ABSTRACT

The development of the Waste Isolation Pilot Plant is supported by a research and development program which addresses various technical issues for geologic isolation of nuclear waste in bedded salt. These issues include waste interaction with the host material, structural response of the rock, repository sealing, and methods for repository design and operation. This paper discusses the status of various laboratory and field investigations in this program, and presents a summary of recent results for selected issues. In addition, an overview of all activities and the relations of each to plans for implementation of WIPP is presented. These plans include full-scale experimentation with various waste forms in 1986.
TECHNICAL ISSUES OF NUCLEAR WASTE ISOLATION
IN THE WASTE ISOLATION PILOT PLANT (WIPP)

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INTRODUCTION

The Waste Isolation Pilot Plant proposed for implementation in southeastern New Mexico provides a facility for the retrievable storage of defense transuranic wastes and has been recommended by the Department of Energy for a demonstration of spent fuel disposal. In addition, WIPP will include a location for experimentation on the interaction of all waste types with bedded salt environments. These experiments will provide the necessary resolution of various technical issues regarding permanent disposal of nuclear wastes. The integration of the planning and design activities to support this mission and a commensurate research and development program will provide the technical basis for assuring that no adverse impact on public safety or the environment will occur.

An assessment of these elements of the WIPP mission has revealed several technical issues which are currently being addressed by research and development activities at Sandia Laboratories. These activities currently consist of development of predictive models, laboratory analyses for corroboratory data, and bench-scale tests for larger-scale investigations and assessment of synergistic effects. In addition, plans are in progress for development of an "underground laboratory" in southeastern New Mexico for in-situ experiments which will provide full-scale investigation of rock salt responses to non-radioactive sources. These field experiments are anticipated to be in progress by late-1979. These various activities form the sequential steps which culminate in waste emplacement and initiation of experiments with radioactive waste forms in the 1986 time frame, if the WIPP project is approved.

Research on these issues is directed towards reduction of any technical uncertainty which impacts the ability to assess the integrity of radionuclide isolation at the WIPP. This research is guided by bounding calculations for assessment of consequences in which the most conservative assumptions about radiation containment mechanisms are made. Projections of the effects of hypothetical release events have indicated that consequences in terms of human radiation exposures are small in comparison with natural sources. These evaluations, therefore, allow confidence in the continued development of WIPP and provide the reference frame for quantification of the additional measures of protection provided by further analysis and experimentation.

The various technical issues to be resolved are associated with each of the waste types which have been identified with WIPP either for storage or as an experimental form. A summary of these waste types and some pertinent characteristics is presented in Table I. An important distinction between waste types is the amount of heat generation. Both spent fuel and reprocessed waste forms exhibit sufficient power output to induce temperature increases in
salt which have been identified with various other subsequent effects such as thermally accelerated creep in salt, enhanced waste degradation, and mobilization of brine inclusions. While none of these effects provide long-term consequences beyond those predicted in the bounding consequence assessments discussed above, the scientific and public controversy associated with their identification requires that further technical evaluation be performed.

**TABLE I**  
**WIPP WASTE FORMS**

**TRANSURANIC WASTES**

<table>
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<tr>
<th>THD CONTENT</th>
<th>CONTACT HANDLED</th>
<th>REMOTE HANDLED</th>
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<tbody>
<tr>
<td>&gt;10 NCi/GM</td>
<td>210/ DRUM 5-10 GM Pu</td>
<td>200 MREM/HR &lt; SDR &lt; 100 REM/HR</td>
</tr>
<tr>
<td></td>
<td>0.04 WATT/DRUM</td>
<td>4 WATTS/(2 X 15 FT CAN)</td>
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**SURFACE DOSE RATE (SDR) < 200 MREM/HR**

**EXPERIMENTAL WASTE FORMS**

**SPENT FUEL**

600 WATTS @ 10 YR/PWR ASSEMBLY (14" X 183" CAN)

SDR ~ $10^5$ REM/HR

~ 4 KG Pu

**HLW (COMMERCIAL)**

3000 WATTS @ 10 YR/(12" X 120" CAN)

SDR ~ $10^5$ REM/HR

120 GM Pu

**DBLW (DEFENSE)**

500 WATTS/(24" X 120" CAN)

SDR ~ $10^4$ REM/HR
The principal technical issues which the WIPP research and development program addresses are summarized in Table II and outlined as follows:

- The interaction of transuranic wastes with the salt environment including assessment of potential degradation mechanisms and the impact on the repository and radionuclide isolation.

- The interaction of thermal and radiation fields from heat producing wastes with the salt environment and the impact on the waste form encapsulating materials.

- Prediction of the response of the host rock to both the ambient conditions upon excavation and the enhanced deformation anticipated with heat-producing waste forms.

- Characterization of the properties of the host rock for permeation of gases or liquids.

- Assessment of the potential for mobilization of natural fluids in the salt and the subsequent interaction with waste containers.

- Quantification of the technology for sealing man-made penetrations into or near the storage horizons.

- Characterization of the potential for radionuclide migration in the WIPP environment.

- Demonstration and certification of safe operational techniques and appropriate design assumptions.

Each of these issues is being methodically evaluated to accumulate additional data and to correlate all available knowledge into an assessment of the potential consequences to the integrity of the waste isolation mechanisms.
<table>
<thead>
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<th>TABLE II</th>
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<td>TECHNICAL ISSUES FOR WIPP</td>
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<td>RESEARCH AND DEVELOPMENT PROGRAM</td>
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**WASTE INTERACTION**

TRU DEGRADATION AND GAS GENERATION

HLW AND CANISTER DEGRADATION

**ROCK RESPONSE**

ROOM CLOSURE AND RETRIEVABILITY

CANISTER MOTION

DEFORMATION OF OVERLYING FORMATIONS

**REPOSITORY SEALING**

PERMEABILITY

FLUID MIGRATION

BOREHOLE AND SHAFT SEALING

RADIONUCLIDE MIGRATION

**REPOSITORY DESIGN AND OPERATION**
Interaction of Transuranic Waste Form

While the majority of transuranic-contaminated wastes are contact-handled and emit little heat or penetrating radiation (approximately 0.04 watts/55 gal drum and less than 200 mrem/hr surface dose rate), degradation in the WIPP environment is being investigated. The potential exists for radiolytic, pyrolytic, chemical, and bacterial degradation mechanisms. Degradation products could include gases whose presence would inhibit closure of the waste rooms. Laboratory studies of these interactions have been initiated, and a summary of early results under ambient and maximum credible conditions (70°C and 150 Bars) has been recently made by M. A. Molecké (Sandia Laboratories). An example of these results is shown in Figure 1 and indicates that the most significant gas production may be due to bacterial interaction if such organisms can survive in the mine environment.

Additional concerns being addressed are the alteration of radionuclide chemical forms so that the sorptive characteristics are reduced, thereby enhancing migration potential, and an assessment of the integrity of the waste containers to operational and long-term isolation environments.

Interaction of Heat-Producing Wastes

Both spent fuel and high-level waste forms can produce thermal power at initial rates on the order of 0.5-5 kW per canister and surface radiation dose rates of $10^5$ rem/hour of penetrating radiation. These fields could produce local salt temperatures from 80 to 200°C depending on emplacement power density and power per canister. Laboratory investigations are underway which will assess the potential for leaching of various waste forms and corrosion of candidate canister or overpack materials. Studies made under maximum credible conditions of temperature and pressure with brine inundation indicate that, while certain waste forms may degrade if these conditions exist, canister materials have been identified which could prevent any direct interaction with the waste form during the period of significant thermal output ($t_{1/2}$ of approximately 30-50 years). Figure 2 shows relative corrosion rates of a few candidate alloys under the most severe conditions tested (250°C, maximum oxygenation). Titanium-based alloys, for example, have shown corrosion rates of less than $10^{-3}$ cm/year at 250°C in representative WIPP brine under oxygenated conditions. The most serious criterion for canister integrity may, however, be imposed by the requirements for retrieval which include sufficient mechanical strength to accommodate stresses induced by local creep of the rock salt. Current research emphasis is on the influence of repository environments on material performance such as stress corrosion cracking and fracture toughness.

Additional studies are being made of the production of stored radiation damage energy and various associated annealing mechanisms in the WIPP mine-ology. The effect of radiolysis on interaction chemistry, including the production of oxidizing species which can influence corrosion, leaching, and the oxidation state of leached radionuclides, are also under current investigation.
Rock Response

Prediction of rock response during mining and operation and after decommissioning will be required to quantify impacts on the biosphere. Both finite element and finite difference computer models have been developed to model the temperature, stress, and displacement fields which are anticipated. Laboratory analysis of the elastic and creep properties of rock salt have been underway for several years. Various creep models have been formulated which have been based on these laboratory measurements. These models have been compiled recently and summarized by Dawson, while Wawersik has summarized the results of quasi-static and creep tests on WIPP rock salt. The development of a more widely applicable constitutive model for the creep behavior of rock salt has been aided by the incorporation by Munson of existing data into a general structure which identifies appropriate regimes for the action of various mechanisms.

Bench-scale experiments with electric heaters are currently in progress. In-situ testing of mine response to excavation has been done on a limited scale in southeastern New Mexico (SENM) potash mines and will be carried out on a much larger scale, including the effect of electric heaters, when an area dedicated to such experiments is obtained. When construction begins, the excavation of the WIPP facility will be carefully monitored to obtain rock mechanics data.

All this information will be integrated into the development of computer modeling including appropriate constitutive models. These models can then be used to predict such canister-scale phenomena as confining stress, local temperatures, and potential for canister buoyancy. In addition, room-scale calculations are performed to predict closure rates, and regional-scale calculations are performed to assess temperature changes and deformation in overlying formations.

Calculations have been performed using the models and the existing data on rock properties which have estimated the limits of canister motion relative to the salt, the potential for uplift of the repository, and the effects of various canister emplacement configurations.

Migration of Radionuclides

Estimates of the consequences of postulated catastrophic events have assumed that waste forms are carried away by groundwater intrusion from overlying or underlying formations. Prediction of the potential dose to representative local populations require knowledge of the sorptive properties of both disposal horizons and adjacent strata. Laboratory studies are being carried out with both batch and flow-through techniques to gather sorptive data.
Conventional practice for characterizing the sorptive capacity of rocks is through the use of the relative concentration of radionuclide sorbed on a solid mineral phase compared to the concentration in the liquid phase under steady-state conditions. The parameter chosen, $K_d$, the distribution coefficient is then:

$$K_d = \frac{C_{\text{solid}}}{C_{\text{liquid}}} \left( \text{per unit mass} \right) \frac{\text{ml/gm}}{\text{per unit volume}}$$

The parameter is incorporated into the general equation for geosphere transport of radionuclides. Commensurate computer models then solve this set of partial differential equations which describe total flow, energy, salinity, radioactive decay and generation, and radionuclide sorption. For a simple one-dimensional system, a more intuitive parameter, the retardation factor ($R$), can be expressed as:

$$R = \frac{1}{1 + \frac{1}{2} \frac{\rho K_d}{\nu_{\text{fluid}}}} = \frac{\nu_{\text{nuclide}}}{\nu_{\text{fluid}}}$$

where

- $R = \text{nuclide velocity/fluid velocity}$
- $\rho = \text{porosity}$
- $\nu = \text{density}$
- $K_d = \text{distribution coefficient}$

Indications of the effectiveness of sorption in reducing the apparent radionuclide velocity is shown in Figure 2 in which $1/R$ is plotted for typical rock parameters. A $K_d$ of 10 in this simple analogy gives a radionuclide velocity which is a factor of 150 lower than the host fluid for rocks with a porosity of 0.1.

Laboratory studies by Dosch and Lynch\textsuperscript{12} have identified distribution coefficients for both actinides and fission products in the WIPP environment. Results for $K_d$'s of plutonium from these studies are shown in Figure 5. These values, while representative of the conditions of the particular experiments, i.e., pH from 6.5 to 8, powdered samples in simulated brines, and radionuclide concentrations of approximately 1 pCi/ml, indicate $K_d$'s greater than 2000 for the Magenta and Culebra dolomites which overlie the WIPP site. These rocks are the principal water-bearing units above the site and are the assumed path in the scenario for radionuclide release. In addition to high sorption in the overlying rock, Figure 5 also shows that significant sorption can be expected for rocks in the vicinity of potential storage horizons, even in the relatively pure halite and especially in some of the local clays.
Laboratory analyses have also identified a sensitivity of sorptive capacity to radionuclide concentration and to the salinity of the brines. Analyses are currently in progress to quantify these effects.

Additional studies have also identified several materials which have potential application as "getter" materials. These materials—sometimes referred to as engineered backfills—can retard the migration of radionuclides in the vicinity of the waste emplacements. Several materials have been identified: Dewey Lake Redbeds (DLR), a near-surface material at the WIPP site, caliche, tuffs, and various clays, e.g., kaolin (KGA-1) and hectorite (SHCA). Representative $K_d$'s of these materials for $^{152}$Eu are shown in Figure 6.

Permeability of Rocksalt

If gas is generated from waste degradation, it is necessary to predict the ability of these gases to diffuse away from the storage location in order to assess the consequences to the repository. Laboratory permeability studies have been made on WIPP site core in which confining pressures were reestablished in triaxial compressive testing machines. These results indicate that permeabilities are less than 100 nanodarcies after specimens "heal." An example of these permeability measurements which show the sensitivity to the duration of the test is shown in Figure 7 for a rock salt specimen taken at a depth of 2105 feet on the proposed WIPP site. Further testing under in-situ conditions will be completed in May 1979 using high-resolution guarded packer systems in drill holes near the WIPP site at depths comparable to the proposed storage horizons.

Migration of Fluids

Limited data are available on the potential for migration of brine inclusions which are known to be present at about 1/2 percent by weight or less in bedded salt typical of the proposed WIPP horizon. While any impact of the migration toward heat sources may be inconsequential, the significant public controversy on this matter requires development of quantitative data to support a realistic assessment of any consequences.

Laboratory studies to identify the basic mechanisms of brine or moisture release from salt upon heating have been performed. These include small 1 kg samples and a larger 1-meter cylinder with a centrally located 1.5 Kw heater. This latter experiment was initiated in December 1978, terminated in March 1979, and is currently undergoing post-test analysis. Moisture release data from 140 days of testing is shown in Figure 8. These data indicate that the total moisture release is small (approximately 100 grams) even under temperatures of 200°C and gradients greater than 10°C/cm. More significant, however, is the response of the rate of moisture evolution to changes in heater power, especially during reductions of power.

This effect has also been observed in the smaller samples and has been qualitatively correlated with changes in stress state which influence intercrystalline migration. Preliminary assessment of these test results indicates that mechanisms other than movement of small inclusions in thermal....
but also the technical basis for assurance that the appropriate experiments are performed. Data and predictive modeling techniques, which are currently available, can bound the consequences associated with these technical issues. Predictions of the impact on public safety based on these analyses indicate that safe waste disposal in WIPP salt beds is achievable; however, a major use of WIPP will be to conduct realistic experiments with HLW forms to address some of the unresolved details of these waste/salt interactions.
LIST OF FIGURES

Figure 1. Estimated maximum gas production for TRU waste from various mechanisms. (Reference, SAND79-0117)

Figure 2. Corrosion rates in various materials for oxygenated 250ºC solutions. (Reference, SAND78-2111)

Figure 3. Relationship of retardation factor and distribution coefficients.

Figure 4. Plutonium distribution coefficients. (Reference, SAND78-0297)

Figure 5. Distribution coefficients of selected potential getter materials. (Reference, SAND79-1110)

Figure 6. Measured permeability of WIPP rock salt samples, depth of 2105 feet, ERDA 9. (Reference, SAND78-2267)

Figure 7. Cumulative moisture evolution of Salt Block II experiment. (Reference, SAND79-0462)

Figure 8. Comparison of data and computer simulation of water loss from 1 Kg salt sample experiment. (Reference, SAND79-1044J)
FIGURE 1

Estimated Maximum Gas Production for TRU Waste from Various Mechanisms.
Corrosion Rates in Various Materials for Oxygenated 250° Solutions.
Relationship of Retardation Factor and Distribution Coefficient.
FIGURE 5

Distribution Coefficients of Selected Potential Getter Materials.
FIGURE 6

Measured Permeability of WIPP Rock Salt Samples, Depth of 2105 feet, ERDA 9.

PERMEABILITY, $K/D$

ASSUMED UPPER LIMIT IN CALCULATIONS

TIME, HOURS
FIGURE 7

Cumulative Moisture Evolution of Salt Block II Experiment.
Comparison of Data and Computer Simulation of Water Loss from 1 Kg Salt Sample Experiment.
REFERENCES


