Historical Background on the Performance Assessment for 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, which is summarized in this special issue of *Reliability Engineering and System Safety*. This performance assessment was the culmination of four preliminary performance assessments conducted between 1989 and 1992. This paper 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.

Key Words: performance assessment; probabilistic risk assessment; risk assessment; radioactive waste disposal; Waste Isolation Pilot Plant; history

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 paper presents a historical summary of the system characterization, engineering analyses, and scientific investigations undertaken by the WIPP Project over the past 25 years and provides a context for the results presented in this special issue of *Reliability Engineering and System Safety*. Although it is not essential to know the WIPP's history to comprehend the engineering analysis, an understanding of the issues that the project faced can be helpful. Many paths were followed during these 25 years 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.

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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 as the location (including what are known now as Los Alamos National Laboratory and Sandia National Laboratories) for assembling the scientists, engineers, and technicians who would develop the first atomic bomb. 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 (Gard, 1968).

After a potential disposal site in bedded salt in Kansas had been rejected in the early 1970s, citizens invited the Atomic Energy Commission (precursor to the U.S. Department of Energy¹) to New Mexico to consider the bedded salt deposits in the Delaware Basin. 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).

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) had been segregating and storing these wastes above ground for eventual disposal since 1970, three years before site selection (Perge, 1982). Nearly ten years later, in December 1979, Congress authorized the DOE to construct the WIPP for eventual disposal of this TRU waste (Public Law 96-164). The first shaft was drilled in 1981, and full construction began in 1983. Construction of the WIPP facility was substantially complete by the end of 1988.

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 (PA), an engineering analysis documented in the Compliance Certification Application (CCA), which was submitted to the EPA in October 1996 (DOE, 1996a) and which is described in this special issue of *Reliability Engineering and System Safety*.

1.2 Compliance Assessment of Waste Isolation Pilot Plant

Over the past 25 years, the process for assessing performance of a deep geologic repository for nuclear waste developed concurrently with the characterization of the WIPP in New Mexico. The WIPP project's first major analysis was for the EIS in 1978. In 1985, the EPA promulgated its radiation protection standard for the management

¹ The Atomic Energy Commission (AEC) was formed by the *Atomic Energy Act of 1946* (Public Law 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).

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).

The overall process of assessing whether a nuclear waste disposal system meets a set of performance criteria is known as a performance assessment (PA), a term defined in 40 CFR 191. Similar to other risk assessments, a PA 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; and (6) sensitivity analysis. In general, a consequence analysis consists of (a) a dose-response assessment, which evaluates the response of a receptor (or system) to a hazard (or stressor), and (b) an exposure pathway assessment, which evaluates the exposure intensity of a hazard that reaches a receptor, e.g., a human. As with any scientific modeling or policy process, steps may overlap. More importantly, an analyst may need to cycle through several of the early 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 Rechar (1995).

Because the main purpose of a PA 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 analysis, which is in principle constrained only by human curiosity.

1.3 Organization of Historical Material

Historical events described here are generally categorized according to the steps of a PA. 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. The engineering analysis of the WIPP disposal system is then presented according to the remaining steps of a PA.

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 AEC also undertook scientific and engineering studies for deep geologic disposal in salt, which 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 for determining 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 Los Alamos National Laboratory, the site and methods for disposal of any 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 any 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.

The AEC, formed in 1946 (Public Law 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 at Idaho National Engineering and Environmental Laboratory (INEEL) in 1952.

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 beds was the most promising method to explore (NAS/NRC, 1957). NAS reaffirmed that recommendation in 1961 (Carter, 1987; 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; Claiborne and Gera, 1974). From 1961 through the early 1970s, Oak Ridge National Laboratory (ORNL) conducted radioactive-waste disposal experiments, most notably Project Salt Vault in an abandoned salt mine near Lyons, Kansas, from 1963 to 1967 (Bradshaw and McClain, eds., 1971) (Fig. 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 (NAS/NRC, 1970).

Fig. 1. Early history of nuclear waste disposal related to the WIPP (adapted from Figure 2-1, Rechard, 1999, which provides extensive references).

2.1.3 Selection of Repository Site

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 16 miles from Denver, Colorado, the fire and subsequent cleanup attracted public attention. The press reported that the waste from the cleanup was to be sent to Idaho. 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. Public anger over the events created high visibility for the issue of waste disposal from nuclear weapons production. The State of Idaho felt it was becoming the nation's waste disposal site for defense-related waste by default. In addition, the Radioactive Waste Management Complex located there was not ideal for long-term disposal of nuclear waste because it was near the Snake River and its associated aquifer; hence, the AEC quickly moved to find a more suitable site. The AEC assured Senator Church from Idaho 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 minerals 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). 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 (Metlay, 1978); however, anti-nuclear groups and the EPA, through comments on the EIS for the facility, claimed the facility as *de facto* permanent disposal (MRSRC, 1989). The criticism prompted the AEC to search again for a more suitable disposal site (Jones et al., 1973, Bachman et al., 1973; Rechard, 1999). Soon after, ORNL and the USGS recommended to the AEC the extensive salt beds of southeastern New Mexico (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 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, e.g., grazing. 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, be studied further. Because the 1969 fire at the Rocky Flats Plant increased concern about this waste, the AEC quickly defined TRU waste using a lower bound definition: any contaminated material with an activity density of 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 International Atomic Energy Agency (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 (Perge, 1982). 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). Publication of the EIS was followed by a period for public comment. NEPA was the first environmental statute applied to the WIPP. Eventually, three EISs were prepared for the WIPP: one in 1979-80 during deliberations on whether to proceed with the construction phase (DOE, 1979; 1980), a supplement in 1989-90 to decide whether to proceed with

a pilot phase (DOE, 1989, 1990c), and a supplement in 1996-97 to decide whether to proceed with the disposal phase (DOE, 1996b; 1997) (Fig. 2). 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. 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.

Fig. 2. Development of performance goals for the WIPP (adapted from Figure 2-2, Rechard, 1999, which provides extensive references)

2.3 Establishment of Regulatory Risk Goals

2.3.1 Historical Events

In 1976, President Ford requested that the EPA accelerate development of applicable standards for proposed waste repositories. (Ford's request is noted in Volume 50 of the *Federal Register*, beginning on page 38067 [50 FR 38067] and the standards were first mentioned in 1976 [41 FR 53363].) The EPA conducted several public meetings to develop societal consensus on regulatory criteria (43 FR 2223). In 1978, the EPA proposed generic criteria on all radioactive waste (43 FR 53262), but after receiving an unfavorable response to these criteria, it withdrew the proposed regulations in March 1981 (EPA, 1985) 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 (47 FR 58196), 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 (50 FR 38066).

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 (as noted in Volume 824 of the *Federal Reporter*, second series, beginning on page 1258 [824 F2d 1258]), but the EPA repromulgated the Standard in 1993 without changes to the Containment Requirements (58 FR 66398). 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. In 1996, the EPA promulgated 40 CFR 194 (61 FR 5224), a regulation to implement its 40 CFR 191 standard specifically for the WIPP, and 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 a PA of the WIPP between 1989 and 1992 (Helton, 1993a; 1994; Helton et al., 1997c).

Besides complying with 40 CFR 191, waste intended for the WIPP also came under a legal ruling in 1984 (586 F. Supp. 1163) and subsequent changes in the EPA's and DOE's definitions of mixed waste in 1986 (51 FR 24504) and 1987 (52 FR 15937), causing as much as 60% of the waste destined for the WIPP to be designated as chemically hazardous (DOE, 1996b). Thus, the WIPP had to comply with a set of regulations for hazardous waste (40 CFR 260-270 and analogous New Mexico regulations) promulgated in response to the Resource, Conservation, and Recovery Act (RCRA) (Public Law 98-616; Public Law 94-580). In 1996, Congress eliminated the need for the WIPP to seek a no-migration variance from RCRA requirements (Public Law 104-201), which required calculations similar to those already provided for compliance with 40 CFR 191. However, other RCRA requirements implemented in 40 CFR 260-270 still applied, such as specifying the types and amounts of waste in state permits. Further details of how the regulations were applied to WIPP are provided in this special issue (Howard et al., this issue).

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, 1990c) and 1996 (DOE, 1996b; 1997), 40 CFR 191 required evaluating the cumulative release of radioisotopes $C(\mathbf{x})$, to the accessible environment. Furthermore, the definition of PA 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 $C(\bullet)$ is a function of parameters $\mathbf{x} = [x_1, x_2, \dots, x_{nP}]$, where nP is the total number of model parameters, and the parameters are uncertain (i.e., are described by a probability model by means of a distribution with a range of values), then the consequences $C(\mathbf{x})$ will also have a range and distribution. Uncertainty analysis involves determining the shape of the distribution for $C(\mathbf{x})$ (Fig. 3). Thus, the risk measure was not the expected value of the results, as in simple insurance annuity 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, PA became a stochastic simulation of possible long-term behaviors of a real system by means of a computer-implemented mathematical model. In this respect, PA remained similar to probabilistic risk assessments (PRAs) for nuclear reactors (Helton, 1994; 1997).

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 and (b) the chance that these limits could be exceeded, i.e., have less than 1 chance in 10 of exceeding L_i and less than 1 chance in 1000 of exceeding $10L_i$ (Fig. 3). For a

mix of radioisotopes, the release $C(\mathbf{x})$ for each radioisotope (i) is normalized with respect to its radioisotope limit (L_i), and then summed, that is,

$$C(\mathbf{x}) = \frac{1}{w_f} \sum_{i=1}^{nR} \frac{F_i(\mathbf{x})}{L_i} < 1 \text{ (or 10)}$$

where $C(\mathbf{x})$ is the total normalized release ("EPA sum"); w_f is a waste unit factor, various factors are defined (e.g., for transuranic waste $w_f = \sum A/10^6$ Ci where A is activity (Ci) of alpha-emitting transuranic radioisotopes in the repository with half-lives greater than 20 yr); L_i is the release limit specified by the EPA for radioisotope i ; $F_i(\mathbf{x})$ is the cumulative release of radioisotope i to accessible boundary integrated over a 10,000-yr period $\int f_i(\mathbf{x})$, during which system performance (such as actinide radioisotope transport in groundwater) had to be modeled, and nR is the number of radioisotopes contributing to release to the accessible boundary.

The Containment Requirements specified the radioisotope limits (L_i); in other words, as customary, the regulatory agency performed the dose-response assessment.² Because the Containment Requirements used cumulative releases of radioisotopes (F_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 allowable EPA limits (L_i) (EPA, 1985).

Fig. 3. Uncertainty in model parameters, $\mathbf{x}$ (expressed as a probability density function [PDF], or cumulative distribution function [CDF], results in uncertainty in the results $C(\mathbf{x})$. This uncertainty (expressed as a complementary distribution function, CCDF or 1 - CDF) is compared to the probabilistic limits in 40 CFR 191.

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 space of parameters, $\mathbf{x}$, and the relationship of those parameters through a model. The model may be only conceptual, but for the WIPP PA, it was a mathematical model. In a mechanical system, the various parts of the system are *defined*. With a radioactive waste disposal system, however, the existing geologic parts must be *characterized*. Characterization of the geologic subsystems of the WIPP was undertaken as soon as the site was selected to (1) satisfy needs for the Draft EIS in 1979 (DOE, 1979) and Supplemental EISs in 1989 (DOE, 1990c), (2) satisfy negotiated agreements with the State of New Mexico (State of NM, 1981), and (3) develop a general understanding of natural phenomena significant to nuclear waste disposal.

² 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 PA is not entirely probabilistic.

3.1 Selection of Site

With the tacit approval of New Mexico's governor Bruce King, ORNL examined 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 area was semi-arid 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. 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].) 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 Arab 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). 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 and, at a depth of 826 m, artesian brine and H₂S gas, causing dismay on the part of ERDA staff who, as former AEC officials, had earlier struggled with difficulties at the Lyons site.

But the large Delaware Basin had other areas that met the various selection criteria. In late 1975, after examining the stratigraphy shown in the ERDA wells, along with confidential data from numerous private wells owned by oil companies and confidential geophysical seismic data, 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 from the first location, away from the ~10-km band around the Capitan Reef where deformation of the salt beds occurred, and with more predictable horizontal stratigraphy (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) (Fig. 4).

Fig. 4. Physical setting of the WIPP in Delaware Basin (after Rechard, 1995, Figure 2.1-1).

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. During this time Sandia collected data on the ecological and archaeological features of the region and its socioeconomic facets (Weart, 1979). Sandia's description of the geology and hydrology (Powers et al., eds., 1978) drew heavily upon information that had been prepared during the search for a regional site location (Bachman et al., 1973; Jones et al., 1973; Bachman, 1973).

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 (Mercer and Orr, 1979).

USGS geohydrologists had suggested that the Magenta and Culebra Members of the Rustler Formation (hereafter shortened to Magenta, Culebra, and 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 nor did it 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 (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 recrystallization or brine flow through the formation since its deposition about 255 million years ago (Powers et al., eds., 1978; Weart, 1979) (Fig. 5). Further information on the geologic setting is discussed in this special issue (Swift and Corbet, this issue).

Fig. 5. Stratigraphy above and below the WIPP repository (after Rechard, 1995, Figure 2.1-2).

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, 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. The test, using

compressed air flow measurements, indicated a formation permeability over a 30-m test section between 5×10^{-19} and $2 \times 10^{-17} \text{ m}^2$ (permeability at an elevation of 548 m and at an elevation of 427 m [depth of 690 m], respectively (Tyler et al., 1988).³ 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 some permeability tests around excavations 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 an underground 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 $2 \times 10^{-17} \text{ m}^2$ (Tyler et al., 1988) (Fig. 6).

Fig. 6. Estimated permeability of Salado Formation near repository horizon continued to decrease over the years (Rechard, 1999).

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 (Lappin et al., eds., 1989). By December, the national press was reporting on the issue of brine flow into the repository. 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 brine entering a disposal room over the first 100 yr (Nowak et al., 1988). Given the resolution of measurements, the BRWM concluded that not enough brine would seep into the rooms to form a slurry of radioactive waste before the

³ 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 $5 \times 10^{-20} \text{ m}^2$.

rooms had closed through salt creep. 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

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 (see Fig. 2).

As part of the negotiated settlement with the State of New Mexico, the DOE deepened WIPP-12 into the Castile in November 1981 (see Figs. 3 and 4). The WIPP Project encountered a brine reservoir at a 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, which is 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 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 Castile's upper anhydrite layer where brine had been encountered earlier (Lappin, 1988; WIPP PA Division, 1991/1992) (Fig. 7).

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 an 860-km² area around the WIPP site (Lappin et al., eds., 1989; Lappin, 1988).

Fig. 7. Site characterization studies at the WIPP (adapted from Figure 3-4, Rechard, 1999, which provides extensive references).

3.4 Site Characterization Studies for CCA

As summarized below, additional site characterization studies were conducted in the 1990s for PA calculations needed to support the CCA (Compliance Certification Application) to be submitted to the EPA. These studies are detailed elsewhere in this special issue (Larson et al., this issue).

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 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 et al., 1997).

In 1996, hydraulic tests and fluid 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 (Mercer and Orr, 1979). Therefore, the hydrologic effective thickness of Culebra in the PA for the CCA was taken as ~4 m (7 m, less the 3 m for the upper portion of the Culebra) (Meigs et al., 1997).

3.4.2 Sorption Studies in Culebra

Sandia conducted several laboratory studies of sorption in the 1980s (Dosch, 1979, 1981; Lynch et al., 1981). 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_d s) recommended for use in the final PA of the WIPP. While experimental data were being obtained, Sandia convened a panel of staff members to estimate ranges and probability distributions of K_d s in support of two preliminary PAs (Trauth et al., 1992). In addition, retardation of radioisotopes in the Culebra was more thoroughly studied in the laboratory. Early results from a batch experimental program using crushed dolomite, described in this special issue, were used in the CCA (Larson et al., this issue; Brush and Storz, 1996; Papenguth and Behl, 1996).

4.0 Waste Properties

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. During the 1970s, the lack of a proven waste disposal scheme 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 nuclear waste was demonstrated (Carter, 1987). Therefore, 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 waste, 640 m below the surface, and the other for remote-handled 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. However, the then powerful House Armed Services Committee strongly opposed commercial waste disposal at the WIPP. 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" (Fig. 8).

Fig. 8. Project goals and characterization of waste properties (adapted from Figure 4-1, Rechard, 1999, which provides extensive references).

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 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. 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 the 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 ratified previous agreements between the State of New Mexico and the DOE, stating that the total combined volumes of CH-TRU and RH-TRU waste were not to exceed 6.2 million cubic 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; 1980).

4.2.2 Evolution 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). The EPA changes in the definition required that the DOE reclassify only a small portion of its TRU waste because less than 20% was between 10 and 100 nCi/g. The activity of most waste was around 10^4 nCi/g (Smith, 1982; WIPP PA Division, 1991, Vol. 3).

4.2.3 TRU Waste Inventory

For preliminary PAs between 1989 and 1992, waste activities and mass were estimated from the Integrated Data Base, produced annually by ORNL for the DOE (see for example, IDB, 1990), and from requests for information by Westinghouse directly to the waste generator sites (see for example, WIPP PA Division, 1991, Vol. 3). For the CCA, the DOE assembled a baseline inventory, referred to as the Transuranic Waste Baseline Inventory Report (TWBIR) (DOE, 1995). 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 these data as described elsewhere in this special issue (Sanchez et al., this issue).

4.3 Waste Characterization Studies

Two aspects of the TRU waste destined for the WIPP were important to characterize for the compliance calculations: the estimated inventory of radioisotopes in the waste,⁴ and the mobility of these radioisotopes within the

⁴ 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). Hazardous waste constituents are no longer important because Congress exempted the DOE from demonstrating "no migration" of hazardous components above

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.

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, the WIPP Project had sent guidance to the generator/storage sites in 1976 that the waste be incinerated to remove combustible and organic material. However 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).

4.3.2 Characterization for PA 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 Lappin et al. (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 the new 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 et al., 1991, 1993). The experiments were completed by September 1995, and the observed gas generation rates were used in the PA 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

health concentration standards. However, the types of hazardous constituents are important for disposal permits with the State of New Mexico

that the WIPP would remain open, even if it could not comply with all EPA regulations during the transition from the pilot 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 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 the scientific need—not just an operational need—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. 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, the waste properties that had been found to be most important in the preliminary PAs were solubility of actinides, primarily plutonium and americium. For the 1991 and 1992 preliminary PAs, an external expert panel had been convened in early 1991 to evaluate solubility data for the radioisotopes (Trauth et al., 1991; 1992). The panel examined existing data and developed probability distributions of solubilities to express the uncertainty thought to pertain at the WIPP repository. Also, related experimental programs were initiated. As described elsewhere in this issue (Ramsey et al., this issue), solubility experiments were conducted with 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, and the solubility of actinides in high ionic strength brines was modeled (see Section 7.3.3). 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 (Fig. 9).

Fig. 9. Solubility of radioisotopes used in calculations between 1989 and 1996 for (a) plutonium and (b) uranium (Rechard, 1999).

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. In the 1977 conceptual design, the basic room layout for CH-TRU disposal was developed (Sandia, 1977), which remained essentially unchanged in the final design although some room dimensions were modified in 1980 in response to numerical modeling results. The modifications to the room dimensions were validated in 1982 by observing deformations in several test rooms (Tyler et al., 1988). 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 the conceptual design of the WIPP surface and underground facilities (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 conceptual design was completed in 1977 (Sandia, 1977). The disposal area for the CH-TRU waste was set at 373 m elevation and consisted of eight closely spaced panels with eight rooms in each panel. Each room was 5 m high, 14 m wide, and 137 m long (Fig. 10a). 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 near the base of the Salado. The repository had five shafts.

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. Also, the volume of high-level waste was not known. However, in December 1979, Congressional legislation limited the WIPP to the disposal of defense TRU waste only. The second level was abandoned and the repository was reconfigured for the Final EIS as a single-level facility with four shafts (DOE, 1980).

Fig. 10. Designs for WIPP repository including (a) 1977 conceptual design (Sandia, 1977) and (b) final construction plans for WIPP repository (Rechard, 1995).

5.1.2 Detailed Design

In 1978, the DOE contracted with Bechtel National, Inc., to be the architect/engineer for the WIPP Project, and Westinghouse Electric Corporation, for technical support. 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 disposal area provided enough space for the disposal of the projected 176,000 m³ 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 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 a later section) 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 (Fig. 10b). 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 all types of nuclear waste.)

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 to further characterize stratigraphy near the repository and validate room dimensions in the preliminary design. As part of the SPDV program, Fenix & Scisson, Inc., Sandia's contractor for the conceptual design and DOE's SPDV contractor, drilled two shafts starting 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 was completed by the end of November. In the second half of the year, four full-sized disposal rooms were excavated, 4 m high, 10 m wide, and 91 m long, separated by 31-m-wide pillars. 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 and instrumented to confirm the stratigraphy and evaluate deformations around the repository.

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. Reaming 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 preparation for operations, the Management and Operations (M&O) contract was awarded to Westinghouse, which conducted the remaining construction activities after October 1986. In 1988, a fourth shaft was added for increased air circulation (Fig. 11). A pilot hole was first drilled from the surface and then reamed 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.

Fig. 11. Events associated with design, construction, and modeling of the WIPP repository (adapted from Figure 5-2, Rechard, 1999, which provides references).

5.3 Experiments and Model Development of Repository Behavior

Early in the history of the WIPP Project, the DOE viewed the WIPP as a research and development facility that would 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.

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 and at the Avery Island salt dome in Louisiana. Laboratory creep tests were started on larger specimens from the underground workings of the WIPP in 1982; in situ salt creep experiments 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; heat was turned on in these rooms in 1985 to measure accelerated salt creep. 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 at the WIPP.

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 and constitutive laws (Tyler et al., 1988), and used the results in 1980 (Krieg et al., 1981) to examine the reasonableness of various proposed modifications to the room dimensions in the 1977 conceptual design, which had been based on experience with the area's potash mines. Predictions from other numerical codes 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 and various other experimental rooms in 1982-83 were nevertheless about three times greater than predicted values (Morgan et al., 1985). Thus between 1985 and 1989 an alternate conceptual model and mathematical expression were incorporated into codes and tested. Thereafter, agreement of predictions with in situ measurements was excellent (Munson et al., 1989).

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. 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 the potential for any underground fire (DOE, 1980). Although backfill in the disposal area was considered part of the baseline design for the repository (Sandia, 1977; DOE, 1980), as reported in the supporting documents for the Supplemental EIS (Lappin et al., 1989) and the four preliminary PAs 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 CO₂ so that brine present in the repository would not become acidic, thereby increasing the solubility of the actinide radioisotopes. Modeling of the MgO chemical backfill is discussed further in this special issue (Hansen et al., this issue).

5.4.2 Shaft Backfill and Sealing Methods

Because of the difficulty 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 Fig. 3). 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 for the shafts through the Salado (usually referred to as "seals" in the WIPP Project), was expected to limit the creation of a preferred pathway for contaminant 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. As described in this special issue (Hansen et al., this issue), compacted salt (through tamping) is planned 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 envisioned large plugs 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., 1989, Figure 4-10). However 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 (Nowak et al., 1990). Details of the various options were developed in 1993 (Van Sambeek et al., 1993). 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, as described in this special issue (Hansen et al., this issue).

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. 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., 1989; Nowak et al., 1990). During the early 1990s, the value of panel seals was questioned (Rechard et al., 1990) because of their limited ability to contain gas given the presence of anhydrite layers above and below the drifts. However, because they afforded some protection, the EPA stipulated that panel seals be used at the WIPP in its ruling on the compliance of the WIPP in 1998 (62 FR 58792; 63 FR 27354).

6.0 Hazard Identification and Scenario Development

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 PAs (Hunter, 1989), and in 1995-96 for the final PA (Galson and Swift, 1995). The latter update is described further in this special issue (Galson et al., this issue). 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 PA 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 PRAs. 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).

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, $C(\bullet)$, "future" and "scenario" can be precisely defined in terms of the parameter space, $\mathbf{x}$, as follows. The parameter space $\mathbf{x}$ can be arbitrarily divided into two subsets: a subset defining conditions of scenarios, $\mathbf{x}_{st}$, and a subset of remaining model parameters, $\mathbf{x}_{sb}$, i.e., $\mathbf{x} = [\mathbf{x}_{st}, \mathbf{x}_{sb}]$. A future is one point in parameter space for scenarios, $\mathbf{x}_{st}$, and a scenario is a grouping of similar futures or subsets of $\mathbf{x}_{st}$, i.e., $S_j \subset \mathbf{x}_{st}$ where S_j is a scenario of what can occur to produce an unwanted outcome. For example, a scenario might be described as $S_j = \{x_{st,1} < a \text{ and } x_{st,2} > b\}$, where $x_{st,1}$ and $x_{st,2}$ defined conditions for human intrusion and location of a brine reservoir, respectively. Characterization of scenarios and their uncertainty, $\mathbf{x}_{st}$, came to be known as stochastic uncertainty within the WIPP Project to distinguish it from the more general epistemic (or subjective) uncertainty associated with model parameters, $\mathbf{x}_{sb}$. Both are described further in this special issue (Helton et al., this issue-a; Tierney et al., this issue).

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 mentioned only. 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 thought to be limited to the drilling crew. Failure of borehole plugs after a well 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 as 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 is 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.

6.3.1 Features, Events, and Processes Considered

For the EIS, Sandia considered events similar to ORNL's list but in a more formal manner. Sabotage, nuclear warfare, erosion, and meteorite impact were quickly dismissed as single events, although retained as initiating events for other events that would affect release. 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 (Fig. 12a). The WIPP futures 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 were analyzed early in 1978 for inclusion in the Draft EIS (DOE, 1979).

Fig. 12. Technique to construct scenarios from various features, events, and processes changed between WIPP EIS in 1979 and PA calculations after 1989.

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 pioneer work on a PRA 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 at ORNL. Later the WIPP Project was to benefit from the NRC's efforts.

6.4.1 Defining Universe (FEPs List)

In 1976, the NRC funded 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 this Sandia group pursue a scenario development process similar to the *Reactor Safety Study* (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 a simple or useful task. 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. The procedure included forming scenarios by combining specific remaining FEPs by means of a logic diagram in which each pathway led to a scenario class. Combinations of all events were used to construct scenarios, and the time (or order) of occurrence was not considered (Fig. 11). 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 the FEPs were included during modeling.

6.4.3 Including the Human Intrusion Event

Because injection boreholes and the salt dissolution around them 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. During the late 1970s, each site with previously buried TRU waste and low level radioactive waste, such as INEEL 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. Usually human intrusion of some type provided a significant portion of the calculated consequences but attaching a probability to the event was difficult (Smith, 1982). Also, no one knew how the human intrusion event related to the robustness of the disposal system except that more varied types of intrusion would likely occur for disposal sites at shallow depths or for sites near unique mineral resources.

In the draft of 40 CFR 191 (47 FR 58196) published in 1982, the EPA reduced the scope of human intrusion to an inadvertent activity, specifically from exploratory drilling primarily for oil and gas deposits. 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 (61 FR 58499), 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 capabilities similar to ours rather than to a society with less or greater technical prowess. In 1995, 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

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 reevaluated in 1989 (Hunter, 1989) in conjunction with a demonstration of the PA 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), 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) (Fig. 13).

Fig. 13. Engineering analysis of the illustrated behavior of the disposal system (adapted from Fig. 6-2, Rechard, 1999, which provides references).

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 were identified for any of the PAs conducted for the WIPP (Hunter, 1989; Galson et al., this issue).

Natural processes retained for the preliminary PAs included climate variability (analyzed separately in 1989 and included in PAs 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 (Fig. 12b). In the 1989 PA, two potential pathways for migration of contaminants were considered. In the first path, the pressure gradient between the waste disposal panels and the Culebra might cause brine and radioisotopes to migrate either through drifts or anhydrite interbeds to the base of the shafts and then upward to the Culebra. In the second path, brine and radioisotopes migrated laterally from the undisturbed repository through thin anhydrite interbeds toward the subsurface boundary of the accessible environment within the Salado. In all PAs 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 PAs, the only disruptive event for scenario construction was inadvertent human intrusion. As described further in Section 8, 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, mathematically described as a Poisson process (Helton, 1993c).

In the human intrusion scenario, it was assumed that if the disposal area of the repository were 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. 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 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: 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. 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. For improved computational resolution, the three scenarios were divided further into computational scenarios on the basis of time of intrusion and radioactivity of the intersected wastes beginning in 1991 (Helton and Iuzzolino, 1993). In addition, E2-type intrusions were 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 that fully documented the reasons for omitting or retaining specific FEPs, as described further in this special issue (Galson et al., this issue) and elsewhere (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). 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. The latter event was based on experience in the Delaware Basin from drilling new oil wells in areas where water flooding had previously occurred to enhance oil recovery in deep reservoirs (Stoelzel and O'Brien, 1996). 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 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. These analyses are discussed further in this special issue (Stoelzel et al., this issue).

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;

DOE, 1980). Thereafter, the whole system was analyzed for the Supplemental EIS in 1989 (Lappin et al., 1989; DOE, 1990c), during the four preliminary PAs 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), as described in several articles in this special issue. Currently, component and system analysis continues in preparation for recertification, which is required every five years during operation of the WIPP.

As previously mentioned, the consequence model is typically divided into two major model categories: (1) a model for the human receptor for a toxicity assessment and (2) a model, $f(\bullet)$ of the environment around the receptor, to conduct an exposure assessment that includes uncertainties. For the original EIS, the EPA performed the toxicity assessment, based on a simple linear bounding response curve, when it established its release criteria in 40 CFR 191, as promulgated in 1985 and 1993 (50 FR 38066; 58 FR 66398).

The various physical scales in the WIPP geologic disposal system made one detailed exposure pathway model, $f(\bullet)$, impractical. Instead $f(\bullet)$ was further divided into many complex component models $f_{\alpha}(\bullet)$; results from one model, $f_{\alpha}(\bullet)$, became variables used by the next model, $f_{\beta}(\bullet)$.⁵ All models $f_{\alpha}(\bullet)$ were dependent on the scenario S_j (stochastic model parameters, $\mathbf{x}_{st}$) under consideration and the subset of epistemic model parameters $\mathbf{x}_{sb}$ related to that scenario. The following sections discuss the component models, $f_{\alpha}(\bullet)$, used for the WIPP.

7.1 Analysis for Site Selection and EIS

7.1.1 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 $f_i(\bullet)$: 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/cm² 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

⁵ For example, a cuttings and cavings model, f_C , was used and called CUTTINGS_S; the two-phase flow model, f_B , was called BRAGFLO in the 1996 PA; both models are discussed further in this special issue.

dilution from the Culebra, as estimated from data from the Gnome site (Gard, 1968), was $1.1 \times 10^5 \text{ m}^3/\text{yr}$ based on a 2.4-km-wide repository, 9-m-thick Culebra, and a Culebra porosity of 0.10.

7.1.2 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 $2.5 \times 10^5 \text{ yr}$, roughly ten half-lives of ^{239}Pu . The consequence analysis used four component models $f_{\alpha}(\bullet)$: the direct release model, the source term model, the Culebra flow model, and the Culebra transport model.

Direct Release Modeling. In the original EIS, consequences measured as dose were evaluated from direct release to either a geologist on the drill rig examining the core or a single-farm family 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.

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, 1980). 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 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 flow model constructed for Sandia by INTERA, Inc. encompassed most of the area shown in Fig. 4. 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, 1980, 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.

7.1.3 NRC Analysis

For the NRC, Sandia initially described a consequence analysis method in 1978 (Campbell et al., 1978). In 1981 that process was more thoroughly developed and then 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, $f(\bullet)$, comprised a series of loosely connected 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), which was the forerunner to NEFTRAN (Longsine et al., 1987). 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).

7.1.4 Supplemental EIS Analysis

The Draft and Final Supplemental EISs for the WIPP, completed ten years later in 1989 and 1990 (Lappin et al., 1989; DOE, 1989; 1990), 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 $f_{\alpha}(\bullet)$ of the exposure pathway model, $f(\bullet)$, 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.

Source Term/Repository Modeling. A distinct source term/repository model was developed for the Supplemental EIS, using an updated finite-difference code, 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 using best and degraded values for parameters. The inventory was limited to either one panel or one room 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} or 10^{-1} mM (see Fig. 8).

The use of a solubility limit to determine concentration required an estimate of the amount of brine available to dissolve the radioisotopes. A constant flow from the Salado of either $1.3 \text{ m}^3/\text{yr}$ or $0.1 \text{ m}^3/\text{yr}$ 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.

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 since 1980 were included in this model. The boundaries of the model were slightly larger than the WIPP site (see Fig. 4). 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 water levels at wells H-11 and DOE-1 required placing a highly transmissive zone south of these wells.

Modeling of Culebra Fluid Flow in 1989. The 1987 study was followed by another modeling study in 1989 (LaVenue et al., 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 1989 study were much larger than those of the 1987 study, extending approximately 30 km north and south and 20 km east and west, with the WIPP site at the center. 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. 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.

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., 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.

7.2 Analysis Logistics for PA

The practical aspects of performing the exposure pathway calculations is daunting for a system comprised of several model components, such as the WIPP disposal system. Important practical problems in the WIPP Project were how to gain proficiency in PA calculations and 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.

7.2.1 Iteration of Calculations

In 1989, the WIPP PA analysts adopted the idea of conducting sequential PAs, that is, conducting an initial PA with simple or incomplete models and preliminary data, followed by other PAs 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 (Morgan et al., 1990) or the 1975 *Reactor Safety Study* (Rasmussen, 1975) 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.

In addition, multiple PA iterations achieved other benefits. First, a long, multiyear project could be divided into annual tasks, with more easily agreed-upon goals and schedules. Also, 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 PAs. In addition, the WIPP Panel of the NAS and the EEG, though not set up in 1978 exclusively to review PA-like calculations (or evaluate compliance), received quarterly presentations and commented on PA calculations.

Also, 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 to address this specific question in 1994. Finally, in combination with sensitivity analysis (Section 8), iterative PAs allowed project managers, PA analysts, and experimentalists to decide how best to allocate resources for supplementary data collection and whether models should be elaborated upon or simplified in later iterations. Consequently, Sandia conducted four preliminary PAs from 1989 through 1992, with each PA building upon the preceding PAs (Marietta et al., 1989; Rechard et al., 1990; Bertram-Howery et al., 1990; WIPP PA Division, 1991/1992; WIPP PA Department, 1992/1993).

7.2.2 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, 1980), 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, analysts had also chosen to emphasize the detailed modeling style. An important reason for using a detailed modeling style was the general acceptance in the United States of its use in PRAs (Rasmussen, 1975; Breeding et al., 1992). Also comments received from the EPA (Bertram-Howery et al., 1990) and the WIPP Panel of the NAS on the 1989 PA encouraged Sandia to move from the simplified NEFTRAN (Longsine et al., 1987) models to a more detailed modeling. 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.

7.2.3 CAMCON Development for PA

The major role of modeling in a PA 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.

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) 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 PA. One approach was to build one code with numerous submodels (e.g., SYVAC), 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]). Serious work on developing a software system for WIPP calculations 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. Simultaneously, a more carefully constructed version was developed for the first complete PA in 1990.

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 codes that were interchangeable but not identical could be used for a model component, i.e., $f_{\alpha}^1, f_{\alpha}^2 \dots f_{\alpha}^{nM}$ where nM designates the number of codes that perform a similar function, and α designates a specific model component such as the Culebra transport model mentioned earlier. Section 7.3 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, $f(\bullet)$. 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 database, 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): (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 simple neutral file format, 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 some details were changed. By the time of the 1996 PA, the costs associated with the stringent QA procedures required the selection of one code for each major component of the consequence model. 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 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. Further details of the final system are discussed in this special issue (Froehlich et al., this issue-b).

7.3 Consequence Models in PAs

For the WIPP PAs, the exposure pathway model grew from four major components to seven: the direct cuttings release model, the direct brine release model, the repository fluid flow model, the source term model, the Culebra transmissivity model, the Culebra fluid flow model, and the Culebra radioisotope transport model. Before 1996, any uncertainty associated with a model's mathematical form or its application was considered only through general reviews of the preliminary PAs. However, for the 1996 PA, a Conceptual Model Peer Review Group studied the issue of correct model form.

7.3.1 Direct Cutting and Brine Release 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. The contribution of these direct releases to the total releases in the 1996 PA is discussed further in this special issue (Berglund et al., this issue).

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 spillings was included in the 1996 PA as CUTTINGS_S, as discussed in this special issue (Knowles et al., this issue).

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 (Fig. 14). Direct brine releases are discussed further in this special issue (Stoelzel et al., this issue).

Fig. 14. Changes in components of the exposure pathway ("consequence") model for immediate releases after exploratory drilling for evaluating compliance of the WIPP.

7.3.2 Repository Fluid Flow Modeling

In the PAs, the intrusion borehole and repository were represented as a separate model component, $f_t(\bullet)$, 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 (Fig. 15). 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. Studies included gas flow from the disposal area to the shaft using the two-phase code BOAST (Fanchi et al., 1987), Salado brine flow through a panel to a borehole using SUTRA, effects of anhydrite layers on Salado brine flow through a panel, and flow from a brine reservoir through a disposal room (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, 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). As described in more detail elsewhere in this issue (Vaughn et al., this issue), 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.

Fig. 15. Changes in components of the exposure pathway model for long-term releases via leaking borehole and Culebra brine aquifer (Rechard, 1999).

7.3.3 Source Term Modeling

The 1989 PA used a source-term submodel in the transport code NEFTRAN, as in the Supplemental EIS (Lappin et al., 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. 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 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 (WIPP PA Department, 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) (Fig. 15). 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. The source term and radionuclide transport in the vicinity of the repository for the 1996 PA are discussed further in this special issue (Stockman et al., this issue).

7.3.4 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; Rechar, 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 the 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) (Rechar, 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) (Rechar, 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.

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 the WIPP 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 (Ramsey et al., this issue).

7.3.5 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 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 PA. 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 previously mentioned. 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.

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 earlier 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 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, 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 (see Section 3.4). As described in this issue, in the article on radionuclide and colloidal model transport in the Culebra (Ramsey et al., this issue), 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 used in a PA (Rechard, 1995; Tierney and Rechard, 1997) a model, $f(\bullet)$, which was discussed in the previous section; a space of model parameters, $\mathbf{x}$, which was conceptually developed during system characterization and hazard identification; and a joint probability density function of the model parameters, $D(\mathbf{x})$. However, given that the parameter space, $\mathbf{x}$, was divided into two

disjoint parts (those parameters associated with scenarios and the remaining parameters, i.e., $\mathbf{x} = [\mathbf{x}_s, \mathbf{x}_{sb}]$), two types of probability evaluations were necessary for the WIPP PAs. Using scenarios for stochastic uncertainty, $\mathbf{x}_s$, was first attempted for the initial WIPP EIS in 1979 (Bingham and Barr, 1979, 1980). Probabilistic descriptions for the model parameters, or subjective uncertainty, $\mathbf{x}_{sb}$, 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 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 models for evaluating scenario probabilities for the EIS (see Section 7) were chiefly subjective judgments. The remaining scenarios were grouped into three scenario classes, which were evaluated by means of consequence models (see Section 7). Typically, 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. The probability of a scenario occurring was evaluated to screen out those with low probabilities.

8.1.2 Parameter Selection

Like the *Reactor Safety Study* (Rasmussen, 1975) conducted four years earlier, there was no attempt to evaluate how the epistemic (subjective) uncertainty in the model parameters ($\mathbf{x}_{sb}$) contributed to the uncertainty in the results in the original EIS. Only stochastic uncertainty ($\mathbf{x}_s$) was evaluated 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.

8.2 Scenario Probabilities for Performance Assessments

As described earlier, PA analysts continued to define 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, $P(S_j)$, had to be calculated by some method. In the 1989 PA, the probabilities of various scenarios were based on subjective judgment with no ranges of uncertainty as for 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 (i.e., $P\{n>0\}=1-e^{-\lambda}$) (Helton, 1993c). Usually the Poisson process was assumed to have a constant expected rate of intrusion, λ , over the 10,000-year regulatory period. The rate was constant throughout any one simulation in the preliminary PAs between 1989 and 1992; however, a different value between 0 and the maximum value, $\lambda_{\max}$, of 30 boreholes/ km² per 10,000 yr was

selected for each of the many simulations. Varying λ accounted for uncertainty in the 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 the groundwater pathway as in 1991 or only one at 1000 yr as in 1992) was evaluated directly with the code CCDFPERM (Fig. 14) (Helton and Iuzzolino, 1993).

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, $\lambda(t)$ (nonstationary Poisson process) (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 (stationary Poisson process) (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 function was as follows: (1) $\lambda = 0$ while active institution controls, such as land control, were present, $t < 100$ yr; (2) $\lambda = 0.01 \cdot \lambda_{\max}$ while passive institutional controls, such as markers about the WIPP site, were present, $100 \text{ yr} < t < 700 \text{ yr}$; and (3) $\lambda = \lambda_{\max}$ thereafter where the maximum rate of intrusion, $\lambda_{\max}$, was increased to 48.5 boreholes/km² per 10,000 yr based on guidance in 40 CFR 194 (61 FR 5224), as described further in an article on stochastic uncertainty in this special issue (Helton et al., this issue-b).

8.3 Parameter Uncertainty in PAs

In 1985, when 40 CFR 191 requested the DOE applicant to "assemble all of the results of the PAs to determine compliance with §191.13 into a 'complementary cumulative distribution function [CCDF]'," an important goal for PA 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. Hence data bases and procedures for developing parameter distributions were important administrative controls.

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). The secondary data base contained distributions of parameters that had been derived from the primary databases specifically for various component models of the exposure pathway model, $f(\bullet)$. The computational data base, generated during each PA, comprised the calculated results. By 1990, the WIPP Project used the latter two data bases directly in the PA calculations; however, the computational database existed only as a collection of catalogued files rather than as a true 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 PA 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 secondary areas: Report Review and Expert Judgment Panels.

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 fundamental requirement of the Software QA procedures was the development of a Software Management System (the CAMCON Modeling System; see Section 7). For the 1996 PA supporting the CCA, these procedures were developed into a full suite of quality assurance procedures, as discussed further in this special issue (Froehlich et al., this issue-a).

8.3.3 Describing Parameter Uncertainty

To evaluate the epistemic or subjective uncertainty of PA results, a joint cumulative distribution function, $D(\mathbf{x}_{su})$, was required that characterized the uncertainty of model parameters, $\mathbf{x}_{su} = [\mathbf{x}_{su,1}, \mathbf{x}_{su,2}, \dots, \mathbf{x}_{su,n}, \dots, \mathbf{x}_{su,nU}, \dots, \mathbf{x}_{su,nP}]$ where nU is the number of uncertain parameters and nP is the total number of parameters. As is standard practice, $D(\mathbf{x})$ was approximated by the product of the cumulative distribution functions (CDFs) of the individual parameters, $D_1(x_1) \cdot D_2(x_2) \cdot \dots \cdot D_n(x_n) \dots \cdot D_{nU}(x_{nU})$, 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, $D_n(x_n)$, 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 Fig. 6). The avoidance of a conservative bias was an important shift from the philosophy pursued for the EIS (DOE, 1979; 1980) and Supplemental EIS (DOE, 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 $D(x_n)$ 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, as discussed further in this special issue (Tierney et al., this issue).

The use of a consistent set of parameter values was initiated for the 1989 PA (Rechard et al., 1990) 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., 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, such as 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). The model parameters in the database were described formally beginning in 1990 and thereafter (Rechard et al., 1990; WIPP PA Division, 1991/1992; WIPP PA Department, 1992/1993). The number of documented parameters increased from 191 parameters, excluding geometric and stratigraphic positioning of layers, to 1561 documented parameters by 1996.

An important practical problem for parameter uncertainty analysis was determining the number of uncertain parameters to propagate. As the computational capability was 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).

8.4 Sensitivity Analysis

Sensitivity analysis is the evaluation of aspects of the system that most influence the calculated or observed results. Specifically, a sensitivity analysis ranks in importance the uncertain parameters $\mathbf{x}_{su}$ (or model forms, e.g., $f_{\alpha}^m(\bullet)$) that determine $C(\mathbf{x}_{su})$ and its cumulative distribution function (CDF), i.e., $Pr\{R \leq r\} = \int \delta[r - C(\mathbf{x}_{su})] dD(\mathbf{x}_{su})$, where $\delta(\bullet)$ is the delta function (whose integral is zero when the argument is negative and one when the argument is positive), $D(\mathbf{x}_{su})$ is the joint probability function for the $\mathbf{x}_{su}$ model parameters, r is an arbitrary variable, and the integral is evaluated over the space of uncertain 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 (LANL, 1987). But the technique applies to any multidimensional integration such as that determining the distribution of $C(\mathbf{x}_{su})$, i.e.,

$$Pr\{R \leq r\} = \frac{1}{nK} \sum_{k=1}^{nK} I[r - C(\mathbf{x}_{su,k})], \text{ where } I(\bullet) \text{ is an indicator function equal to zero when the argument is negative}$$

and one when the argument is positive, $\mathbf{x}_{su,k}$ is a set of sampled parameters drawn from $D(\mathbf{x}_{su})$ and nK equals the number of Monte Carlo samples. However, using a large number of samples, as is necessary with the crude Monte Carlo method, is impractical when evaluation of the function $C(\mathbf{x})$ 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 PAs 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). It was also proposed for use in analyses for the geologic disposal of radioactive waste as early as 1978 (Iman et al., 1978). 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, as conjectured in the early 1980s (Iman and Conover, 1980; Iman, 1981) and more thoroughly demonstrated later in the decade (Iman and Helton, 1988; 1991), encouraged the WIPP Project to adopt the LHS technique to propagate parameter uncertainty and determine the distribution of $C(\mathbf{x})$ 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, $C(\mathbf{x})$, to changes in parameter values could be approximated and conveniently determined in several ways, including (1) examining scatterplots or (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). Details of the application of LHS in the 1996 PA are described further in the stochastic uncertainty article in this special issue (Helton et al., this issue-b).

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, time of intrusion into the repository, and 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 upper bound 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 exposure pathway model (Rechard et al., 1990).

In 1990, regression analysis techniques to determine sensitivity were used (Helton, 1993b). The same three parameters (solubility of radioisotopes, time of intrusion, and assumed diameter 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 PA 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 (Fig. 15). 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 ranking for the diameter of the intrusion borehole (WIPP PA Division, 1991/1992; Helton et al., 1993b). The sensitivity analysis in 1991 also included an evaluation of alternative conceptual models (Helton et al., 1995).

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). Thus, by 1992 it was evident that regulatory assumptions on the style of modeling, specifically with regard to assumptions about human intrusion, were dominating the results. 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 volatile organic compounds (VOCs, RCRA hazardous constituents) was still required at this time (Helton et al., 1993a). In the 1996 PA, releases occurred only from directly drilling into the repository. Parameters influencing these releases are described in two articles in this special issue (Helton et al., this issue-c, this issue-d).

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 PA 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, 1991, and 1992 some general guidance was provided (Helton et al., 1991; WIPP PA Division, 1991/1992).

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). Also, in its practical application, the analysis was not probabilistic because the time needed to run a sufficient simulation would have been excessive. Given these time constraints, only a deterministic simulation of each activity, using an ad hoc combination of mean and median parameter values, could be run.

9.0 Compliance Assessment and Summary

An important difference between risk assessments or other large-scale policy analyses, which are usually conducted to gain understanding of the behavior of a system, and the WIPP PA is that the WIPP PA was specifically designed to test *compliance* to a set of standards so that decisions can 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 by the WIPP Project over the years was initially the National Environmental Policy Act (NEPA) of 1969 and later the regulations in 40 CFR 191, Subpart B (58 FR 66398) and 40 CFR 194 (61 FR 5224).

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., 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., 1989). No calculations were conducted for the farm family in the 1996 EIS (DOE, 1996b).

9.1.2 Releases from Preliminary PAs

Preliminary results in the form of CCDFs from the PAs were compared with the EPA regulations between 1989-1992 (Fig. 15). Over the years, the predicted cumulative releases immediately after drilling into the repository were 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. However, 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 became practically nonexistent by 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 (Fig. 16); hence, the EPA requirement that the human intrusion event be included indirectly defines the point at which disposal system characterization is sufficient.

Fig. 16. 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 (Rechard, 1999).

9.1.3 PA 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 pathway exposure model, $f(\bullet)$, 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 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 AlphaTM 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 (Fig. 16).

Throughout the winter and spring of 1997 (during the first 6 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. Separate from the formal review of the CCA but as part of the EPA's evaluation, the EPA also directed 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 DOE to conduct a PA verification test using the 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. In October 1997, one year after the submission of the CCA, the EPA proposed a rule in the *Federal Register* to certify operation of the WIPP (62 FR 58792); in May 1998, the EPA certified operation of the WIPP (63 FR 27354).

9.2 Summary

Almost immediately after the advent of nuclear weapons research and development, waste disposal became an issue. 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, e.g., meteorite impact and catastrophic faulting, in 1974. 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 disposal system that were pertinent to the hazard under evaluation, i.e., a "scenario". Hazards such as sabotage or nuclear criticality, with very small subjective estimates of probability or consequences, were eliminated. Then consequences of the remaining hazards, such as exposure to drill cuttings containing TRU waste, were formally estimated to determine whether those remaining hazards were sufficient to disqualify the site. The EIS predicted no significant adverse long-term environmental impacts to human health that would support arguments against construction. Thus in 1979, six years after site selection, Congress authorized the DOE to build the WIPP near Carlsbad, New Mexico, for disposal of waste contaminated with TRU radioisotopes. After an exploratory phase in 1981, full construction of the WIPP began in 1983, was essentially complete 5 years later at the end of 1988, and officially complete by January 1990.

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. 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. Between 1989 and 1992, Sandia conducted four preliminary PAs. 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 1998, 19 years after authorization and 25 years after site selection, the EPA certified that the WIPP disposal concept complied with its regulations. The EPA's decision was based on the results from the 1996 PA, which is summarized in this special issue of *Reliability Engineering and System Safety* (Marietta et al., this issue). Pending favorable court rulings on recently filed lawsuits and the issuance by the State of New Mexico of the required RCRA permits for 60% of the TRU waste that contains hazardous waste contaminants, the opening of the WIPP is very near, the culmination of an assessment process of over 25 years.

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1942 LANL Site chosen

1943 AEC's 1st waste decision

1945 Atomic test in NM

1957 NAS recommends exploring waste disposal in salt beds

1963 ORNL Project Salt Vault

- 1942 - Manhattan Engineering District of Corps of Engineers selects site for Los Alamos National Laboratory (LANL) to develop a nuclear bomb. All types of waste initially dumped in canyons at LANL.
- 1943 - Plutonium operations commence and disposal of nuclear waste begins on site at Oak Ridge National Lab (ORNL) in trenches and Clinch River.
- 1944 - Disposal of nuclear waste begins on site at LANL (using trenches, ponds, augered holes) and Hanford Reservation (using railroad cars, trenches, ponds, tanks, underground caissons).
- 1945 - Atomic bomb exploded at Trinity Site near Alamogordo, NM.
- 1946 - Atomic Energy Act (AEA) of 1946:
 - creates Atomic Energy Commission (AEC)
 - establishes government monopoly on atomic weapons and nuclear material
- 1952 - Idaho National Engineering and Environmental Lab (INEEL) completes Radioactive Waste Management Complex (RWMC) for storing and burying waste.
- 1953 - Savannah River Plant (SRP) begins waste storage and disposal on site at "Old Burial Ground".
- 1954 - Rocky Flats Plant near Denver, CO, begins shipping transuranic (TRU) waste to INEEL for disposal at RWMC. AEA of 1954 seeks peaceful uses of atomic energy, thus allows private atomic energy development.
- 1955 - AEC asks National Academy of Sciences (NAS) to examine issue of disposal of radioactive wastes.
- 1957 - NAS recommends radioisotope waste disposal in salt as most promising method. ORNL begins research in salt (1957-61). Plowshare program started to look at peaceful uses of nuclear explosives.
- 1961 - Dec: NAS reaffirms use of New Mexico salt beds for disposal.
- 1962 - US Geological Survey (USGS) reports on domestic salt deposits suitable for waste disposal; the Permian Basin in parts of NM, KS, TX, and OK is one area identified.
- 1963 - ORNL begins Project Salt Vault, a large-scale field test in which irradiated fuel elements and electric heaters are placed in an existing salt mine at Lyons, KS; up to 1967, the tests primarily study near-field effects.

1969 Congress passes National Environmental Policy Act (NEPA)

1970 NAS concludes bedded salt disposal safest choice now available

1970 Lyons site selected for geologic repository

1972 Lyons site judged unacceptable

- 1965 - Savannah River Plant (SRP) begins disposing TRU waste in trenches on site.
- 1966 - NAS reaffirms use of salt beds for disposal and strongly criticizes current disposal practices.
- 1968 - Committee on Radioactive Waste Management established by NAS, later permanent "Board" (BRWM); first task is to reevaluate the use of bedded salt. AEC asks NAS to revisit the issue of nuclear waste disposal. At request of Congress, General Accounting Office (GAO) audits AEC waste management practices and finds faults with records and management. AEC forms task force to address criticisms.
- 1969 - May: Rocky Flats Plant catches fire and cleanup waste sent to Idaho for disposal at RWMC; event focuses public attention on AEC nuclear waste problems. Dec: Congress passes National Environmental Policy Act (NEPA):
 - requires federal agencies to consider environmental consequences of any major action through environmental impact statement (EIS)
 - first US environmental law to be applied to the Waste Isolation Pilot Plant (WIPP)
 International Atomic Energy Agency (IAEA) forms advisory committee to categorize nuclear waste; Alpha contaminated waste one category defined.
- 1970 - Jun: AEC tells Sen. Church that the waste from fire stored at INEEL will be removed by 1980. AEC tentatively selects salt mine in Lyons, KS, as repository. Nov: BRWM of NAS issues report concluding bedded salt satisfactory and safest choice available for nuclear waste disposal.
- 1971 - Many drill holes and some solution mining discovered at Lyons, KS. AEC directs HLW be solidified within 5 yr, stored retrievably, and delivered to federal repository within 10 yr. Congress directs AEC to stop Lyons project until safety is certified. Appeals court requires AEC to look at all environmental impacts in EIS.
- 1972 - May: AEC abandons Lyons project. AEC announces plans for Retrievable Surface Storage Facility (RSSF) for radioactive wastes. EPA and anti-nuclear groups claim RSSF de facto permanent disposal in RSSF EIS.

TRI-6342-5836-2

Fig. 1. Early history of nuclear waste disposal related to the WIPP (adapted from Fig. 2 in Rechard, 1998, which provides references).

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TRI-6342-5892-2

Fig. 3 Uncertainty in model parameters x (expressed as a probability density function [PDF], d , or cumulative distribution function [CDF], D) results in uncertainty in the results $C(x_1 \dots x_{nP})$. This uncertainty (expressed as a complementary distribution function CCDF or 1-CDF) is compared to the limits in 40 CFR 191.

Fig. 5. Stratigraphy above and below the WIPP repository (after Rechard, 1995, Figure 2.1-2)

TRI-6342-5845-0

Fig. 6. Estimated permeability of Salado Formation near repository horizon continued to decrease over the years (Rechard, 1998).

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TRI-6342-5840-2

(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

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Combinations and nomenclature

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

TRI-6342-5851-1

Fig. 12. Technique to construct scenarios from various features, events and processes changed between WIPP EIS and PA calculations after 1989.

1974 Draft of 1st PRA on nuclear reactors

1974 ORNL conducts scenario development and consequence analysis of WIPP

1978 SNL adapts PRA for reactors to nuclear waste repositories

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

1981 SNL reports on PA of hypothetical salt site

- 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. Oct: ORNL conducts first scenario development and deterministic analysis for WIPP. Probability of meteorite impact, probability of fault (and volcanism) and exploratory drilling intersecting disposal area estimated.

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1989 Draft supplemental EIS identifies gas generation as issue

1989 Demo for WIPP PA

1990 First full PA of WIPP used CAMCON

1991 Major models linked in WIPP PA

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

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

- 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. DOE issues Draft Supplemental EIS. Dec: SNL reevaluates release scenarios and issues WIPP PA demonstration outlining process for future PAs.

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- 1995 - Mar: DOE submits DCCA to EPA for review. 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.

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TRI-6342-5841-2

Fig. 13. Engineering analysis of the illustrated behavior of the disposal system (adapted from Fig. 13 in Rechard, 1998, which provides references).

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(a) Direct Releases to the Surface during Drilling

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TRI-6342-6012-0

Fig. 16. Changes in the distribution of cumulative releases (normalized by the EPA limits) for (a) direct releases to the surface during drilling and (b) release via groundwater pathway through Culebra. Since the first complete performance in 1990, direct releases to the surface have dominated overall releases to the accessible environment of the WIPP (Rechard, 1998).

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Fig. 4. Physical setting of the WIPP in Delaware Basin (after Rechard, 1995, Figure 2.1-1)

TRI-6342-5861-1

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