumulative distribution function, $F(\mathbf{x}^p)$, was required that characterized the uncertainty of model parameters, $\mathbf{x}^p = x_1^p, x_2^p, \ldots, x_n^p, \ldots, x_{nU}^p, \ldots, x_{nP}^p$, where nU is the number of uncertain parameters and nP is the total number of parameters. As is standard practice, $F(\mathbf{x}^p)$ was approximated by the product of the cumulative distribution functions of the individual parameters, $F(x_1^p) \bullet F(x_2^p) \bullet \ldots F(x_n^p) \ldots \bullet F(x_{nU}^p)$, an approximation that is exact when the parameters vary independently. Parameter independence was assumed for the preliminary PAs; however in the 1996 PA, very strong correlations (-0.99, -0.99, and -0.75) were specified between two parameters, permeability and bulk compressibility, in three materials (Salado halite, Salado anhydrite, and brine reservoir anhydrite).

The cumulative distribution function, $F_n(x_n)$, of a parameter, x_n, ideally represented what was known and not known about the parameter range and the likelihood that these values were appropriate for consequence or probability models without assuming a "conservative" bias (see Figure 3-4). The avoidance of a conservative bias was an important shift from the philosophy pursued for the EIS (DOE, 1979; 1980a) and Supplemental EIS (DOE, 1989b; 1990c).

Because each parameter distribution function must be tailored to the type of data available and to the parameter's role in the computational models, parameter distribution characterization was not guided by a rigid series of steps. In most cases, each $F_n(x_n)$ included subjective factors representing the "degree of belief" of the WIPP investigators. Beginning in 1990 (Rechard et al., 1990), a maximum entropy formalism was tried and then used extensively by 1991 (WIPP PA Division, 1991/1992) to provide a consistent procedure for

constructing the distributions (Tierney, 1994). In practice, the maximum entropy formalism involved connecting data points or subjectively estimated points with straight lines.

The use of a consistent set of parameters was initiated for the 1989 PA and had become an important aspect by the 1990-1992 calculations. The data preparation code, MATSET (Rechard, ed., 1992), extracted data directly from the secondary data base for use by the modeling codes. This process ensured that the same parameter values were used consistently throughout the calculation. The 1989 PA primarily used parameter values from the supplemental EIS (Lappin et al., eds., 1989). Uniform, normal, lognormal, and beta distributions were fit to available data by the PA analysts as appropriate. Each year thereafter, however, more data were elicited directly from investigators, a process that was formalized in a quality assurance procedure (Rechard et al., 1992a). In some cases, parameters were evaluated through a formal expert panel while experimental data were collected, e.g., values for solubility of actinides in the repository and retardation in the Culebra for the 1991 PA (Trauth et al., 1992; Rechard et al., 1993b).

8.4 Sensitivity Analysis

Sensitivity analysis is the evaluation of aspects of the system that most influence the calculated or observed results. Specifically, a sensitivity analysis determines the uncertain parameters x_n (or model forms, e.g., M_ℓ^m) that most influence the result $R(\mathbf{x})$ and its cumulative distribution function, i.e., $Pr\{R \le r\} = \int \delta[r - R(\mathbf{x}^p)]dF(\mathbf{x}^p)$, where $\delta(\bullet)$ is the delta function (whose integral is zero when the argument is negative and one when the argument is positive), $dF(\mathbf{x}^p)$ is the joint probability function for the $\mathbf{x}^p$ model parameters, r is an arbitrary variable, and the integral is evaluated over the space of uncertain epistemic parameters. A sensitivity analysis can be conducted after the probability, consequence, or compliance steps.

8.4.1 LHS Technique

During the 1940s, the advent of computers allowed new problem-solving techniques to address issues of nuclear weapon design. An important practical tool developed at this time—the Monte Carlo solution technique—was designed to integrate the multidimensional integrals that arose in the study of the physics of weapons and first documented in 1949 (LANL, 1987; Metropolis and Ulam, 1949). But the technique applies to any multidimensional mathematical integration such as determining the distribution of $R(\mathbf{x}^p)$, i.e.,

$$Pr\{R \le r\} \simeq \frac{1}{nK} \sum_{k=1}^{nK} I\left[r - R\left(\mathbf{x}_k^p\right)\right]$$

where $I(\bullet)$ is an indicator function equal to zero when the argument is negative and one when the argument is positive, $\mathbf{x}_k^p$ is a set of sampled parameters drawn from $F\left(\mathbf{x}_k^p\right)$ and nK equals the number of Monte Carlo samples. However, making a large number of samples, as is necessary with the rudimentary Monte Carlo method, is impractical when evaluation of the function $R(\mathbf{x}^p)$ is time consuming, as in the WIPP calculations.

Many procedures have been developed to judiciously sample the domain of parameters to reduce the required total number of samples in a Monte Carlo analysis. A simple scheme developed in 1975, Latin Hypercube Sampling (LHS) (McKay et al., 1979), has been frequently used in the United States in performance assessments and probabilistic risk assessments because sample points are easily selected and there is frequently a good matching of results from more extensive random sampling. The LHS technique for Monte Carlo analysis was developed for a 1975 study to determine the important parameters in a complex code that modeled pipe ruptures in nuclear power plants (McKay et al., 1979). LHS was later applied to a 1980 examination of important parameters of a geologic disposal system (Iman and Conover, 1980).

The robustness of the procedures conjectured in the early 1980s (Iman and Conover, 1980; Iman, 1982) and more thoroughly demonstrated in the later 1980s (Iman and Helton, 1988; 1991) encouraged the WIPP Project to adopt the LHS technique to propagate parameter uncertainty and determine the distribution of $R(\mathbf{x}^p)$ for comparison with 40 CFR 191. Other techniques for sensitivity uncertainty analysis, such as developing surrogate analytic expressions for the results ("response surface development") or differential analysis with normalized partial derivative of parameters (e.g., "adjoint procedure"), were also proposed in the 1980s (Helton, 1993b). However, these techniques have never been used routinely for large-scale sensitivity analyses with several complex and linked submodels. Because the Monte Carlo technique was used to propagate uncertainty in the WIPP analysis, sensitivity of the results, $R(\mathbf{x}^p)$, to changes in parameter values could be approximated and conveniently determined in several ways, including (1) examining scatterplots and (2) developing a statistical regression model and comparing the size of the standardized regression coefficients or the associated

partial correlation coefficient (Helton et al., 1993a, 1996).

8.4.2 Sensitive Parameters

The 1989 PA used sensitivity analysis but the method was ad hoc because CAMCON was not yet ready and so there was no easy method to input parameters and results into regression analysis codes. In the 1989 PA and in all subsequent analyses, radioisotope releases occurred only after inadvertent human intrusion. Hence, out of the 28 parameters sampled in the 1989 PA, the most important parameters were those associated with the human-intrusion scenario: solubility of radioisotopes, the time of intrusion into the repository, and the assumed permeability of the resulting but abandoned borehole. The 1989 PA did not evaluate separately the release of radioisotopes from cuttings brought directly to the surface in the drilling operation because it was set at a constant value of three drums of CH-TRU waste. Simultaneous with the 1989 PA, a sensitivity analysis was conducted on the importance of alternative conceptual models and various modeling techniques for components of the repository submodel, as mentioned previously (Rechard et al., 1990).

With the initial version of CAMCON ready in 1990, regression analysis techniques to determine sensitivity were used (Helton, 1993b). The same three parameters (solubility of radioisotopes, time of intrusion, and assumed permeability of the borehole) were again the most important out of the selected 39 uncertain parameters in the 1990 PA (Bertram-Howery et al., 1990). This performance assessment also evaluated release of radioisotopes from cuttings brought directly to the surface; these releases controlled the shape of the CCDF at probabilities greater than 0.5. Below probabilities of 0.5, radioisotope releases from the repository by means of groundwater transport through the intrusion borehole into the brine aquifer overlying the repository were important.

In the 1991 PA, release of radioisotopes from cuttings was clearly the dominant pathway and again controlled the shape of the CCDF for probabilities greater than 0.5. Out of 46 parameters sampled, two were important for this release: the rate constant (λ) in the Poisson distribution for modeling the rate of human intrusion, and the borehole diameter (d_b). However, because other parameters—solubility of plutonium, uranium, and americium; permeability of the borehole; permeability of the halite surrounding the repository; and retardation distribution coefficients for radioisotopes during groundwater transport—could markedly vary groundwater transport releases, their ranking in importance was higher than the diameter of the intrusion borehole, which influenced the amount of cuttings brought directly to the surface (WIPP PA Division, 1991/1992; Helton et al., 1993b).

In the 1992 PA (and also in 1996), releases from cuttings again dominated total radioisotope release for the mid to highest probabilities of the CCDF. Of 49 parameters, the three most important were the rate constant λ, borehole permeability, and solubility of americium (WIPP PA Department, 1992/1993; Helton et al., 1996). A separate sensitivity study was also conducted to determine parameters important to migration of gas and brine in the vicinity of the repository, because an evaluation of the potential for migration of RCRA hazardous constituents such as volatile organic compounds (VOCs) was still required at this time (Helton et al., 1993b).

8.4.3 Project Guidance from Sensitivity Analyses

For the WIPP Project, the sensitivity analyses helped to (1) verify the correctness of the calculations, i.e., errors were occasionally found when unexpected behavior was examined more thoroughly, (2) gain understanding and insight about the system, and (3) evaluate the influence of various options (Rechard, 1995). In addition, sensitivity analysis, in combination with multiple iterations through the performance assessment process, provided some guidance to project managers on how to direct resources for the collection of information about significant model parameter values and model forms, based on what was already known about the site or waste. In 1990 and 1991 some general guidance was provided (Helton et al., 1991; 1992), as mentioned previously. In the 1992 WIPP PA (and also in 1996), releases from cuttings again dominated total radioisotope release for the mid to highest probabilities of the CCDF. Of 49 parameters, the three most important were the rate constant, borehole permeability, and solubility of americium (WIPP PA Department, 1992/1993, Vol. 3; Helton et al., 1993a).

Thus, by 1992 it was evident that required regulatory assumptions about human intrusion were dominating the results. Hence, continued, extensive evaluation of the characteristics of the disposal system was not considered to be warranted, except for specific areas such as an evaluation of radioisotope solubilities in the repository, retardation distribution coefficients, and alternate conceptual models for transport in the

overlying brine aquifer. These results coincided with a decision by the DOE, in concert with urging from the NAS, to forego in situ testing. Thus the DOE, which was also undergoing a change in administration, decided to omit the pilot phase of the WIPP and move ahead to the compliance phase, limiting testing to only that required with regard to compliance.

Beginning in 1994, an effort was made to combine the PA process directly with decision analysis in order to more definitively determine the best combination of scientific investigations, engineered alternatives, and waste acceptance criteria to support the CCA. The first attempt, called the System Prioritization Methodology (SPM), began in March 1994. The calculations and decision analysis were completed in December (Helton et al., 1997a; 1997b). A second iteration of the methodology was conducted in 1995 (Prindle et al., 1996). The new process produced additional information and thus helped the new DOE management team at the WIPP to allocate resources in 1995. However, the SPM cost much more in time and money than a general sensitivity analysis and the additional information it supplied mainly confirmed earlier sensitivity studies. In addition, basic tenants of decision analysis, such as developing an explicit utility function, were not followed (Lee, 1996). In its practical application, the analysis was not probabilistic because the time needed to run a sufficient simulation would have been excessive. Only a deterministic simulation of each activity, using an ad hoc combination of mean and median parameter values, was run.

9.0 Compliance Assessment and Summary

An important difference between risk assessments or other large-scale policy analyses (Morgan et al., 1990), which are usually conducted to elucidate understanding of the behavior of a system, and the WIPP PAs is that the WIPP PAs were specifically designed to test *compliance* to a set of standards so that decisions could be made about safety, rather than just to elucidate understanding.

9.1 Assessing Compliance of the WIPP

The focus of the extensive engineering analyses conducted over the years by the WIPP Project was initially the National Environmental Policy Act (NEPA) of 1969 and later the regulations in 40 CFR 191, Subpart B (EPA, 1993b) and 40 CFR 194 (EPA, 1996a; 1996b). The EPA regulatory criteria are the societal consensus on what constitutes acceptable risk for radioactive waste disposal to be considered safe (Lowrance, 1976). Internationally, this last step of a performance assessment is often called a probabilistic safety assessment (NEA, 1991) or safety assessment (when uncertainty is omitted).

9.1.1 Predicted Doses for EIS

For each EIS for the WIPP, dose to an individual has been calculated as one of several measures of evaluating the impact of the repository on humans and the natural environment. In all EISs, doses immediately after human intrusion dominated the results. The predicted dose to the geologist examining a 7.6-cm mineral core was estimated to be 1 mrem in 1979 (DOE, 1979); 0.08 mrem in 1989 (Lappin et al., eds., 1989); and 0.01 mrem in 1996, assuming the core extracted CH-TRU waste (DOE, 1996b). The predicted dose to a maximally exposed driller ingesting fragments was 0.37 mrem in 1996. In addition, a separate probabilistic analysis of potential radiation exposure was conducted in 1996 (Helton et al., 1998). The predicted dose (primarily through inhalation) to a farm family 500 m downwind from the drilling mud pit was 0.036 mrem in 1979 (DOE, 1979); the maximum committed dose equivalent over 50 yr from inhalation to the farm family was ~0.77 mrem in 1989 (Lappin et al., eds., 1989). No calculations were conducted for the farm family in the 1996 EIS (DOE, 1996b).

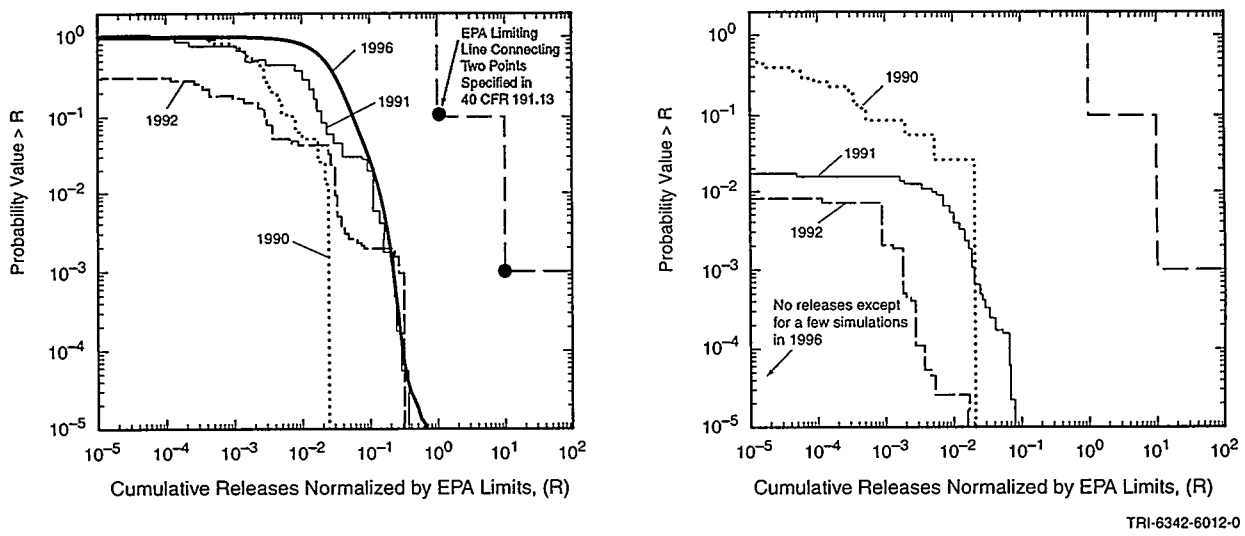
9.1.2 Releases from Preliminary Performance Assessments

Preliminary results in the form of CCDFs from the performance assessments were compared with the EPA regulations between 1989-1992 (Figure 9-1). Uncertainty has decreased somewhat, as indicated by the CCDFs becoming more vertical yet, over the years, the predicted cumulative releases immediately after drilling into the repository have remained similar even as more mechanisms for release were added (i.e., spallings and direct brine release in 1996). The releases have remained similar because the general required assumptions for exploratory drilling were specified in the regulations. Only when drilling assumptions were changed based on results from a "futures panel" were drilling releases dramatically reduced (see Section 8.2).

The cumulative releases from leakage through the intrusion borehole and into the brine aquifer in the Culebra, a pathway much more dependent upon scientific knowledge about the WIPP disposal system, decreased until they were practically nonexistent in 1996. Although more scientific knowledge could be acquired about the WIPP disposal system, the results, in the form of cumulative releases, currently are contingent on the modeling assumptions or "style" required by the EPA regulations. That is, the regulations require inclusion of inadvertent human intrusion through exploratory drilling using current technology, and this regulatory requirement now directly determines the maximum releases. Continued collection of information about the disposal system is not likely to change the estimated overall releases substantially (Figure 9-1); hence, the EPA requirement that the human intrusion event be included indirectly defines the point at which disposal system characterization is sufficient.

9.1.3 Performance Assessment for CCA

In October 1996, the calculations and description of the 1996 PA were completed for the 80,000-page Compliance Certification Application (CCA) (DOE, 1996a). The overall exposure pathway model, $q(\bullet)$, for the 1996 PA was run 100 times with LHS samples. Furthermore, the 100 LHS samples were 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



(a) Direct Releases to the Surface during Drilling

(b) Releases via Groundwater Pathway through Culebra

Figure 9-1. Changes in the distribution of cumulative releases (normalized by the EPA limits) for (a) direct releases to the surface during drilling and (b) releases via groundwater pathway through Culebra. Since the first full PA in 1990, direct releases to the surface have dominated overall releases to the accessible environment of the WIPP.

pathways (e.g., direct, groundwater, anhydrite beds), scenarios (E0, E1, E2, E2E1), and times of intrusion (e.g., 350 and 1000 yr for BRAGFLO for E1 and E2 scenarios) used in the analysis. In total, the calculations ran on 40 DEC Alpha™ processors for 37,000 CPU hours; although Tbytes of data were created, ~100 Gbytes of data were retained in 97,000 files. The releases, which were solely from drill cuttings, cavings, spallings, and direct brine flow immediately after intrusion, showed compliance with 40 CFR 191 (Figure 9-1).

Throughout the winter and spring of 1997 (during the first six months after submittal), additional PA analysis and documentation, totaling 20,000 pages, were provided to the EPA at its request. For example, the EPA conducted an extensive review of the justifications for the parameter values and mathematical models used in the analysis, some of which required elaboration. In addition, before final submittal of the CCA, a few EPA staff members and several contractors had spent about two years becoming familiar with the models and assumptions of the WIPP PA. Separate from the formal review of the CCA but as part of the EPA's evaluation, the EPA also directed the DOE to run an abbreviated version of the PA in March 1997, varying models and parameters to bolster confidence in the WIPP disposal system. In May, the EPA directed the DOE to conduct a PA verification test PA using EPA's own selected modeling assumptions together with changes in distributions for 26 parameters (MacKinnon et al., 1997). The EPA declared the DOE's application complete in May 1997.

9.2 Summary

Over the decades, the United States has progressed from the burial of solid nuclear waste in shallow trenches and augered holes by the Manhattan Engineering District to the concept of deep geologic disposal. The examination of radioactive waste disposal in general and geologic disposal at the WIPP in particular started with the informal generic hazard identification by the NAS in 1955, followed by a selection of the disposal mode (deep geologic) to eliminate or mitigate those identified general hazards in some way. After repeated recommendations by the NAS for disposal in salt, the large salt beds of the Delaware Basin in southeastern New Mexico were selected in 1973 for the WIPP.

After the selection of the Delaware Basin and a specific site, ORNL evaluated the consequences of those

hazards that remained in 1974 (e.g., meteorite impact, catastrophic faulting, etc.). A more formal hazard identification occurred in conjunction with the draft and final EIS of 1979 and 1980. Probabilities and consequences were evaluated only on aspects of the WIPP disposal system that were pertinent to the hazard under evaluation, i.e., a "scenario". Hazards (e.g., sabotage or nuclear criticality) with very small subjective estimates of probability or consequences were eliminated. Then consequences of the remaining hazards (e.g., exposure to drill cuttings containing TRU waste) were formally estimated to determine whether those remaining hazards were sufficient to disqualify the site. The Draft EIS predicted no significant adverse long-term environmental impacts to human health that would support arguments against construction in 1979. At the same time, Congress authorized the DOE to build the WIPP near Carlsbad, New Mexico, for disposal of waste contaminated with TRU radioisotopes six years after the site selection. After an exploratory phase in 1981, full construction of the WIPP began in 1983 and was officially complete by 1990 (Figure 9-2).

Modeling a nuclear waste disposal system presented not only new technical challenges for assessment but also societal challenges in developing a consensus on the criteria under which a disposal system would be accepted. Consequently, the interplay between developing criteria and creating a corresponding assessment technique was important. During this time, analysts at Sandia remained advocates for a comprehensive assessment approach, which was the indirect result of their participation on the 60-member *Reactor Safety Study* team between 1972 and 1975 and continued involvement with several reactor accident studies for the NRC (which culminated in the update to the Reactor Safety Study in 1990). Eventually, the method that was conceived and accepted by the engineering community and the EPA, as regulator, was a probabilistic risk assessment as specified in the criteria promulgated in 1985.

In 1986, Sandia accepted the task of evaluating the compliance of the WIPP with the EPA regulation and began construction of an analysis system for the required probabilistic simulations in about 1988 just as construction was being completed (Figure 9-3). An important goal of the analyses was to consistently examine hazards that remained after screening such that individual consequences from each could be combined in an overall distribution, i.e., the entire distribution of consequences was desired not just the expected value. Also, WIPP analysts found that explicitly defining scenarios, as had been done for the EIS, remained useful both to simplify modeling and to call attention to inadvertent human intrusion of the disposal system as specified by the EPA.

Between 1989 and 1992, Sandia conducted four preliminary performance assessments. Each iteration examined the behavior of the WIPP repository, based on current understanding, incorporated uncertainty, and then compared its preliminary results to the regulatory criteria. Because the physical and chemical processes that determine the behavior and evolution of the disposal system are complex, many of the models that represented the physical and chemical processes had become technically sophisticated by the 1992 PA and were carried forward for the 1996 PA.

In October 1996, the EPA began its review of the CCA for the WIPP. In October 1997, one year later, the EPA proposed in the *Federal Register* to certify that the WIPP disposal concept, based on results from the 1996 PA, complied with its regulations (EPA, 1997). In May 1998, 19 years after Congressional authorization, the EPA issued the certification (EPA, 1998). The State of New Mexico has not yet issued a RCRA permit for 60% of the TRU waste that also contains hazardous waste, but in mid 1999, the state began the necessary hearings. The permit should be issued toward the end of 1999. A few lawsuits are still pending; however, four days after Judge Penn lifted his injunction associated with a 1992 lawsuit by the State of New Mexico, the WIPP repository received its first shipment of non-RCRA waste from Los Alamos National Laboratory (Taugher and Smallwood, 1999). This auspicious opening on March 26, 1999, was the culmination of 25 years of evaluation.

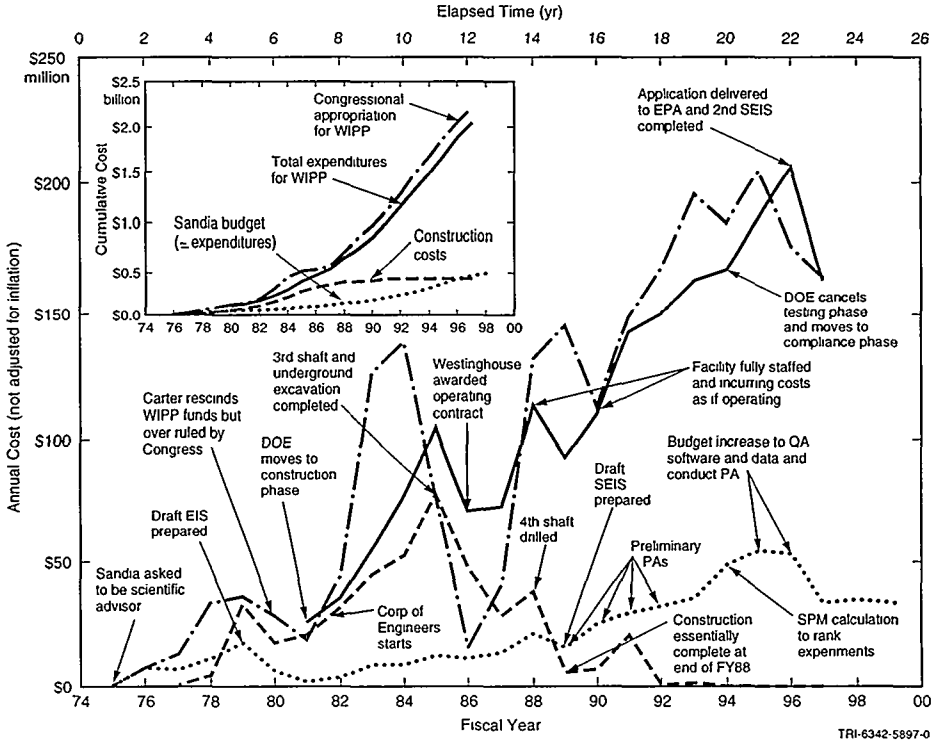


Figure 9-2. Annual outlays for construction and assessments of the Waste Isolation Pilot Plant (WIPP) (Sandia Corporation budget records).

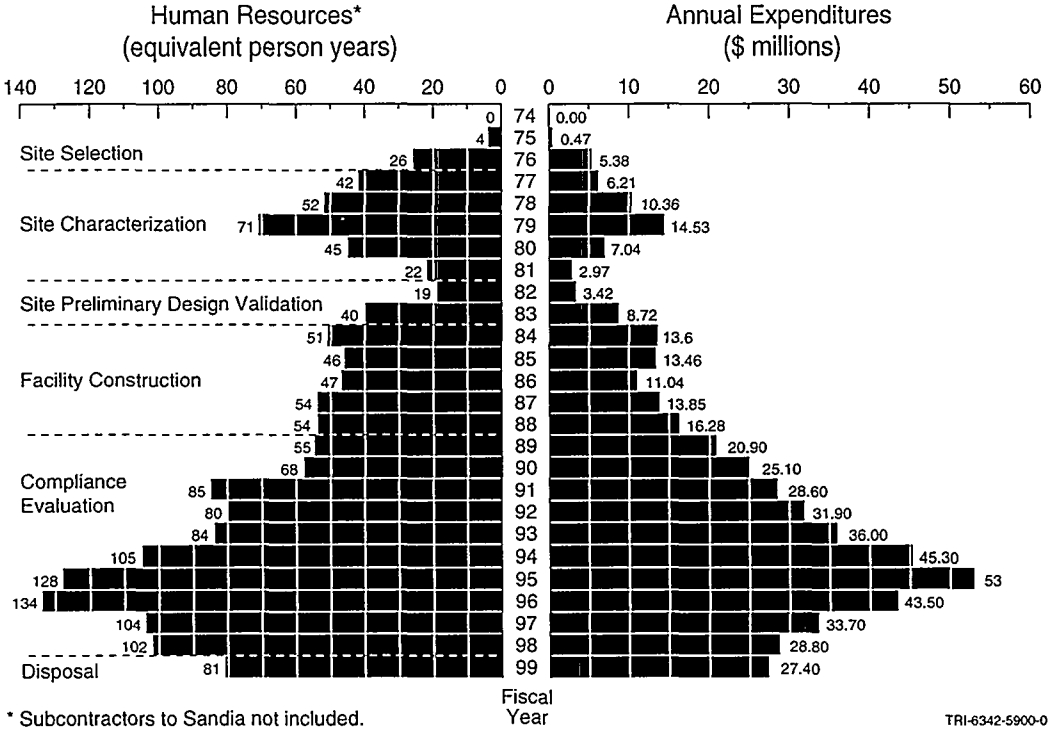


Figure 9-3. History of expenditures and human resources for WIPP Project at Sandia National Laboratories (Sandia Corporation budget records).

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(Note: Reference citations for Figures 2-1, 2-2, 3-5, 4-1, 5-2, and 6-2 are provided after the main reference list.)

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