DOE/EIS-0026-FS

Volume 2
Appendices

FINAL SUPPLEMENT
ENVIRONMENTAL IMPACT STATEMENT

Waste Isolation Pilot Plant

Volume 2 of 13

January 1990

U.S. DEPARTMENT OF ENERGY
Office of Environmental Restoration
and Waste Management

COVER SHEET

RESPONSIBLE AGENCIES:

Lead Agency: U.S. Department of Energy (DOE)
Cooperating Agency: U.S. Department of the Interior, Bureau of Land Management (BLM)

TITLE:

Final Supplement, Environmental Impact Statement, (SEIS), Waste Isolation Pilot Plant (WIPP)

CONTACT:

For further information, contact:

1) Project Manager
   WIPP Supplemental Environmental Impact Statement Office
   P. O. Box 3090
   Carlsbad, New Mexico 88220
   (800) 274-0585

2) Carol Borgstrom, Director
   Office of NEPA Project Assistance
   Office of the Assistant Secretary for Environment, Safety and Health
   U.S. Department of Energy (EH-25)
   1000 Independence Avenue, SW
   Washington, D.C. 20585
   (202) 586-4600

3) William Dennison
   Acting Assistant General Counsel for Environment
   U.S. Department of Energy (GC-11)
   1000 Independence Avenue, SW
   Washington, D.C. 20585
   (202) 586-6947

ABSTRACT:

In 1980, the DOE published the Final Environmental Impact Statement (FEIS) for the WIPP. This FEIS analyzed and compared the environmental impacts of various alternatives for demonstrating the safe disposal of transuranic (TRU) radioactive waste resulting from DOE national defense related activities. Based on the environmental analyses in the FEIS, the DOE published a Record of Decision in 1981 to proceed with the phased development of the WIPP in southeastern New Mexico as authorized by the Congress in Public Law 96-164.

Since publication of the FEIS, new geological and hydrological information has led to changes in the understanding of the hydrogeological characteristics of the WIPP site as

they relate to the long-term performance of the underground waste repository. In addition, there have been changes in the information and assumptions used to analyze the environmental impacts in the FEIS. These changes include: 1) changes in the composition of the TRU waste inventory, 2) consideration of the hazardous chemical constituents in TRU waste, 3) modification and refinement of the system for the transportation of TRU waste to the WIPP, and 4) modification of the Test Phase.

The purpose of this SEIS is to update the environmental record established in 1980 by evaluating the environmental impacts associated with new information, new circumstances, and proposal modifications. This SEIS evaluates and compares the Proposed Action and two alternatives.

The Proposed Action is to proceed with a phased approach to the development of the WIPP. Full operation of the WIPP would be preceded by a Test Phase of approximately 5 years during which time certain tests and operational demonstrations would be carried out. The elements of the Test Phase, tests and operations demonstration, continue to evolve. These elements are currently under evaluation by the DOE based on comments from independent groups such as the Blue Ribbon Panel, the National Academy of Sciences, the Environmental Evaluation Group, and the Advisory Committee on Nuclear Facility Safety. At this time, the Performance Assessment tests would be comprised of laboratory-scale, bin-scale, and alcove-scale tests. The DOE, in December 1989, issued a revised draft final Test Phase plan that focuses on the Performance Assessment tests to remove uncertainties regarding compliance with long-term disposal standards (40 CFR 191 Subpart B) and to provide confirming data that there would be no migration of hazardous constituents (details are available in Subsection 3.1.1.4 and Appendix O). The tests would be conducted to reduce uncertainties associated with the prediction of natural processes that might affect long-term performance of the underground waste repository. Results of these tests would be used to assess the ability of the WIPP to meet applicable Federal standards for the long-term protection of the public and the environment. The operational demonstrations would be conducted to show the ability of the TRU waste management system to certify, package, transport, and emplace TRU waste in the WIPP safely and efficiently. Waste requirements for the Integration Operations Demonstration remain uncertain. A separate document would be developed to describe in detail the Integration Operations Demonstration following the DOE's decision as to the scope and timing of the demonstration.
During the Test Phase, National Environmental Policy Act (NEPA) requirements would be reviewed in light of the new information developed and appropriate documentation would be prepared. In addition, the DOE will issue another SEIS at the conclusion of the Test Phase and prior to a decision to proceed to the Disposal Phase. This SEIS will analyze in more detail the system-wide impacts of processing and handling at each of the generator/storage facilities and will consider the system-wide impacts of potential waste treatments.

Upon completion of the Test Phase, the DOE would determine whether the WIPP would comply with U.S. Environmental Protection Agency (EPA) standards for the long-term disposal of TRU waste (i.e., 40 CFR Part 191, Subpart B; 40 CFR Part 268). The WIPP would enter the Disposal Phase if there was a favorable Record of Decision based on the new SEIS to be prepared prior to the Disposal Phase and if there was a determination of compliance with the EPA standards and other regulatory requirements. During this phase, defense TRU waste generated since 1970 would be shipped to and disposed of at the WIPP. After completion of waste emplacement, the surface facilities would be decommissioned, and the WIPP underground facilities would serve as a permanent TRU waste repository.

The first alternative, No Action, is similar to the No Action Alternative discussed in the 1980 FEIS. Under this alternative, there would be no research and development facility to demonstrate the safe disposal of TRU waste, and TRU waste would continue to be stored. Storage of newly generated TRU mixed waste would be in conflict with the Resource Conservation and Recovery Act (RCRA) Land Disposal Restrictions; treatment would be required to avoid such conflict. The WIPP would be decommissioned as a waste disposal facility and potentially put to other uses.

The second alternative to the Proposed Action is to conduct the bin-scale tests at a facility other than the WIPP and to delay emplacement of TRU waste in the WIPP underground until a determination has been made of compliance with the EPA standards for TRU waste disposal (i.e., 40 CFR Part 191, Subpart B). The bin-scale tests could be conducted outside the WIPP underground facilities in a specially designed, aboveground facility. The implications of this alternative include delays in both the operational demonstrations and alcove-scale tests, the lack of alcove-scale test data for the compliance demonstration, and placing the WIPP facilities in a "standby" mode. The specialized facility for aboveground bin-scale tests could be constructed at any one of the DOE facilities. In order to analyze the environmental impacts of this alternative in the final SEIS, the DOE has evaluated the Idaho National Engineering Laboratory in Idaho as a representative facility for the aboveground bin-scale tests.

ADDITIONAL INFORMATION:

The 1980 FEIS was reprinted and provided to the public with the draft SEIS which was published April 21, 1989. Public comments on the draft SEIS were accepted for a period of 90 days after publication. During that time, public hearings were conducted in Atlanta, Georgia; Pocatello, Idaho; Denver, Colorado; Pendleton, Oregon; Albuquerque, Santa Fe and Artesia, New Mexico; Odessa, Texas; and Ogden, Utah.

This final SEIS for the WIPP project is a revision of the draft SEIS published in April 1989. It includes responses to the public comments received in writing and at the public hearings and revisions of the draft SEIS in response to the public comments. Revisions of importance have been identified in this final SEIS by vertical lines in the margins to highlight changes made in response to comments. Volumes 1 through 3 of the final SEIS contain the text, appendices, and the summary comments and responses, respectively. Volumes 6 through 13 of the final SEIS contain reproductions of all of the comments received on the draft SEIS, and Volumes 4 and 5 contain the indices to Volumes 6 through 13. An Executive Summary and/or Volumes 1 through 5 of the final SEIS have been distributed to those who received the draft SEIS or requested a copy of the final SEIS. Although not distributed to all who commented on the draft SEIS, Volumes 1 through 13 of the final SEIS have been placed in the reading rooms and libraries listed in Appendix K; these volumes will be mailed to the general public upon request.

A notice of availability of the final SEIS has been published by the EPA in the Federal Register. The DOE will make a decision on implementation of the Proposed Action or the alternatives no earlier than 30 days after publication of the EPA notice of availability. The DOE's decision will be documented in a publicly available Record of Decision to be published in the Federal Register and distributed to all who receive this final SEIS.

Foreword

In October 1989, the Secretary of Energy issued a draft Decision Plan for the Waste Isolation Pilot Plant (WIPP). The Decision Plan listed all key technical milestones and institutional activities for which Departmental, Congressional, or State actions are required prior to receipt of waste for the proposed Test Phase, which is the next step in the phased development of the WIPP. The Plan was issued for review to States, Congressional representatives, other Federal agencies (including the Environmental Protection Agency and the Department of the Interior), and oversight groups (e.g., the Advisory Council for Nuclear Facility Safety, the Blue Ribbon Panel, the National Academy of Sciences, and the Environmental Evaluation Group). Revision 1 of the Plan was issued in December 1989.

Departmental activities required prior to receipt of waste at the WIPP include completion of the "as-built" drawings for the facility, the Energy Systems Acquisition Advisory Board review process, waste-hoist repairs, preoperational appraisal and operational readiness review, mining and outfitting of the alcoves for the proposed Test Phase, and completion of this Supplement to the Environmental Impact Statement.

Other Departmental activities include completion of the Final Safety Analysis Report (FSAR) and issuance of the FSAR addenda to address the proposed Test Phase and associated waste retrieval (if necessary). Future Departmental activities include the planned issuance of the EPA Standards Compliance Summary Report and the evaluation of waste form treatments and design modifications that may be required to meet the EPA Subpart B disposal standards.

Key activities involving oversight groups include final development of an acceptable retrievability program to demonstrate that waste emplaced during the first five years of the facility operation are fully retrievable, and an integrated waste handling demonstration using simulated wastes to ensure system-wide readiness for receipt of wastes for the Test Phase.

Institutional activities include concurrent pursuance of legislative and administrative land withdrawal (legislative withdrawal is the process preferred by the Department); the EPA's ruling on the DOE's No-Migration Variance Petition in compliance with the Land Disposal Restrictions under the Resource Conservation and Recovery Act (RCRA); resolution of regulatory issues, including the State of New Mexico's authority to regulate mixed waste under the RCRA and the designation of routes to be used for transport of transuranic waste; Departmental resolution of any mineral lease at the WIPP; and completion of appropriate agreements with the Western Governors Association and Southern States Energy Board.

This Supplemental Environmental Impact Statement (SEIS) is one of a number of milestones which are critical to the opening of the Waste Isolation Pilot Plant. This SEIS provides an upper bound of the potential impacts of the Proposed Action and alternatives. Based on this final SEIS, the Department will issue a Record of Decision no sooner than 30 days after the EPA publishes a notice of availability in the Federal Register.

TABLE OF CONTENTS

VOLUME 2

APPENDIX A

WASTE ISOLATION PILOT PLANT
WASTE ACCEPTANCE CRITERIA

TABLE OF CONTENTS

LIST OF TABLES

A.1 **INTRODUCTION**

The DOE has established Waste Acceptance Criteria (WAC) for the safe handling and long-term disposal of TRU radioactive waste at the WIPP (DOE, 1989). These criteria establish conditions governing the physical, radiological, and chemical composition of the waste to be emplaced in the WIPP, in addition to specifications for waste packaging to provide for the health and safety of workers and the public. Prior to any waste shipment departing any generator or storage facility, the shipment will be certified to meet the WAC. Similarly, the certification of shipments received at the WIPP will be verified prior to emplacement. The changes to the WAC since 1980 are summarized in Subsection 2.3.1.

The WAC were developed by a DOE-wide committee of experts on the handling and transportation of radioactive material. The basic concepts and limits chosen as WAC requirements are based on personnel safety, handling and storage restrictions at the WIPP facilities, methods of handling equipment, and procedures. Technical justification for the selection of the various requirements is provided in the WAC support documents.[1]

Revisions have been incorporated into the WAC as the WIPP project has evolved. These revisions have been reviewed and commented on by the storage/generator facilities, and others. The WAC is being modified as necessary to ensure compatibility with regulatory requirements such as the TRUPACT-II Certificate of Compliance issued by the Nuclear Regulatory Commission (NRC), the Resource Conservation and Recovery Act (RCRA), and the Department of Transportation (DOT) regulations. Modifications may also result from the Test Phase.

The WAC were established with the assumption that the radiological hazards of TRU mixed waste containing hazardous materials listed in 40 CFR Part 261, Subparts C and D, are much greater than any hazards from associated chemical constituents (Appendix B). Therefore, the WAC focus on the radiological properties of the waste, and the chemical criteria of the WAC are primarily for the prevention of immediate hazards such as fire and explosion. The labeling and data packaging criteria of the WAC also provide for the identification of hazardous waste.

To ensure compliance with the WAC, the DOE has established the WIPP Waste Acceptance Criteria Certification Committee (WACCC) and requires that each facility certify that the WIPP-bound waste meets the WAC. Certification will be directed by the following documents as revised:

DOE 5820.2A, "Radioactive Waste Management"

WIPP-DOE-069, "TRU Waste Acceptance Criteria for the Waste Isolation Pilot Plant"

WIPP-DOE-114, "TRU Waste Certification Compliance Requirements for Acceptance of Newly Generated Contact-Handled Wastes to be Shipped to the WIPP"

---

[1] Vertical lines in the margins denote changes to the draft SEIS made in response to comments.

WIPP-DOE-120, "Quality Assurance Requirements for Certification of the TRU Waste for Shipment to WIPP"

WIPP-DOE-137, "TRU Waste Certification Compliance for Acceptance of Contact-Handled Wastes Retrieved from Storage to be Shipped to the WIPP"

WIPP-DOE-157, "Data Package Format for Certified Transuranic Waste for the Waste Isolation Pilot Plant (WIPP)"

WIPP-DOE-158, "TRU Waste Certification Compliance Requirements for Remote-Handled Wastes for Shipment to the WIPP"

SOP 6.6, "Quality Assurance Audit Program"

These documents may be reviewed in the DOE WIPP Project Office, Carlsbad, New Mexico and all DOE reading rooms.

Each waste generating or storage facility will prepare a TRU Waste Certification Plan that describes the Site Certification Program and how that program meets the WAC and the requirements of the documents listed above. Each facility will also prepare a TRU Waste Quality Assurance Plan that describes their QA program designed to meet the requirements of WIPP-DOE-120. Both of these plans must be approved by the WACCC.

Following the formal approval of Certification and Quality Assurance Plans for the waste generator or storage facility, a compliance verification audit will be performed by the WACCC. Subsequent periodic audits will be performed to verify that the facility is following the approved plans. Audit frequency will be determined by the Chairperson of the WACCC, in consideration of systematic requirements and facility certification status, but will generally be conducted on an annual basis at all facilities. The management of the generator or storage facility is expected to respond to findings and recommendations noted in the audit report, indicating the corrective action taken (or to be taken) to preclude recurrence. If subsequent facility audits determine that corrective action has not been satisfactorily implemented, the WACCC will decertify the waste so that it cannot be accepted at the WIPP.

Since publication of the FEIS, the WAC have been modified twice, and these modifications are summarized in Subsection 2.3.1. A detailed discussion of the WAC and the basis for these criteria are provided in the TRU Waste Acceptance Criteria for the WIPP (DOE, 1989); a summary of the current WAC is provided in Table A.1.1.

REFERENCES FOR APPENDIX A

DOE (U.S. Department of Energy), 1989. "TRU Waste Acceptance Criteria for the Waste Isolation Pilot Plant," WIPP-DOE-069, Rev. 3, U.S. Department of Energy, Carlsbad, New Mexico.

DOE (U.S. Department of Energy), 1987. The DOE Evaluation Document for DOT Type 7A, Type A Packaging, MIL 3245/DOE/DP 0053-HI, Mound Applied Technology, Dayton, Ohio.

Table A.1.1 Summary of WIPP Waste Acceptance Criteria

Criterion	Contact-handled TRU waste	Remote-handled TRU waste
Waste Containers	Waste containers for emplacement at the WIPP shall be noncombustible and meet all the applicable requirements of 49 CFR Part 173.412 for Type A packaging. Waste containers of various sizes shown to meet DOT Type A requirements by the methods detailed in the DOE Evaluation Document for DOT Type 7A, Type A Packaging (DOE, 1987) are acceptable to the WIPP. In addition, they shall have a design life of at least 20 yr from the date of certification. Any waste containers that appear to be bulged or otherwise damaged shall be repackaged or overpacked in a container meeting the above requirements.	RH TRU waste containers shall be noncombustible and meet, as a minimum, the structural requirements and design conditions for Type A packaging contained in 49 CFR 173.412. Due to the special characteristics and application of the RH TRU canister, the compression test requirement in 49 CFR 173.465 (d) is not applicable. In addition, all RH TRU waste containers shall be certified to a WIPP approved specification to have a design life of at least 20 yr from the date of certification.
Waste Container Size	CH TRU waste containers or container assemblies shall not exceed 12 by 8 by 8.5 ft in overall length by width by height dimensions.	RH TRU waste containers shall be no larger than a nominal 26 inches in diameter with a maximum length of 10 ft, 1 inch including the pintle.
Waste Container Handling	All waste containers shall be provided with cleats, offsets, chimes, or skids for handling by means of fork trucks, cranes, or similar handling devices. Lifting rings and other auxiliary lifting devices on the containers, if provided, shall be recessed, offset, or hinged in a manner which does not inhibit stacking the containers.	RH TRU waste containers shall be equipped with an axial lifting pintle of a design acceptable to the WIPP. The containers shall have no other lifting devices.
Specific Activity of Waste	For purposes of TRU waste certification, the 100 nCi/g TRU waste limit shall be interpreted as 100 nCi/g of waste matrix. The weight of added external shielding and the containers should be subtracted prior to performing the nCi/g calculation.	Same as CH TRU waste.

Table A.1.1 Continued

Criterion	Contact-handled TRU waste	Remote-handled TRU waste
Waste Package Weight	CH TRU waste packages or package assemblies shall weigh no more than 21,000 lbs.	RH TRU waste packages shall weigh no more than 8,000 lbs.
Nuclear Criticality	The fissile or fissionable radionuclide content for CH TRU waste containers shall be no greater than the following values, in plutonium-239 fissile gram equivalents: 200 g/55-gal drum 100 g/30-gal drum 500 g/DOT 6M container 5 g/ft^3 in boxes, up to 350 g maximum For materials other than plutonium-239, uranium-235, and Uranium-233, which shall be treated as equivalent, fissile equivalents shall be obtained using ANSI/ANS-8.15-1981.	The fissile or fissionable radionuclide content of RH TRU waste shall not exceed 600 g total (in Pu-239 fissile g equivalents). For materials other than Pu-239, U-235, and U-233, which shall be treated as equivalent, fissile equivalents shall be obtained using ANSI/ANS-8.15-1981.
Plutonium-239 Equivalent Activity[a]	Waste packages shall not exceed 1,000 Ci of Pu-239 equivalent activity (Plutonium Equivalent Curies or PE-Ci).	Same as CH TRU waste.
Surface Dose Rate	Waste containers shall have a maximum surface dose rate at any point no greater than 200 mrem/hr. Neutron contributions of greater than 20 mrem/hr to the total container dose rate shall be reported separately in the data container.	RH TRU waste containers shall have a surface dose rate at any point no greater than 1,000 rem/hr. Neutron contributions are limited to 270 mrem/hr. Neutron contributions of greater than 20 mrem/hr to the total container dose rate shall be reported in the data package. WIPP prior approval is required before RH TRU canisters with a dose rate in excess of 100 rem/hr but less than 1,000 rem/hr may be shipped to the WIPP.[b]

Table A.1.1 Continued

Criterion	Contact-handled TRU waste	Remote-handled TRU waste
Surface Contamination	CH TRU waste containers or container assemblies shall have a removable surface contamination no greater than 50 pCi/100 cm^2 for alpha-emitting radionuclides and 450 pCi/100 cm^2 for beta-gamma-emitting radionuclides.	Same as CH TRU waste.
Thermal Power	Individual CH TRU waste packages in which the average thermal power density exceeds 0.1 watt per cubic foot (W/ft^3) shall have the thermal power recorded in the data container.	The thermal power generated by waste materials in any RH TRU waste container shall not exceed 300 W. The thermal power shall be recorded in the data container.
Gas Generation	Waste containers containing waste forms known or suspected of gas generation, such that a combination of overpressure and explosive mixtures might damage the container in the long term, shall be provided with an appropriate method of pressure relief. Any liner other than plastic bagging shall be provided with positive gas communication to the outer container. Each CH TRU waste shipper shall provide the following data for each waste container: - Total activity (alpha Ci) - Waste form description (from Certification Plan) - Mass and volume percent of organic content For purposes of transportation and emplacement (short term), there will be no mixture of gases or vapors in any container which could, through any credible spontaneous increase of heat or pressure, or through an explosion, significantly reduce the effectiveness of the packaging.	All RH TRU waste containers shall be vented.

Table A.1.1 Continued

Criterion	Contact-handled TRU waste	Remote-handled TRU waste
Labeling	In addition to DOT labeling requirements, each waste container shall be uniquely identified by means of a label permanently attached in a conspicuous location. The container identification number (to be standardized) shall be in medium to low density Code 39 bar code symbology per MIL-ST-1189 in characters at least 1 inch high, and alpha-numeric characters at least 1/2 inch high. The label must be reasonably expected to remain legible and affixed to the container for a period of 10 yrs under anticipated conditions of retrievable storage before shipment to the WIPP and emplacement underground.	Each RH TRU waste container shall be uniquely identified by means of an identification number permanently attached to the container in a conspicuous location using characters at least 2 inches high. The label must be reasonably expected to remain legible and affixed to the container for a period of 10 yr under anticipated conditions of retrievable storage before shipment to the WIPP and emplacement underground.
Data Package	There shall be transmitted to the WIPP operator in advance of shipment, a Data Package/Certification attesting to the fact that the waste package meets the requirements of these criteria. This Data Package/Certification shall be based upon a quality assurance program subject to audit and verification and shall provide information on the items specified below: - Package identification number - Package assembly identification number (if applicable) - Date of waste package certification - WAC exception number (if applicable) - Waste generation site - Date of packaging (closure date) - Maximum surface dose rate in mrem/hr and specific neutron dose rate if greater than 20 mrem/hr.	The data package requirements for RH TRU waste shipments are the same as those for CH TRU waste shipments with the following exceptions: - The container assembly identification requirement does not apply to RH TRU waste shipments. - The cask number shall be used in place of the TRUPACT number.

Table A.1.1 Continued

Criterion	Contact-handled TRU waste	Remote-handled TRU waste
	- Weight (in kilograms) - Container type - Physical description of waste form (content code) - Assay information, including PE-Ci, alpha Ci, and Pu-239 fissile gram equivalent content - Radionuclide information including radionuclide symbol, quantity, and measure (in g or Ci) - Radioactive mixed waste [identity and quantity of hazardous waste characteristic(s)] - Weight and volume percent of organic materials content - Measured or calculated thermal power (if over 0.1 W/ft^3 - Shipment number - Date of shipment - Vehicle type - TRUPACT number(s) - Other information considered significant by the shipper - Name of certifying official who approves the Data Package	
Activity Density	No criterion.	The maximum activity concentration for a RH TRU waste container shall not exceed 23 curies/liter (Ci/l). The concentration may be averaged over the waste container.
Immobilization	Powders, ashes and similar particulate waste materials shall be immobilized if more than 1 weight percent of the waste matrix in each container is in the form of particles below 10 microns in diameter, or if more than 15 weight percent is in the form of particles below 200 microns in diameter.	Same as CH TRU waste.
Liquid Wastes	CH TRU waste shall not be in free-liquid form. Minor liquid	Same as CH TRU waste.

Table A.1.1 Continued

Criterion	Contact-handled TRU waste	Remote-handled TRU waste
	residues remaining in well drained bottles, cans, and other containers are acceptable.	
Pyrophoric Materials	Pyrophoric materials, other than radionuclides, shall be rendered safe by mixing with chemically stable materials (e.g., concrete, glass, etc.) or processed to remove their hazardous properties. No more than 1 percent by weight of the waste in each container may be pyrophoric forms of radionuclides, and these shall be generally dispersed in the waste.	Same as CH TRU waste.
Explosives and Compressed Gases	CH TRU waste shall contain no explosives or compressed gases as defined by 49 CFR Part 173, Subparts C and G.	Same as CH TRU waste.
Radioactive Mixed Waste	CH TRU waste shall contain no hazardous wastes unless they exist as co-contaminants with transuranics. Waste containers containing hazardous materials shall be identified with the appropriate DOT label. TRU contaminated corrosive materials shall be neutralized, rendered noncorrosive, or containered in a manner to ensure container adequacy through the design lifetime. Hazardous materials to be reported are listed in 40 CFR Part 261, Subparts C and D.	Same as CH TRU waste

[a] The Plutonium Equivalent Curies (PE-Ci) concept is described in Appendix F.

[b] The Agreement on Consultation and Cooperation with the State of New Mexico limits the amount of TRU waste that can have a surface dose rate of over 100 rem/hr to 5 percent of the total amount of RH TRU waste.

Table A.1.1 Concluded

Criterion	Contact-handled TRU waste	Remote-handled TRU waste

APPENDIX B

WASTE CHARACTERISTICS

TABLE OF CONTENTS

LIST OF TABLES

B.1 **INTRODUCTION**

This appendix provides information on the characteristics and quantities of the TRU waste that may be emplaced at the WIPP. This information is necessary for assessing the potential impacts of transportation and WIPP operations, as well as the performance of the WIPP over the long term.

Current information and assumptions regarding TRU waste have changed substantially since the WIPP FEIS (DOE, 1980) was published. As explained below, these changes have resulted from changes in the definition of TRU waste, changes in the sources of the waste (i.e., the DOE facilities at which TRU waste is generated or stored), the elimination of experiments with defense high-level waste from the plans for the WIPP, the addition of high-curie radioactive waste and neutron-emitting waste, the decision to evaluate the potential impacts of the hazardous chemicals that are contained in the TRU waste, and an extensive effort to accurately characterize the waste at each of the generator or storage facilities. The characterization effort has provided information about the radionuclide inventory (i.e., the radioactivity, the mass, and the longevity [the half-life] of radionuclides in the waste) and the hazardous chemicals that are present in the waste.

Between 1970, when the category of TRU waste was established, and 1982, TRU waste was defined as waste containing long-lived alpha-emitting radionuclides at a concentration greater than 10 nCi (i.e., 10 one-billionths of a Ci) per g of waste. In 1982, the DOE, having evaluated the potential hazards of TRU waste, decided to change its definition. This new definition was accepted by the EPA (1982) and TRU waste is now defined as waste containing alpha-emitting transuranic radionuclides that have half-lives of 20 years or more and that occur in concentrations exceeding 100 nCi per g of waste. ("Transuranic" in this case means uranium and several radionuclides that are heavier than uranium.) As a result, some waste formerly classified as TRU waste is now classified as low-level radioactive waste, and therefore it is not eligible for disposal in the WIPP. In general, as a result of this change, the average radioactivity of TRU waste has increased.

As in the FEIS, a distinction is made between TRU waste known as contact-handled (CH) waste and TRU waste known as remote-handled (RH) TRU waste (DOE, 1989a). For the CH TRU waste, the radiation-dose rate at the external surface of a waste container (drum or box) must be below 200 mrem (200 one-thousandths of a rem) per hour. This waste can be handled directly by personnel without excessive radiation exposure. The RH TRU waste has surface-radiation-dose rates between 0.2 and 1,000 rem per hour, but only 5 percent of this waste can exceed 100 rem per hour.

In general, the FEIS analyses were based on waste from only two sources: the Idaho National Engineering Laboratory, which was expected to send both CH and RH stored TRU waste, and the Rocky Flats Plant in Colorado, which was expected to send newly generated CH TRU waste. The DOE now expects that post-1970 TRU waste would eventually come from 10 generator and/or storage facilities as discussed in Subsection 3.1.1. Thus, in order to establish the upper limit for the potential impacts, the analyses in this SEIS, like those in the draft Final Safety Analysis Report (FSAR--DOE, 1989b), are based on waste from 10 facilities, and 5 of these facilities have both CH and RH TRU waste.

The consideration of 10 facilities significantly affected assumptions about the contents of average containers of TRU waste, which vary from facility to facility (see Tables B.2.5, B.2.10, B.2.11, and B.2.12). For example, a facility not previously considered, the Savannah River Site, will contribute 92 percent of the plutonium-238 that may be disposed of at the WIPP, and plutonium-238 accounts for nearly half (46 percent) of the total radioactivity of the CH TRU waste that may be emplaced at the WIPP. Similarly, the combined waste from three of the new facilities--Savannah River Site, Los Alamos National Laboratory, and Hanford Reservation--account for 73 percent of the plutonium-241.

Although waste may be received from more facilities, the change in the definition of TRU waste has decreased estimates of waste volumes. The WIPP was designed to receive 6.2 million cubic ft of CH TRU waste and 250,000 cubic ft of RH TRU waste, and the analyses in the FEIS (DOE, 1980) were based on those volumes. However, the DOE's Integrated Data Base, which contains information on the various types of radioactive waste in the United States and is revised annually, shows a decreasing trend. In 1987, the Integrated Data Base (DOE, 1987) reported 5.6 million cubic ft as the estimate for CH TRU waste, both retrievably stored and to be produced from 1987 through 2013 ("newly generated"), whereas the 1988 edition (DOE, 1988) reported a volume of 4.8 million cubic ft, and the 1989 document (DOE, 1989d) estimated a total volume of 4.2 million cubic ft. To provide conservative (i.e., pessimistic) upper limits for the estimated potential impacts of the WIPP, the DOE decided to base the SEIS analyses on the design capacity of the WIPP. Therefore, for the purposes of this SEIS, the volumes given for each generator or storage facility in the 1987 Integrated Data Base were proportionately scaled up to the total design capacity of the WIPP.

Since the publication of the FEIS in 1980, the DOE has attempted to better define the characteristics of the waste. These efforts have included improved sampling of the waste, examination by x-raying, assays of the radioactive-material content, and implementation of improved methods for tracking and recordkeeping. In the FEIS, the information on the RH TRU waste was based on the data available for defense high-level waste, which contains significant amounts of short-lived fission products and therefore has more radioactivity than does the RH TRU waste. The information in the SEIS is based on data collected specifically for RH TRU waste.

The rest of this appendix is divided into two parts: Section B.2, which discusses the radionuclide inventory of the TRU waste, and Section B.3, which covers the hazardous chemical constituents of the TRU waste. The section on the radionuclide inventory includes information on waste volumes and the radioactivities, half-lives, and masses of the radionuclides in the waste. In addition, it explains the procedure used in calculating the following quantities used in various impact analyses: the average radioactivity per shipment of waste, which was used in the analyses of transportation impacts; the average radioactivity per container of waste, which was used in analyzing the safety of WIPP operations; and the radionuclide inventory for the assessment of long-term performance. Section B.2 also discusses two types of TRU waste that were not considered in the FEIS analyses: high-curie and neutron-emitting waste. Section B.3. discusses the hazardous chemical constituents in both CH and RH TRU waste.

The comments on the draft SEIS and continued discussions with personnel at the various waste generating and storage facilities led to the following revisions in this appendix:

- This introduction was rewritten to explain why there are differences in the radionuclide inventory of the FEIS and this final SEIS.

- Tables B.2.2 and B.2.3 were revised to use the correct number of significant digits for waste volumes and to reflect minor redistribution of volume projections for Argonne National Laboratory-East for RH TRU waste.

- The waste volumes in Table B.2.4 were scaled up for all waste facilities in proportion to the volume given for each facility in the 1987 Integrated Data Base (DOE, 1987).

- The text in Subsection B.2.4.1 was modified to more clearly explain how the values given in Table B.2.6 for the radioactivity per waste shipment were calculated. The values were corrected to account for the misapplication of various data.

- Tables B.2.8 and B.2.9 were rearranged to more clearly demonstrate the calculations made to determine the radioactivity per waste shipment.

- The discussion of the transport index in Subsection B.2.4.1 was revised to more clearly explain the source of the radiation that determines the transport index.

- The assumption that the drums of CH TRU waste are filled to 80 percent of their capacity was eliminated because the calculations based on this assumption greatly overestimated the volume of waste to be emplaced in the WIPP.

- Tables B.2.13 and B.2.14, which show the radionuclide inventory used in assessing the long-term performance of the WIPP, were revised by increasing the inventory to represent a volume equal to the design capacity of the WIPP. In addition, the radionuclides in the latter inventory were assumed to have undergone radioactive decay for 100 years to account for the period of institutional control.

- The text on high-activity waste, Subsection B.2.3.2, was modified to more clearly discuss the radioactivity of plutonium-238.

B.2 RADIONUCLIDE INVENTORY OF TRU WASTE

This section discusses the radionuclide inventory of TRU waste and explains how the initial amounts of material needed for assessing environmental impacts were calculated. These quantities serve as the basis for the estimation of the amounts of radioactive material that would be released in a given situation, such as transportation, operation under normal conditions, various accident scenarios that may occur during operations, or unintentional human intrusion after the WIPP has been permanently closed.

B.2.1 WASTE ACCEPTANCE CRITERIA

All waste must be certified to meet the WIPP Waste Acceptance Criteria (DOE, 1989a) before it is transported to the WIPP. The Waste Acceptance Criteria have been refined to reflect the requirements of regulations issued by the U.S. Nuclear Regulatory Commission (NRC) and the Department of Transportation for the transportation of waste and to enhance the safety of long-term isolation. The original criteria were described in Chapter 5 of the FEIS (DOE, 1980); the current criteria are summarized in Subsection 2.3.1 and Appendix A, Table A.1.1.

The Waste Acceptance Criteria that are relevant to the radionuclide source term include the following:

- The surface contamination on containers of CH or RH TRU waste may not exceed 50 percent of the limits specified in Department of Transportation regulations in 49 CFR 173.442.

- The thermal power (the heat-generating capacity) of a package of CH TRU waste must be labeled if it exceeds 0.1 W per cubic ft. The thermal power of RH TRU waste may not exceed 300 W per canister.

In addition, the total plutonium-equivalent curies (PE-Ci) are limited to 1,000 per container. (The PE-Ci concept is discussed in Appendix F). In order to ensure that nuclear criticality (i.e., a self-sustaining nuclear chain reaction) will not occur, the total quantity of fissile material is limited to 200 g per drum. Fissile-material concentrations in boxes (e.g., the standard waste box that may be shipped to the WIPP--see Appendix D) are restricted to a maximum of 5 g per cubic ft, up to a maximum of 350 g per box.

B.2.2 WASTE VOLUMES

The WIPP was designed to receive about 6.2 million cubic ft of CH TRU waste and about 250,000 cubic ft of RH TRU waste, or a total of about 6.45 million cubic ft. These quantities were used in designing the WIPP and in estimating radionuclide inventories for the analyses in the FEIS (DOE, 1980). However, as explained in the introduction to this appendix, the estimated volumes of waste that may be sent to the WIPP have decreased over the years.

When the preparations for the SEIS analyses began, the recent information available on waste volumes was the information given in the 1987 edition of the DOE's Integrated Data Base (DOE, 1987), which is revised annually. This data base showed that the volumes of TRU waste that had been stored since 1970 or were projected to be generated between 1987 and the year 2013 were lower than those estimated for the design of the WIPP: the 1987 estimates were 5.6 million cubic ft for the CH TRU waste and about 95,000 cubic ft for the RH TRU waste, or a total of about 5.7 million cubic ft. The radionuclide inventory for these waste volumes is shown in Table B.2.1, and the waste volumes reported in the 1987 Integrated Data Base are given for each generator or storage facility in Tables B.2.2 and B.2.3 for CH and RH TRU waste, respectively.

The data-base reports issued since 1987 continue to show a decrease in waste volumes. The 1988 Integrated Data Base (DOE, 1988) and the report for 1989 (DOE, 1989d) cite 4.8 and 4.5 million cubic ft, respectively, for the total volume of the TRU waste. However, in order to establish conservative (i.e., pessimistic) upper limits for the potential impacts of the WIPP, the DOE decided to base the analyses in this SEIS on the maximum assumed volume of 6.45 million cubic ft of TRU waste. This was done by scaling up, for each waste generating or storage facility, the volume given in the 1987 data base for CH and RH TRU waste to correspond with the design capacity of the WIPP, with the scaling up being in proportion to the volumes reported in 1987. For CH TRU waste, the 1987 volume was multiplied by 10.7 percent. The scaling-up factor (10.7 percent) was determined by subtracting the volume in the 1987 data base report from the design capacity of the WIPP and dividing this difference by the volume in the 1987 data base report. For RH TRU waste, the volume at each waste facility that may ship RH TRU waste to the WIPP was increased by 163 percent. The scaled-up volumes for each facility are given in Table B.2.4.

B.2.3 **RADIONUCLIDE CHARACTERISTICS**

B.2.3.1 **General Radiation and Radioactivity Characteristics**

In addition to waste volumes, the SEIS analyses of potential impacts from waste transportation and WIPP operations and the assessment of long-term performance required information on the radionuclide composition of the TRU waste (radionuclides and weight fractions) and radioactivity (i.e., number of curies from plutonium and other alpha-emitting TRU radionuclides). These data were obtained from the 1987 Integrated Data Base (DOE, 1987) and additional information that was obtained from each of the waste facilities on fission-product fractions, the total quantities of radionuclides (in curies), and the numbers of actual waste containers in storage and projected through the year 2013. This additional information has been published as a report that documents the waste-characterization data base for the WIPP (DOE, 1989c). Together with the 1987 data base, this report constitutes the basis for the radiological analyses reported in this SEIS and in the WIPP draft FSAR (DOE, 1989b). The 1987 Integrated Data Base (DOE, 1987) was consistently used to establish the volume of waste from

TABLE B.2.1 Currently projected total radionuclide inventories by facility for CH and RH TRU waste

Waste facility[b]	Radionuclide inventory (curies)[a]		
	Retrievably stored waste[c]	Newly generated waste[d]	Total
CH TRU waste			
Idaho National Engineering Laboratory	3.74×10^5	7.61×10^2	3.75×10^5
Rocky Flats Plant[e]	0	1.05×10^6	1.05×10^6
Hanford Reservation	6.85×10^5	1.10×10^6	1.78×10^6
Savannah River Site	8.59×10^5	3.70×10^6	4.56×10^6
Los Alamos National Laboratory	5.96×10^5	1.61×10^6	2.21×10^6
Oak Ridge National Laboratory	2.80×10^4	3.51×10^4	6.31×10^4
Nevada Test Site[f]	4.73×10^2	0	4.73×10^2
Argonne National Laboratory--East[e]	0	7.13×10^2	7.13×10^2
Lawrence Livermore National Laboratory[e]	0	8.45×10^4	8.45×10^4
Mound Laboratory[e]	0	1.87×10^2	1.87×10^2
Subtotal	2.54×10^6	7.58×10^6	1.01×10^7
RH TRU waste			
Idaho National Engineering Laboratory	1.51×10^3	2.28×10^4	2.43×10^4
Hanford Reservation	4.04×10^3	1.93×10^4	2.33×10^4
Los Alamos National Laboratory	3.64×10^3	2.42×10^2	3.88×10^3
Oak Ridge National Laboratory	2.71×10^3	1.84×10^2	2.89×10^3
Argonne National Laboratory--East	0	1.03×10^3	1.03×10^3
Subtotal	1.19×10^4	4.36×10^4	5.54×10^4
GRAND TOTAL	2.58×10^6	7.62×10^6	1.02×10^7

[a] Radionuclide inventories for the waste volumes estimated in the 1987 Integrated Data Base (DOE, 1987)--that is, 5.6 million ft^3 of CH TRU waste and 95,000 ft^3 of RH TRU waste.
[b] Unless indicated otherwise, these facilities both generate TRU waste and are designated as a TRU waste storage facilities.
[c] Stored as of December 31, 1986.
[d] Generated between 1987 and 2013.
[e] Facility that generates but does not store TRU waste.
[f] Facility that does not generate TRU waste, but is designated a TRU waste storage facility.

TABLE B.2.2 Estimated volumes of CH TRU waste in retrievable storage or projected to be generated through the year 2013

Waste facility[b]	Estimated volume (ft^3)[a] Retrievably stored waste[c]	Newly generated waste[d]	Total
Idaho National Engineering Laboratory	1,073,710	9,920	1,083,630
Rocky Flats Plant[e]	0	2,037,600	2,037,600
Hanford Reservation	293,250	537,800	831,050
Savannah River Site	91,465	615,700	707,165
Los Alamos National Laboratory	250,910	302,300	553,210
Oak Ridge National Laboratory	19,160	42,000	61,160
Nevada Test Site[f]	21,290	0	21,290
Argonne National Laboratory--East[e]	0	3,800	3,800
Lawrence Livermore National Laboratory[e]	0	259,400	259,400
Mound Laboratory[e]	0	40,100	40,100
TOTAL	1,749,785	3,848,620	5,598,405

[a] Estimated volumes correspond to the Integrated Data Base for 1987 (DOE, 1987). The volumes of waste used for the environmental analyses in this SEIS are higher and are based on the design capacity of the WIPP.
[b] Unless otherwise indicated, these facilities both generate TRU waste and are designated TRU waste storage facilities.
[c] Stored as of December 31, 1986. From Table 3.5 in the Integrated Data Base for 1987 (DOE, 1987).
[d] Generated from 1987 through 2013. From Table 3.16 in the Integrated Data Base for 1987 (DOE, 1987).
[e] Facility that generates but does not store CH TRU waste (except limited quantities pursuant to RCRA regulations).
[f] Facility that does not generate TRU waste, but is a designated TRU waste storage facility.

TABLE B.2.3 Estimated volumes of RH TRU waste in retrievable storage or projected to be generated through the year 2013

	Estimated volume (ft^3)[a]		
Waste facility[b]	Retrievably stored waste[c]	Newly generated waste[d]	Total
Idaho National Engineering Laboratory	985	4,820	5,805
Hanford Reservation	848	28,600	29,448
Los Alamos National Laboratory	1,020	191	1,211
Oak Ridge National Laboratory	45,478	9,540	55,018
Argonne National Laboratory--East[e]	0	3,500	3,500
TOTAL	48,331	46,651	94,982

[a] Estimated volumes correspond to the Integrated Data Base for 1987 (DOE, 1987). The volumes of waste used for the environmental analyses in this SEIS are higher and are based on the design capacity of the WIPP.

[b] Unless otherwise indicated, these facilities both generate RH TRU waste and are designated TRU waste storage facilities.

[c] Stored as of December 31, 1986. From Table 3.5 in the Integrated Data Base for 1987 (DOE, 1987).

[d] Generated from 1987 through 2013. From Table 3.16 in the Integrated Data Base for 1987 (DOE, 1987).

[e] Facility that generates but does not store RH TRU waste.

each facility that may be placed at the WIPP. The waste-characterization data base (DOE, 1989c) was consistently used to estimate the facility-specific isotopic mixes and radionuclide concentrations. The differences between the waste characteristics assumed in the FEIS (DOE, 1980) and the FSAR are shown in Table B.2.5.

B.2.3.2 **High-Curie Waste**

TRU waste with a high-curie content will be subject to the same surface dose equivalent rate restrictions as other waste; therefore, no unique handling or storage procedures or precautions will be required for this waste. The heat generating (thermal power) capability of high-curie waste may be a concern.

TRU waste generates some heat, most of which is produced when the alpha radiation that is emitted in the radioactive decay of plutonium isotopes interacts with waste materials and the walls of the waste container. The amount of heat that is generated for a given volume depends on the activity (curies) and the average energy of the nuclear disintegrations that release the alpha particles. Waste containing significant fractions of plutonium-238 normally have a higher activity than waste without plutonium-238. This happens because the specific activity (the disintegration rate per gram of material) of plutonium-238 is 100 to 1,000 times higher than that of the other plutonium isotopes. Thus, waste containing large quantities of plutonium-238 is designated high-specific-activity waste, or high-curie waste. Because of the greater heat-generating capacity of plutonium-238, it is also referred to as "heat-source plutonium."

Plutonium-238 is a major contributor to the total radionuclide content of CH TRU waste. This contribution comes mainly from the waste generated at Savannah River Site in South Carolina. This waste has a higher specific activity and heat-generating capacity than the waste considered in the FEIS analyses. Typically, the average plutonium-238 content reported in the FEIS represented 1.2 percent of the total radioactivity of CH TRU waste. The data used for this SEIS indicate that the overall activity of plutonium-238 is 46 percent of the total activity of the waste proposed for disposal in the WIPP, and the activity of the plutonium-238 in the waste from Savannah River Site is approximately 92 percent of the total activity of plutonium-238 in WIPP waste. The higher proportion of plutonium-238 in the total waste has modified the average radionuclide composition of the source term used in this SEIS analyses.

TRU waste with a high-curie content will be subject to the thermal power limits and labeling requirements of the Waste Acceptance Criteria (DOE, 1989a).

B.2.3.3 **Neutron-Emitting Waste**

Since the publication of the FEIS, the DOE has determined that the Oak Ridge National Laboratory in Tennessee may be contributing a small amount of waste containing californium-252. A portion of the radioactive decay for this radionuclide occurs by spontaneous fission with the emission of neutrons (DOE, 1989b). The californium-252 will contribute about 0.03 percent of the total radioactivity in CH TRU waste. Neutron-emitting waste will be subject to the same surface-radiation-rate restrictions as other waste and requires no special precautions or procedures for handling or storage.

TABLE B.2.5 Summary of average TRU waste characteristics[a]

	CH TRU waste		RH TRU waste	
Characteristic[b]	FEIS[c]	FSAR[d]	FEIS[c]	FSAR[d]
Surface dose rate (millirem per hour)[e]				
Drum	3.1	14		
Standard waste box	1.0	14		
Canister		200-100,000	30,000	
Thermal power (watts)[f]				
Drum (maximum)	0.5	0.5		
Standard waste box (maximum)	0.8	0.8		
Canister (average)			70	60
Radioactivity (curies)				
Drum	3.4	20.6		
Standard waste box	5.5	77		
Canister			260[g]	37[g]
Total plutonium content (g)				
Drum	8	15.5		
Standard waste box		13	86.3	
Canister			12.8	120
Fissile material content[h]				
Drum	7.5	17		
Standard waste box	12.2	90		
Canister			12	110

[a] The reasons for the differences between the FEIS and the FSAR values are discussed in Section B.1.
[b] For a discussion of waste containers, see Appendices A and L.
[c] From the WIPP FEIS (DOE, 1980).
[d] From the WIPP draft FSAR (DOE, 1989b). These values were also used in the SEIS. The values in the draft FSAR were derived from DOE, 1989c.
[e] The radiation exposure rate at the outside surface of the package.
[f] The heat-generating capability of the radionuclides.
[g] Daughter products are not included. Average radioactivity per container as reported by facilities. The maximum plutonium-239-equivalent curie (PE-Ci) activity per container is 1000 PE-Ci (DOE, 1989c).
[h] Expressed as the plutonium-239-equivalent fissile content in g. For materials other than plutonium-239, uranium-235, and uranium-233, which are treated as equivalent, fissile equivalents are calculated in accordance with standard ANSI/ANS-8.15-1981 of the American National Standards Institute and the American Nuclear Society.

B.2.4 **CALCULATION OF SOURCE TERMS FOR VARIOUS RELEASE SCENARIOS**

This subsection briefly explains how radionuclide source terms were calculated for the various radioactivity-release scenarios that are included in impact and performance analyses. It shows these calculations for the analysis of potential transportation impacts, for the analysis of safety during WIPP operations, and for the assessment of long-term performance. Examples of calculations are included for greater clarity.

The source term for a particular release scenario is the material at risk multiplied by the fraction of that material that is released (the release fraction) into the environment. The material at risk is the TRU waste material and the surface contamination on a TRU waste container that are potentially available for release under the conditions of the scenario. Examples of the material at risk are the contents of a TRUPACT-II shipping container in a transportation-accident scenario, the contents of two waste drums in an operational-accident scenario in which the drums are punctured by a forklift, the surface contamination on drums with surface contamination plus the contents of drums that are leaking when received in the normal operations scenario, and the total contents of one underground waste disposal panel in the WIPP in a long-term-performance scenario involving human intrusion.

B.2.4.1 **Source Terms for Transportation Analyses**

In calculating the source term for transportation analyses, average radionuclide compositions were derived for each waste facility (DOE, 1989c). These average mixes were derived for four different waste categories: CH TRU waste, RH TRU waste, waste that is retrievably stored, and waste generated between 1987 and 2013 (newly generated waste). These compositions were then used to estimate the radioactivity per waste category as well as the activity per waste container (drum, box, or RH waste canister) (DOE, 1989c).

For the transportation analyses, it was also necessary to determine the average radioactivity per waste shipment (i.e., one trailer load). To determine the average activity per shipment, it is necessary to determine the following:

1) How much of the total radioactivity of the waste at a given facility is in each waste category

2) The normalized radioactivity fractions (as derived in DOE, 1989c) for each radionuclide

3) The average activity per unit volume for the particular waste facility

4) The volume of the transporter (e.g., TRUPACT-II shipping container or a cask for RH waste)

5) The number of transporters per shipment.

These quantities were then used to calculate the average facility-specific quantity of radionuclides per shipment (in curies per trailer load). The results served as the material-at-risk term for calculating the amounts of respirable radionuclides assumed to be released in the hypothetical transportation accidents analyzed in this SEIS (Tables B.2.6 and B.2.7), except in the bounding case scenarios, in which maximum values were assumed.

To be more specific, at any waste facility, for each radionuclide i, the number of curies per shipment was calculated from the following equation:

$$\text{container type Ci/trailer load}_i = \sum_j (AF_j \times RF_{ij} \times AA \times VOL \times TTL)$$

where:

- the container type is the container (drum, box, or canister) for the stored or the newly generated waste and the other terms are defined as follows:

- AF_j =the activity fraction for container type j

- $AF_j = \dfrac{\text{total activity for container type } j}{\text{total activity for the facility}}$

- RF_{ij} = the normalized radionuclide activity fraction for radionuclide i in container type j (DOE, 1989c)

- AA =the average activity per unit volume (in curies per cubic meter) for the waste facility

  $AA = \dfrac{\text{total activity for the facility}}{\text{total volume for the facility}}$

- VOL = the volume (in cubic meters) of the shipping container or cask (2.8 m^3 for the container used for CH TRU waste and 0.89 m^3 for the cask used for RH TRU waste)

- TTL = the number of shipping containers or casks per shipment (three containers for CH TRU waste and one cask for RH TRU waste)

As described in Appendix L, the shipping container for CH TRU waste will be the TRUPACT-II; for RH TRU waste, a shipping cask (e.g., the NuPac 72B cask now being developed) will be used. The total volume of waste for each facility was based on the volume given in the 1987 Integrated Data Base (DOE, 1987) and scaled up to the design capacity of the WIPP, as explained earlier in this appendix. Examples of the calculations made with the equation given above are shown in Tables B.2.8 and B.2.9 for CH waste from Rocky Flats Plant and RH waste from Los Alamos National Laboratory, respectively.

TABLE B.2.7 Average radioactivity in a shipment of RH TRU waste[a]

Radionuclide	Waste facility[b] ANLE	HANF	INEL	LANL	ORNL
Cobalt-60	0.00×10^{0}	2.97×10^{0}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}
Strontium-90	0.00×10^{0}	6.76×10^{0}	4.08×10^{0}	7.99×10^{0}	1.12×10^{0}
Ruthenium-106	0.00×10^{0}	1.89×10^{-3}	0.00×10^{0}	6.31×10^{0}	0.00×10^{0}
Antimony-125	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	1.95×10^{-1}	0.00×10^{0}
Cesium-137	8.83×10^{0}	9.46×10^{0}	5.81×10^{0}	6.18×10^{0}	4.42×10^{-2}
Cerium-144	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	6.22×10^{1}	0.00×10^{0}
Europium-155	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	3.13×10^{-1}	0.00×10^{0}
Thorium-232	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}
Uranium-233	0.00×10^{0}	5.41×10^{-4}	0.00×10^{0}	0.00×10^{0}	4.56×10^{-3}
Uranium-234	0.00×10^{0}	8.11×10^{-5}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}
Uranium-235	1.21×10^{-5}	2.43×10^{-6}	8.68×10^{-2}	9.48×10^{-5}	1.87×10^{-6}
Uranium-238	0.00×10^{0}	5.41×10^{-5}	2.46×10^{-2}	0.00×10^{0}	1.96×10^{-6}
Neptunium-237	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}
Plutonium-238	0.00×10^{0}	9.73×10^{-2}	1.63×10^{-2}	0.00×10^{0}	1.18×10^{-3}
Plutonium-239	2.52×10^{-1}	1.38×10^{0}	8.80×10^{1}	8.29×10^{-1}	3.67×10^{-2}
Plutonium-240	9.27×10^{-2}	4.05×10^{-1}	3.58×10^{1}	2.73×10^{-1}	0.00×10^{0}
Plutonium-241	0.00×10^{0}	8.11×10^{0}	0.00×10^{0}	1.26×10^{1}	0.00×10^{0}
Plutonium-242	0.00×10^{0}	8.65×10^{-5}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}
Americium-241	0.00×10^{0}	5.95×10^{-1}	3.27×10^{-3}	0.00×10^{0}	1.88×10^{-2}
Curium-244	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	1.69×10^{-1}
Californium-252	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	2.91×10^{-1}
TOTAL	9.18×10^{0}	2.98×10^{1}	1.34×10^{2}	9.68×10^{1}	1.68×10^{0}

[a] Radioactivity in curies per shipment for the volumes of waste assumed for the SEIS analyses (i.e., volumes scaled up to correspond to the design capacity of the WIPP--see last column, Table B.2.4). The volume per shipment is 0.89 m^3 (one shipping cask per shipment).

[b] Key: ANLE, Argonne National Laboratory--East; HANF, Hanford Reservation; INEL, Idaho National Engineering Laboratory; LANL, Los Alamos National Laboratory; ORNL, Oak Ridge National Laboratory.

TABLE B.2.8 Quantities used in estimating the average radioactivity in a shipment of CH TRU waste from Rocky Flats Plant[a]

Container type[c]	Total volume[b] (m^3)	Total radioactivity[b] (curies)	Activity fraction
Drums	27,600	880,000	0.771
Boxes (4 x 4 x 7 ft)	4,250	32,000	0.028
TRUPACT-efficient box (TEB)	30,800	230,000	0.201
Total	62,650	1,142,000	1.000

Radioactivity (curies) per container and shipment

Radionuclide	Normalized radionuclide activity fraction[b,d]	Drum	Box	TEB	Total per shipment[e]
Plutonium-238	3.50×10^{-3}	4.13×10^{-1}	1.50×10^{-2}	1.08×10^{-1}	5.37×10^{-1}
Plutonium-239	1.19×10^{-1}	1.40×10^{1}	5.09×10^{-1}	3.66×10^{0}	1.82×10^{1}
Plutonium-240	2.71×10^{-2}	3.19×10^{0}	1.16×10^{-1}	8.33×10^{-1}	4.15×10^{0}
Plutonium-241	8.45×10^{-1}	9.96×10^{1}	3.62×10^{0}	2.60×10^{1}	1.29×10^{3}
Americium-241	5.63×10^{-3}	6.64×10^{-1}	2.42×10^{-2}	1.74×10^{-1}	8.62×10^{-1}

[a] This is an example of the calculations performed for one facility; the calculations for the other nine facilities would be similar.
[b] All of the waste from Rocky Flats Plant is in the newly generated category.
[c] DOE, 1989c.
[d] Same for drums, boxes, and TRUPACT-efficient boxes (TEB) for this facility.
[e] Three loaded TRUPACT-II containers per shipment.

TABLE B.2.9 Quantities used in estimating the average radioactivity in a shipment of RH TRU waste from Los Alamos National Laboratory[a]

Waste type[c]	Total volume[b] (m^3)	Total radioactivity[b] (curies)	Activity fraction
Stored	1.98×10^{1}	2.50×10^{3}	0.912
Newly generated	5.40×10^{0}	2.42×10^{2}	0.088
Total	2.52×10^{1}	2.74×10^{3}	1.000

	Normalized radionuclide activity fraction[b]		Radioactivity (curies) per canister shipment		
Radionuclide	Stored	Newly generated	Stored canisters	Newly generated canisters	Total per shipment[d]
Strontium-90	0.0816	0.0914	7.20×10^{0}	7.78×10^{-1}	7.99×10^{0}
Ruthenium-106	0.0645	0.0723	5.69×10^{0}	6.16×10^{-1}	6.31×10^{0}
Antimony-125	0.0020	0.0022	1.77×10^{-1}	1.88×10^{-2}	1.95×10^{-1}
Cesium-137	0.0632	0.0707	5.58×10^{0}	6.02×10^{-1}	6.18×10^{0}
Cerium-144	0.6356	0.7098	5.61×10^{1}	6.04×10^{0}	6.22×10^{1}
Europium-155	0.0032	0.0036	2.83×10^{-1}	3.08×10^{-2}	3.13×10^{-1}
Uranium-235	0.0000	0.0000	9.18×10^{-5}	2.97×10^{-6}	9.48×10^{-5}
Plutonium-239	0.0091	0.0030	8.03×10^{-1}	2.56×10^{-2}	8.29×10^{-1}
Plutonium-240	0.0030	0.0010	2.65×10^{-1}	8.52×10^{-3}	2.73×10^{-1}
Plutonium-241	0.1377	0.0461	1.22×10^{1}	3.94×10^{-1}	1.26×10^{1}

[a] This is an example of the calculations performed for one facility; the calculations for the other four facilities would be similar.
[b] DOE, 1989c.
[c] All of the RH TRU waste is packaged in a metal canister.
[d] One cask per shipment.

For transportation under normal conditions, the radiological risk depends on the radiation field at the surface of the shipping container or cask. This field is measured in terms of the transport index (TI), which is the radiation-dose rate (in mrem per hour) at 1 m from the surface of the container or cask and is used in calculating radiation exposures under normal transportation conditions.

The radiation field measured by the transport index comes mainly from the gamma radiation released by fission products and other radionuclides (i.e., activation products) in the TRU waste. In CH waste, these products exist in trace amounts and do not contribute sufficient gamma radiation to exceed the limit of 200 mrem per hour for the radiation-dose rate at the surface. These trace amounts are therefore not usually reported in the CH waste inventories. In RH waste, the activation and fission products exist in more significant amounts, as shown in Table B.2.7. The gamma radiation from these products results in radiation-dose rates exceeding 200 millirem per hour and is the reason the waste is assigned to the category of remotely handled, rather than contact-handled, waste.

For the TI used in these SEIS analyses, data from the 1987 Integrated Data Base and the updated radionuclide data (DOE, 1989c) were supplemented with information from the waste facilities. This supplemental information concerned field measurements of the gamma radiation levels around Type A TRU waste containers such as drums and standard waste boxes. The objective of this data-collection effort was to develop a listing of waste containers in terms of the maximum surface dose rates for each facility. From this information, an average for the maximum surface dose rate for the containers from each waste facility was calculated. To ensure that the radiation field was not underestimated, it was assumed that this field resulted entirely from radionuclides emitting photons with an energy of 1 million electron-volts (MeV). In actuality, most of the gamma radiation from CH TRU waste results from the radioactive decay of americium-241 and has an energy of 0.060 MeV. The 0.060 MeV gamma radiation would be significantly attenuated by the TRUPACT-II, while the 1 MeV gamma radiation would not be. The assumption of 1 MeV gamma radiation resulted in radiation levels that exceeded and bounded the expected radiation levels. Shielding models of the TRUPACT-II containers and the shipping cask for RH waste were then developed to calculate the transport index from the 1-MeV radiation fields.

In some cases, the lack of waste-specific information (as in the case of the RH waste from Hanford Reservation) necessitated an assumption about the radiation field. For this SEIS, the Hanford RH waste was assumed to produce a field of 100 rem per hour from the 1-MeV photons (100 rem per hour is the upper limit for 95 percent of the RH waste to be received at the WIPP; the remaining 5 percent may have radiation fields of up to 1,000 rem per hour). This very conservative assumption resulted in a high transport index for RH waste shipments from Hanford Reservation in comparison with the other facilities.

For the CH waste from each waste facility, the number of truck shipments (three TRUPACT-II containers per shipment) was estimated by multiplying the volume per drum (0.2 cubic m) by the number of drums per shipment (42 drums) and dividing this number into the total volume (in cubic meter) of TRU waste (stored and newly generated) at the facility. For rail shipments from facilities with rail access, it was assumed that each shipment carried six TRUPACT-II containers.

For the RH waste, since only one cask will be sent per shipment, the number of shipments was obtained by dividing the volume per shipment (in cubic meters) by the volume per shipping cask (0.89 cubic m). Rail shipments were assumed to carry two casks per shipment. In all of the shipment calculations, the waste was assumed to be the same as in the above-described calculations of radioactivity per container and the Transport Index.

B.2.4.2 **Source Term for WIPP Operational Analysis**

For this SEIS, the analysis of radiation safety during WIPP operations (waste receiving, handling, and emplacement underground) was derived from the WIPP draft FSAR (DOE, 1989b). The safety analyses in the draft FSAR were based on waste inventories reported in Radionuclide Source Term for the WIPP (DOE, 1989c). These safety analyses were scaled up to correspond to the volume design capacity of the WIPP. Scaled-up inventories were used to calculate the number of containers (55-gal drums, standard waste boxes, canisters) that may be processed annually at the WIPP. Average characteristics were also calculated for containers of CH waste (55-gal drums and standard waste boxes) and RH waste (canisters), as shown in Tables B.2.10, B.2.11, and B.2.12. The average radioactivity per container was used in the draft FSAR and the SEIS to analyze the impacts of both normal operations and accidents. Impacts from accidents involving containers with the maximum allowable contents, per the Waste Acceptance Criteria (DOE, 1989a), were also assessed. In assessing occupational safety, the radiation exposures of workers handling waste at the WIPP were based on the same assumptions about radiation fields as those used to calculate the transport index in the transportation-impact analysis.

B.2.4.3 **Source Term for Long-Term Performance Analyses**

The source term used in assessing the long-term performance of the WIPP was derived from the scaled-up waste volumes (Table B.2.4) and the radionuclide composition reported in the waste-characterization data base for the WIPP (DOE, 1989c). A discussion of the source term requirements for the long-term performance analyses, including the decay chains, is in Lappin et al. (1989).

The total inventory of CH TRU waste of approximately 11.4 million curies (Table B.2.13) was modified to account for the decay of short-lived nuclides and the buildup of daughter products with high radiotoxicity (100 years for institutional controls). In addition, radionuclides with low radiotoxicity were eliminated from the inventory. The modified inventory (Table B.2.14) is approximately 3.8 million curies.

The RH TRU waste is not included in the long-term performance-assessment inventory because RH TRU waste constitutes less than 2 percent by activity. Also, as discussed by Lappin et al. (1989), the procedures for emplacing waste in the WIPP will minimize the interaction of RH waste canisters and CH waste rooms. And many of the short-lived radionuclides (which are typically the reason for the waste being assigned to the RH category) will have minimal consequences over the long term. An analysis has been made of the consequences of RH TRU waste being brought directly to the surface by an intruding borehole (see Subsection 5.4.2.6).

TABLE B.2.10 Mass and radioactivity of the radionuclides in an average drum of CH TRU waste[a]

	Mass (g)		Radioactivity (curies)	
Radionuclide	FEIS[b]	FSAR[c]	FEIS[b]	FSAR[c]
Thorium-232	NP	6.0×10^{0}	NP	6.6×10^{-7}
Uranium-233	NP	1.7×10^{0}	NP	1.7×10^{-2}
Uranium-235	NP	4.0×10^{-1}	NP	8.8×10^{-7}
Uranium-238	NP	1.0×10^{1}	NP	3.5×10^{-6}
Neptunium-237	NP	3.1×10^{-2}	NP	2.2×10^{-5}
Plutonium-238	2.5×10^{-3}	6.2×10^{-1}	4.2×10^{-2}	1.1×10^{1}
Plutonium-239	7.5×10^{0}	1.4×10^{1}	4.6×10^{-1}	8.5×10^{-1}
Plutonium-240	5.0×10^{-1}	8.5×10^{-1}	1.1×10^{-1}	1.9×10^{-1}
Plutonium-241	2.7×10^{-2}	6.6×10^{-2}	2.8×10^{0}	6.8×10^{0}
Plutonium-242	2.4×10^{-3}	7.8×10^{-3}	9.4×10^{-6}	3.1×10^{-5}
Americium-241	1.5×10^{-3}	4.9×10^{-1}	5.2×10^{-3}	1.7×10^{0}
Curium-244	NP	4.2×10^{-4}	NP	3.4×10^{-2}
Californium-252	NP	1.0×10^{-5}	NP	5.4×10^{-3}
TOTAL	8.0×10^{0}	3.4×10^{1}	3.4×10^{0}	2.1×10^{1}

[a] The reasons for the differences between the 1980 FEIS and the draft FSAR values are discussed in Section B.1.

[b] From the WIPP FEIS (DOE, 1980). NP indicates that data were not provided in the FEIS.

[c] From the WIPP draft FSAR (DOE, 1989b). These values were also used in the SEIS analyses.

TABLE B.2.11 Mass and radioactivity of the radionuclides in an average standard waste box of CH TRU waste[a]

	Mass (g)		Radioactivity (curies)	
Radionuclide	FEIS[b]	FSAR[c]	FEIS[b]	FSAR[c]
Thorium-232	NP	1.2×10^{1}	NP	1.3×10^{-6}
Uranium-233	NP	6.7×10^{0}	NP	6.5×10^{-2}
Uranium-235	NP	9.6×10^{-1}	NP	2.1×10^{-6}
Uranium-238	NP	2.5×10^{1}	NP	8.3×10^{-6}
Neptunium-237	NP	4.4×10^{-4}	NP	3.1×10^{-7}
Plutonium-238	4.0×10^{-3}	4.2×10^{-2}	6.8×10^{-2}	7.2×10^{-1}
Plutonium-239	1.2×10^{1}	7.9×10^{1}	7.5×10^{-1}	4.9×10^{0}
Plutonium-240	8.1×10^{-1}	6.5×10^{0}	1.8×10^{-1}	1.5×10^{0}
Plutonium-241	4.4×10^{-2}	6.7×10^{-1}	4.5×10^{0}	6.9×10^{1}
Plutonium-242	3.9×10^{-3}	7.5×10^{-2}	1.5×10^{-5}	2.9×10^{-4}
Americium-241	2.5×10^{-3}	2.1×10^{-1}	8.4×10^{-3}	7.3×10^{-1}
Curium-244	NP	8.6×10^{-5}	NP	7.0×10^{-3}
Californium-252	NP	2.1×10^{-6}	NP	1.1×10^{-3}
TOTAL	1.3×10^{1}	1.3×10^{2}	5.5×10^{0}	7.7×10^{1}

[a] The reasons for the differences between the FEIS and the draft FSAR values are discussed in Section B.1.

[b] From the WIPP FEIS (DOE, 1980). NP indicates that data were not provided in the FEIS.

[c] From the WIPP draft FSAR (DOE, 1989b). These values were also used in the SEIS analyses.

TABLE B.2.12 Radioactivity of the radionuclides in an average canister of RH TRU waste[a]

Radionuclide	Radioactivity (curies) FEIS[b,d]	Radioactivity (curies) FSAR[c,d]
Cobalt-60	1.6×10^{0}	1.7×10^{-1}
Strontium-90	2.5×10^{2}	5.1×10^{0}
Ruthenium-106	2.2×10^{0}	3.5×10^{-2}
Antimony-125	NP	1.1×10^{-3}
Cesium-137	1.2×10^{0}	4.3×10^{0}
Cerium-144	NP	3.4×10^{-1}
Uranium-233	NP	5.5×10^{-3}
Uranium-235	NP	3.0×10^{-3}
Uranium-238	NP	1.5×10^{-3}
Plutonium-238	6.5×10^{-2}	5.7×10^{0}
Plutonium-239	7.5×10^{-1}	6.8×10^{0}
Plutonium-240	1.8×10^{-1}	2.2×10^{0}
Plutonium-241	4.6×10^{0}	$1.2 \times 10^{+1}$
Plutonium-242	NP	3.8×10^{-4}
Americium-241	1.2×10^{-2}	2.1×10^{-1}
Curium-244	NP	1.6×10^{-1}
Californium-252	NP	2.8×10^{-1}
TOTAL	2.6×10^{2}	3.7×10^{1}

[a] The reasons for the differences between the FEIS and the draft FSAR values are discussed in Section B.1.

[b] From the WIPP FEIS (DOE, 1980). NP indicates that data were not provided in the FEIS.

[c] From the WIPP draft FSAR (DOE, 1989b). These values were also used in the SEIS analysis.

[d] Daughter products not included.

TABLE B.2.13 Initial radionuclide inventory in CH TRU waste for the assessment of long-term performance[a]

Radionuclide	Half-life (years)	Radioactivity (curies)
Thorium-232	1.41×10^{10}	3.07×10^{-1}
Uranium-233	1.59×10^{5}	9.48×10^{3}
Uranium-235	7.04×10^{8}	4.59×10^{-1}
Uranium-238	4.47×10^{9}	1.84×10^{0}
Neptunium-237	2.14×10^{6}	1.08×10^{1}
Plutonium-238	8.77×10^{1}	5.25×10^{6}
Plutonium-239	2.41×10^{4}	4.89×10^{5}
Plutonium-240	6.54×10^{3}	1.20×10^{5}
Plutonium-241	1.44×10^{1}	4.70×10^{6}
Plutonium-242	3.76×10^{5}	2.13×10^{1}
Americium-241	4.32×10^{2}	7.72×10^{5}
Curium-244	1.81×10^{1}	1.57×10^{4}
Californium-252	2.64×10^{0}	2.51×10^{4}
TOTAL		1.14×10^{7}

[a] This source term is different from that given by Lappin et al. (1989), because it was scaled up to correspond to the design volume of the WIPP. This was done by scaling the source term, by radionuclide, at each waste facility by the volume increment for that facility.

TABLE B.2.14 Modified radionuclide inventory in CH TRU waste for the assessment of long-term performance[a]

Radionuclide	Half-life (years)	Radioactivity (curies)	Mass (g)
Plutonium-238	8.77×10^{1}	2.38×10^{6}	1.39×10^{5}
Plutonium-239	2.41×10^{4}	4.89×10^{5}	7.87×10^{6}
Plutonium-240	6.54×10^{3}	1.20×10^{5}	5.26×10^{5}
Uranium-233	1.59×10^{5}	9.48×10^{3}	9.82×10^{5}
Uranium-234	2.44×10^{5}	1.03×10^{3}	1.64×10^{5}
Uranium-235	7.04×10^{8}	4.59×10^{-1}	2.12×10^{5}
Uranium-236	2.34×10^{7}	0[b]	0
Americium-241	4.32×10^{2}	7.94×10^{5}	2.31×10^{5}
Neptunium-237	2.14×10^{6}	1.08×10^{1}	1.53×10^{4}
Thorium-229	7.43×10^{3}	0[b]	0
Thorium-230	7.70×10^{4}	0[b]	0
Radium-226	1.60×10^{3}	0[b]	0
Lead-210	2.23×10^{1}	0[b]	0
TOTAL		3.79×10^{6}	

[a] The radionuclide inventory in Table B.2.13 was modified by assuming that the radioactivity has decayed for 100 years and, therefore, removing the nontransuranic radionuclides, except uranium.

[b] The radionuclides with zero activity are listed to establish initial amounts for all radionuclides in the decay chains shown in Table 4-3 of the report by Lappin et al. (1989).

B.3 HAZARDOUS CHEMICAL CONSTITUENTS

The FEIS (DOE, 1980) addressed only the impacts of the radioactive component of TRU waste. Since that time, it has been determined that TRU waste is subject to dual regulation under the Atomic Energy Act and the Resource Conservation and Recovery Act (RCRA) because it may also contain hazardous chemical constituents; such waste is called TRU mixed waste. TRU mixed waste is defined as waste that is contaminated with transuranic radionuclides at levels exceeding 100 nCi per g of waste and with hazardous chemical constituents. Information provided by the DOE waste generators indicates that 60 percent of the total TRU waste proposed to be sent to the WIPP over 25 years of operation will contain hazardous waste that is subjected to regulation under RCRA. All shipments of mixed waste are required to meet the conditions of RCRA and the U.S. Department of Transportation (WEC, 1989).

Until recently, few records were required to document the hazardous chemical constituents in TRU waste. The waste was and currently is not routinely sampled and analyzed, because some of the waste is contained in complex matrices and such sampling activities might expose personnel to unacceptable levels of radiation. However, it was possible to determine the composition and other characteristics of TRU mixed waste from knowledge about the waste and the industrial processes from which it was generated. For example, because of the requirements for strict product quality and concerns for safety in handling radioactive materials, production and research activities are highly structured. The ingredients used in a given process and the process conditions are highly controlled. This precision both requires and generates extensive knowledge of the ingredients and the processes involved; it also facilitates the characterization of TRU mixed waste.

This section discusses the hazardous chemical constituents in TRU waste. This information serves as the basis for estimation of the amount of hazardous chemicals that would be released in a given situation.

B.3.1 CH TRU MIXED WASTE

The DOE facilities that may ship waste to the WIPP have used very conservative approaches characterizing their CH TRU mixed waste (i.e., approaches that are likely to overestimate the hazardous chemical constituents in the waste). The conservative approaches were chosen to facilitate preparation of the permit application to operate the WIPP as an "interim status" facility under the RCRA. The characteristics of the waste were recently reported in the Radioactive Mixed Waste Compliance Manual (WEC, 1989) and represent a conservative upper bound for the concentrations of hazardous chemicals in the waste. In other words, if a chemical is present in the waste, it is identified even though its concentration in the waste may be below the regulatory limit.

The identification of the hazardous chemical constituents in CH TRU mixed waste is based on newly generated waste from the Rocky Flats Plant and waste from the Rocky Flats Plant that is currently in retrievable storage at the Idaho National Engineering Laboratory. It is estimated that this waste represents approximately 86 percent by volume of the total CH TRU mixed waste proposed to be emplaced in the WIPP over the 25-year operating life. Furthermore, the Rocky Flats Plant generates many different forms of waste from a variety of processes. Other DOE facilities generate smaller quantities of TRU mixed waste, fewer categories of waste, and waste that contains a narrower range of hazardous chemical constituents (WEC, 1989). Therefore, data on the stored or newly generated waste from Rocky Flats Plant represent a conservative upper bound for the potential risks associated with the chemical components of the CH TRU mixed waste.

In the WIPP Waste Acceptance Criteria (See Section 2 and Appendix A), CH TRU waste is divided into several categories based on the physical characteristics of the materials in the waste. These categories or forms are used by each DOE waste facility to classify its TRU mixed waste. Before shipment to the WIPP, each waste form must be certified by the DOE for compliance with the WIPP Waste Acceptance Criteria. Waste forms identified by the Rocky Flats Plant as containing hazardous chemical constituents are cemented and uncemented aqueous and organic waste, cemented process and laboratory solids, combustible waste, metal and filter waste, inorganic solids, and leaded rubber waste. Each of these waste forms is briefly described below:

- Cemented and uncemented aqueous process waste. This waste consists of a wastewater-treatment sludge that is precipitated at a pH of 10 to 12. The sludge contains alcohols and halogenated organics from the cleaning of equipment and glassware and the degreasing of metal. Some aqueous process waste may also contain metals (e.g., cadmium and lead), although no analyses have been performed to determine specific concentrations. Since 1984, aqueous process waste has been solidified in a process involving neutralization, precipitation, flocculation, clarification, filtration, and solidification with portland cement. Before 1984, this waste was not cemented and it exists today as a damp solid.

- Cemented and uncemented organic waste. Organic waste consists of lathe coolants and degreasing solvents used in plutonium fabrication. Organic waste containing oil and halogenated organic solvents is solidified with Envirostone cement and an emulsifier. Before 1984, this waste was not solidified with cement; it is a damp solid.

- Cemented (immobilized) process and laboratory solids. This waste consists of ion-exchange resins and incinerator ash that has been neutralized and solidified with portland cement. The solvents in this waste come from plutonium-recovery operations.

- Combustible waste. This waste consists of paper and cloth (dry and damp); various plastics, such as polyethylene and polyvinyl chloride; wood; and filters contaminated with trace quantities of halogenated organic solvents. These materials are generated during plutonium recovery and fabrication and in analytical laboratories.

- Metals. The principal constituents of this waste are lead, tantalum, stainless steel, and aluminum. This waste includes equipment, tools, crucibles from laboratories, and molds. Residual halogenated organic solvents may also be present.

- Filters. This waste consists of polypropylene filters and high-efficiency particulate air filters as well as processed filter media. Portland cement is added to absorb any residual liquid and to neutralize residual acids. Exhaust-stream filters may be contaminated with volatile organic solvents used in plutonium fabrication and recovery.

- Inorganic solid waste. This waste contains materials like firebrick, Oil Dri, concrete, and soil. It is generated from the decontamination and decommissioning of plutonium-recovery areas. Oil Dri, concrete, and soil may be contaminated with residual halogenated organic solvents.

- Leaded-rubber waste. This waste consists of the leaded rubber dry-box gloves and aprons that are used throughout plutonium-processing areas. It is considered an RCRA-regulated hazardous waste according to the EPA extraction procedure toxicity test (40 CFR Part 261) for lead, although no analysis has been done to establish the lead concentrations. The EPA toxicity test is used to characterize waste as hazardous under the RCRA.

The estimated quantity of each waste form is given in Table B.3.1. The above descriptions indicate that most of the organic solvents are present in residual quantities from the cleaning of equipment, plastics, glassware, and filters. A major constituent in CH TRU mixed waste is lead, which is present mainly in shielding, dry-box parts, and lead-lined gloves and aprons.

The types and estimated maximum concentrations of hazardous chemical constituents in the various forms of CH TRU mixed waste are given in Table B.3.2. This information is used to determine the types of hazardous chemicals expected in various waste forms and their relative abundance. The concentrations, estimated by the Rocky Flats Plant (Rockwell International, 1988) from knowledge of the waste-generating processes, are very conservative and do not represent the actual concentrations of these chemicals. Information from Clements and Kudera (1985) indicates that the volatile organic compounds in the headspace of drums are well below saturation values for the various chemicals and that the source is limited. A description of the actual hazardous chemical source term used in the hazardous chemical risk assessment is provided in Subsection 5.2.4.

B.3.2 **RH TRU MIXED WASTE**

As discussed in Subsection 2.3, RH TRU waste represents a much smaller portion than CH TRU waste of the total waste proposed for shipment to the WIPP site: the design capacity for RH TRU waste at the WIPP is 250,000 cubic feet. Oak Ridge National Laboratory reported the following two major waste forms in the Radioactive Mixed Waste Compliance Manual (WEC, 1989):

TABLE B.3.1 Estimated quantities of TRU mixed waste (by waste form) from Rocky Flats Plant[a,b]

Description of waste form	Quantity (kilogram)
Cemented and uncemented aqueous waste	1.35×10^7

Cemented and uncemented organic waste	3.27×10^6
Immobilized process and laboratory solids	3.38×10^5
Combustible waste	6.66×10^6
Metal waste	9.65×10^6
Filter waste	2.21×10^6
Inorganic solid waste	4.15×10^5
Leaded rubber waste	3.64×10^5
Total	3.64×10^7

[a] From the Radioactive Mixed Waste Compliance Manual, (WEC, 1989), Appendix 6.4.1.

[b] Quantities include waste projected to be generated through the year 2013 and waste in retrievable storage at the Idaho National Engineering Laboratory.

- Solid RH TRU mixed waste. This waste contains mixtures of combustible materials (e.g., paper, polyvinyl chloride, polypropylene, polyethylene, and Neoprene) and noncombustible materials (e.g., laboratory equipment, tools, and small electric motors) that were removed from an experimental facility at the Oak Ridge National Laboratory (the Alpha Gamma Hot Cell Facility). This waste does not contain free liquids or particulates.

- Sludges. This waste consists of fuel and process sludges that are currently stored in tanks but will be solidified before shipment (with cement or by exposure to microwaves). This waste will be solid packaged in lead-shielded canisters.

The primary hazardous chemical constituent of RH TRU mixed waste is lead, which is used to provide shielding against gamma radiation. Trace quantities of mercury, barium, chromium, and nickel have also been reported in some of the sludges.

REFERENCES FOR APPENDIX B

Clements, T. L., Jr., and D. E. Kudera, 1985. TRU-Waste Sampling Program, Informal Report, Vols. I and II, EEG-WM-6503, TLC-82-85, EG&G Idaho, Idaho Falls, Idaho.

DOE (U.S. Department of Energy), 1989a. TRU Waste Acceptance Criteria for the Waste Isolation Pilot Plant, WIPP-DOE-069, Rev. 3, Carlsbad, New Mexico.

DOE (U.S. Department of Energy), 1989b. Final Safety Analysis Report, Waste Isolation Pilot Plant, draft, DOE/WIPP 89-XXX, Carlsbad, New Mexico.

DOE (U.S. Department of Energy), 1989c. Radionuclide Source Term for the WIPP, 88-005, Carlsbad, New Mexico.

DOE (U.S. Department of Energy), 1989d. Integrated Data Base for 1989: Spent Fuel and Radioactive Waste Inventories, Projections, and Characteristics, DOE/RW-0006, Rev.5.

DOE (U.S. Department of Energy), 1988. Integrated Data Base for 1988: Spent Fuel and Radioactive Waste Inventories, Projections, and Characteristics, DOE/RW-0006, Rev. 4.

DOE (U.S. Department of Energy), 1987. Integrated Data Base for 1987: Spent Fuel and Radioactive Waste Inventories, Projections, and Characteristics, DOE/RW-0006, Rev. 3.

DOE (U.S. Department of Energy), 1980. Final Environmental Impact Statement, Waste Isolation Pilot Plant, WIPP-DOE-0026, Vols. 1 and 2, Carlsbad, New Mexico.

EPA (U.S. Environmental Protection Agency), 1982. Federal Register, Vol. 47, No. 250, p. 58196.

Lappin et al. (A. R. Lappin, R. L. Hunter, D. Garber, and P. B. Davies, eds.), 1989. Systems Analysis, Long-Term Radionuclide Transport, and Dose Assessments, Waste Isolation Pilot Plant (WIPP), Southeastern New Mexico, March 1989, SAND89-0462, Sandia National Laboratories, Albuquerque, New Mexico.

Rockwell International, 1988. "Hazardous Constituents of Rocky Flats Transuranic Waste," Internal Letter for distribution from J. K. Paynter, May 24, 1988, WCP02-31, Golden, Colorado.

WEC (Westinghouse Electric Corporation), 1989. Radioactive Mixed Waste Compliance Manual, Appendix 6.4.1, WP-02-07, Rev. 0, Waste Isolation Pilot Plant, Carlsbad, New Mexico.

TABLE B.2.4 Volumes of stored and newly generated TRU waste, scaled up to equal the design capacity of WIPP[a]

Estimates from 1987 IDB[b]

Waste facility[c]	Stored waste	Newly generated waste	Total base	Volume scale-up	Estimate used in SEIS analyses
CH TRU waste					
Idaho National Engineering Laboratory	1.07×10^6	9.92×10^3	1.08×10^6	1.16×10^5	1.20×10^6
Rocky Flats Plant[d]	0.00×10^0	2.04×10^6	2.04×10^6	2.19×10^5	2.26×10^6
Hanford Reservation	2.93×10^5	5.38×10^5	8.31×10^5	8.93×10^4	9.20×10^5
Savannah River Site	9.15×10^4	6.16×10^5	7.07×10^5	7.60×10^4	7.83×10^5
Los Alamos National Laboratory	2.51×10^5	3.02×10^5	5.53×10^5	5.95×10^4	6.13×10^5
Oak Ridge National Laboratory	1.92×10^4	4.20×10^4	6.12×10^4	6.77×10^3	6.77×10^4
Nevada Test Site[e]	2.13×10^4	0.00×10^0	2.13×10^4	2.29×10^3	2.36×10^4
Argonne National Laboratory--East[d]	0.00×10^0	3.80×10^3	3.80×10^3	4.10×10^2	4.22×10^3
Lawrence Livermore National Laboratory[d]	0.00×10^0	2.59×10^5	2.59×10^5	2.79×10^4	2.87×10^5
Mound Laboratory[d]	0.00×10^0	4.01×10^4	4.01×10^4	4.31×10^3	4.44×10^4
TOTAL	1.75×10^6	3.85×10^6	5.60×10^6	6.02×10^5	6.20×10^6
RH TRU waste					
Idaho National Engineering Laboratory	9.85×10^2	4.82×10^3	5.80×10^3	9.48×10^3	1.53×10^4
Hanford Reservation	8.48×10^2	2.86×10^4	2.94×10^4	4.80×10^4	7.75×10^4
Oak Ridge National Laboratory	4.55×10^4	9.54×10^3	5.50×10^4	8.97×10^4	1.45×10^5
Argonne National Laboratory--East[d]	0.00×10^0	3.50×10^3	3.50×10^3	5.76×10^3	9.29×10^3
Los Alamos National Laboratory	1.02×10^3	1.91×10^2	1.21×10^3	1.97×10^3	3.18×10^3
TOTAL	4.83×10^4	4.46×10^4	9.29×10^4	1.57×10^5	2.50×10^5

[a] All quantities are in cubic feet (ft^3). The design capacity of the WIPP is 6.2 million ft^3 of CH TRU waste and 250,000 ft^3 of RH TRU waste.
[b] Estimates from 1987 Integrated Data Base (DOE, 1987) for waste stored as of December 21, 1986, and waste generated from 1987 through 2013.
[c] Unless otherwise indicated, these facilities both generate TRU waste and are designated TRU waste storage sites.
[d] Facility that generates but does not store TRU waste.
[e] Facility that does not generate TRU waste, but is a designated TRU waste storage facility.

TABLE B.2.6 Average radioactivity in a shipment of CH TRU waste [a]

	Waste facility[b]									
Radionuclide	ANLE	HANF	INEL	LANL	LLNL	Mound	NTS	ORNL	RFP	SRP
Thorium-232	0.00×10^{0}	0.00×10^{0}	5.17×10^{-5}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	4.26×10^{-4}	0.00×10^{0}	0.00×10^{0}
Uranium-233	0.00×10^{0}	0.00×10^{0}	1.53×10^{-1}	2.95×10^{-2}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	3.85×10^{1}	0.00×10^{0}	0.00×10^{0}
Uranium-235	0.00×10^{0}	0.00×10^{0}	5.79×10^{-6}	8.37×10^{-5}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	1.15×10^{-3}	0.00×10^{0}	0.00×10^{0}
Uranium-238	0.00×10^{0}	0.00×10^{0}	9.72×10^{-6}	3.61×10^{-4}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	4.59×10^{-3}	0.00×10^{0}	0.00×10^{0}
Neptunium-237	9.65×10^{-4}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	4.09×10^{-3}
Plutonium-238	5.39×10^{0}	3.08×10^{0}	1.08×10^{1}	1.67×10^{2}	3.42×10^{-1}	1.36×10^{0}	3.82×10^{-2}	5.75×10^{1}	5.37×10^{-1}	1.83×10^{3}
Plutonium-239	3.41×10^{0}	3.30×10^{1}	5.89×10^{0}	8.86×10^{1}	8.23×10^{0}	1.18×10^{-2}	6.46×10^{-1}	1.24×10^{2}	1.82×10^{1}	2.20×10^{0}
Plutonium-240	1.56×10^{0}	1.18×10^{1}	1.44×10^{0}	2.04×10^{1}	2.36×10^{0}	3.10×10^{-3}	1.53×10^{-1}	0.00×10^{0}	4.15×10^{0}	8.81×10^{-1}
Plutonium-241	3.10×10^{1}	5.98×10^{2}	4.55×10^{1}	6.88×10^{2}	7.84×10^{1}	1.19×10^{-3}	5.76×10^{0}	0.00×10^{0}	1.29×10^{2}	6.61×10^{1}
Plutonium-242	0.00×10^{0}	2.66×10^{-3}	0.00×10^{0}	4.00×10^{-3}	1.29×10^{-4}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	3.70×10^{-4}	7.19×10^{-4}
Americium-241	1.41×10^{1}	0.00×10^{0}	3.89×10^{1}	2.90×10^{2}	6.81×10^{0}	0.00×10^{0}	0.00×10^{0}	1.04×10^{1}	8.62×10^{-1}	1.81×10^{-1}
Curium-244	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	6.90×10^{1}	0.00×10^{0}	0.00×10^{0}
Californium-252	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	1.10×10^{1}	0.00×10^{0}	0.00×10^{0}
TOTAL	5.55×10^{1}	6.46×10^{2}	1.03×10^{2}	1.25×10^{3}	9.62×10^{1}	1.38×10^{0}	6.59×10^{0}	3.10×10^{2}	1.53×10^{2}	1.89×10^{3}

[a] Radioactivity in curies per shipment for the volumes of waste assumed for the SEIS analyses (ie., volumes scaled up to correspond to the design capacity of the WIPP--see last column, Table B.2.4). The volume per shipment is 8.4 m^3 (three TRUPACT-II containers per shipment, with 2.8 m^3 per TRUPACT-II shipping container).

[b] Key: ANLE, Argonne National Laboratory--East; HANF, Hanford Reservation; INEL, Idaho National Engineering Laboratory; LANL, Los Alamos National Laboratory; LLNL, Lawrence Livermore National Laboratory; Mound, Mound Laboratory; NTS, Nevada Test Site; ORNL, Oak Ridge National Laboratory; RFP, Rocky Flats Plant; SRP, Savannah River Plant.

TABLE B.3.2 Estimated maximum concentrations of hazardous chemical constituents in CH TRU mixed waste from the Rocky Flats Plant

Hazardous chemical constituent[b]	Estimated maximum concentration (milligrams per kilograms)[a]							
	Aqueous waste[c]	Organic waste[c]	Process and laboratory solids[d]	Combustible waste	Metal waste	Filter waste	Inorganic solids	Leaded rubber waste
1,1,1-Trichloroethane	75	150,000	200	2,000	15	150	900	0
Carbon tetrachloride	25	50,000	25	750	10	150	100	0
1,1,2-Trichloro-1,2,2-trifluoroethane	100	50,000	200	1,500	75	100	8,000	0
Methylene chloride	700	0	100	750	200	50	700	0
Methyl alcohol	25	0	15	0	0	0	0	0
Xylene	50	0	50	0	0	0	0	0
Butyl alcohol	10	0	10	0	0	0	0	0
Cadmium	10	0	0	0	0	0	0	0
Lead	10	0	400	0	1×10^6	0	0	6×10^5

[a] Data from Rockwell International (1988).

[b] The hazardous chemical constituents were determined from knowledge of the processes used in generating the waste. The given maximum concentrations for the specific waste forms were calculated in an extremely conservative manner and hence are likely to be greatly overestimated. No analytical data are available for the hazardous chemical constituents in these waste forms.

[c] Cemented and uncemented sludges.

[d] Neutralized and immobilized (cemented) solids.

APPENDIX C

TRANSPORTATION EMERGENCY PLANNING

TABLE OF CONTENTS

LIST OF FIGURES

C.1 **INTRODUCTION**

Of 500 billion domestic shipments annually, about 100 million, or 0.02 percent, are shipments of hazardous materials, and 3 million, or 0.0006 percent, are shipments of radioactive materials. The vast majority (95 percent) of the radioactive-material shipments involve small quantities for general users like hospitals, research laboratories, and industries. The remaining 5 percent are large quantity shipments for commercial reactors or shipments related to national defense (Wolff, 1984).

The safety record of the radioactive-material shipments is outstanding. No serious injuries or deaths have ever resulted from the radioactive materials carried in these shipments. The main reason for this outstanding safety record is the stringent Federal requirements for the packagings, shipping containers, and shipping casks that must be used for radioactive materials. Accidents that have released radioactive material from limited quantity, or Type A containers, have resulted in insignificant consequences and in each case the material was cleaned up, and no one was injured from the radioactivity. Large quantity, or Type B containers and casks are occasionally involved in transportation accidents; fifty such containers or casks were involved in accidents between 1971 and 1985 (DOE, 1989a). No Type B packages have ever released their radioactive contents because of impact or fire, except for a radiography camera failure.

As described in Appendix L, the packagings that will be used for shipping TRU waste to the WIPP are in the Type B category and are designed to withstand severe accidents without releasing their contents. However, as an additional precaution the DOE continues to ensure its emergency-response capabilities and procedures to protect public health and safety after transportation accidents. The current status of those capabilities and the plans for their future development are discussed in this appendix.

Planning for radiological emergency preparedness, including transportation activities, began several years ago. State, Tribal, and local governments as well as the DOE and several other Federal agencies have been closely involved in this effort. The Federal effort includes developing transportation-specific planning guidance and reviewing generic State radiological emergency-response plans.

This appendix describes the responsibilities and resources available for responding to emergencies in general and transportation accidents in particular. Then it presents a detailed discussion of the emergency-response responsibilities in transportation to the WIPP and presents the procedures to be followed by the carrier of the waste (i.e., the WIPP trucking contractor); the State, Tribal, and local governments; and various organizations in, or employed by, the DOE. The subsection on procedures is followed by a discussion of the training programs that the DOE has conducted in various States. To illustrate how the carrier, the State and local governments, and the DOE would respond in a given accident situation, the last subsection in this appendix describes a hypothetical accident and emergency-response scenario. In addition, it describes the responses to actual transportation accidents and incidents involving radioactive materials.

This appendix has been rewritten in response to the many comments received which requested additional clarification and detail concerning emergency-response capabilities and plans in the event of transportation accidents.

C.2 **OVERVIEW OF RESPONSIBILITIES AND RESOURCES IN EMERGENCY RESPONSE**

In the Civil Defense Act of 1950, the U.S. Congress broadly defined the roles and responsibilities of the Federal Government in responding to nuclear attacks and other emergencies in general. Following a tradition established early in the history of the United States, the Act assigned to State and local governments primary responsibility for implementing measures to protect life and property, whereas Federal agencies were given responsibility for providing assistance when requested by State, Tribal, and local governments. Subsequently, responsibilities were also defined for the shippers and carriers of hazardous materials, including radioactive waste.

This subsection reviews emergency-response responsibilities and roles. It also discusses the resources that are available for emergency response. The discussion is not specific to WIPP transportation; emergency response for WIPP transportation is discussed in Subsection C.3.

C.2.1 **OVERVIEW OF RESPONSIBILITIES**

The general roles of shippers; carriers; State, Tribal, and local governments; and Federal agencies can be summarized as follows:

- Shippers. The shipper is required to provide to the carrier, at the time of shipment, any special precautions required for each shipment. If called on in case of an accident, the shipper will also provide information that may be necessary for, or helpful in, emergency-response activities.

- Carriers. The carrier has the initial responsibility for minimizing radiation hazards to the public and notifying State, Tribal, and local authorities of accidents in their jurisdictions.

- States, Tribal, and local governments. These entities have primary responsibility for implementing measures at the scene of the accident in order to protect life, property, and the environment.

- Federal agencies. If requested, assistance from Federal agencies is available to support the emergency-response measures taken by State, Tribal, and local governments.

In the case of transportation to the WIPP, the DOE has responsibilities in two of the above categories: 1) the DOE is the shipper, and 2) the DOE is a Federal agency that can provide assistance if requested by State, Tribal, or local governments. As shipper and owner, the DOE would respond directly to transportation accidents involving the TRU waste.

This subsection describes the State, Tribal, and local responsibilities for emergency response. Although State, Tribal, and local governments have a more important role in emergency response, and Federal assistance is rarely required in a transportation accident, this subsection also presents a comprehensive discussion of Federal emergency-response resources which allows the reader to understand the types of assistance that are available to State, Tribal, and local governments.

C.2.2 **GENERAL RESPONSIBILITIES AND RESOURCES OF STATE, TRIBAL, AND LOCAL GOVERNMENTS**

In the event of a transportation accident involving radioactive waste, State, Tribal, and local governments are responsible for taking measures to protect life, property, and the environment. This might entail direct actions, such as rescuing people from a wreck, extinguishing fires, and giving first aid to the injured, as well as protective actions, such as keeping people away from the area of the accident. These are activities that usually occur within the first 30 minutes of a response and are normally performed by local governments. If the local government determines that its response capabilities have been exceeded, which is often the case in incidents involving radioactive materials, they would request additional radiological monitoring and assessment help from a State government organization. In addition, State, Tribal, and local governments must ensure that cleanup and decontamination activities, if necessary, meet their standards.

In 1980, the Nuclear Regulatory Commission (NRC) published a survey[1] (Mitter et. al, 1980) of State emergency-response capabilities for responding to transportation accidents. The NRC Survey reports that the number of requests for State assistance in transportation accidents involving radioactive materials is 275 per year, or a mean of 5.6 requests per State per year. Many of the States responding to the survey stressed that most of these accidents are not serious, the shipping containers or casks retain their integrity, and there is rarely any release of radioactive material. Some of the respondents mentioned that they were more concerned about accidents involving hazardous chemicals. However, knowledge that most transportation accidents involving radioactive materials are not serious does not diminish the need for technical expertise at the scene, because hasty decisions or actions by uninformed personnel can lead to unnecessary panic. In one accident, for example, a civil-defense volunteer who was among the first responders used a pocket dosimeter that had not been calibrated for more than a year. The worker's defective dosimeter indicated a near-lethal reading of radiation dose, causing an entire township to panic. The State response team later determined that there had been no radiation leakage.

Forty-six States responding to the NRC survey (Mitter et al., 1980) reported that they had never needed to call on Federal assistance in transportation accidents involving radioactive material. Four of these States, however, have DOE installations within their borders; these installations are routinely notified and respond on behalf of the State, if they are the nearest source of qualified personnel. Only three States reported having called for Federal personnel, and one of these stated that they asked for Federal assistance to verify the integrity of shipping casks that had been involved in a rail accident. In addition, several States mentioned that in some incidents involving shipments from or to Federal installations, the drivers had notified the Federal installation, which sent personnel to respond. As discussed in Subsections C.2.3.1 and C.2.3.2, when the DOE is the shipper, the DOE will respond automatically. If DOE receives notification of an accident from its carrier, they will provide this information to the State and coordinate the response.

[1] This survey is currently being updated.

To be prepared to respond, it is necessary to develop and implement emergency-response plans. The rest of this subsection briefly describes planning by State, Tribal, and local governments; guidance for evacuation plans; and capabilities.

C.2.2.1 **Response Plans**

State, Tribal, and local governments are generally responsible for providing the first response to a transportation accident. In addition, according to a guidance document issued by the Federal Emergency Management Agency (FEMA, 1988), the local government must determine the action required in order to prevent further damage to life or property. (State and local statutes should be consulted to determine specific responsibilities.) Cleanup and decontamination may be performed by any of a number of organizations, but the carrier and shipper have ultimate financial responsibility. The State does have a responsibility to assure that cleanup is in compliance with State-established levels. In the event State, Tribal, or local governments expend resources for activities needed to mitigate the effects of the accident, these expenses would be reimbursable (see Subsection C.2.3.6).

Under Federal and State regulations, each State, Tribal, and local government is responsible for developing emergency-response plans and for providing the first response to emergencies involving radioactive material. As discussed in the subsequent subsections, assistance is available from the Federal government for planning for emergency preparedness and evaluating the adequacy of the plans.

States have generic plans for responding to emergencies involving radioactive materials. These plans include procedures for notifying the organizations that can provide the required assistance and lists of organizations to call in order to initiate the proper response. There is no requirement for State, Tribal, and local governments to develop specific plans for responding to transportation accidents involving radioactive materials. The guidance document issued by the FEMA (FEMA, 1988) suggests that planning for transportation accidents be closely integrated into generic emergency operating plans for all types of disasters and emergencies.

C.2.2.2 **Evacuation Plans**

In a transportation accident, the State, Tribal, or local government has the responsibility for taking emergency protective actions, like evacuation. It should be noted, however, that a transportation accident involving radioactive materials, unlike an accident involving explosives or noxious gases, is not likely to require an evacuation in the ordinary sense. At most, in the unlikely event that some radioactive material is released, it would be necessary to establish a small control zone (with a radius of 150 feet from the source) from which people would be excluded until cleanup was completed.

Federal agencies clearly have the responsibility to coordinate emergency preparedness with other jurisdictions. To this end, the DOE, through its States Training and Education Program (STEP), has attempted to provide decision makers at the State, Tribal, and local levels with accurate information to develop written procedures for making protective-action decisions, such as evacuations.

For example, DOE's STEP training course presents the recommendations of the FEMA guidance document (FEMA, 1988) and the DOT's Emergency Response Guidebook (DOT, 1987) to establish "an upwind exclusion area of at least 150 feet" after an accident involving radioactive materials. In addition, radiological health and environment professionals at the State and county level have been given specific information about the generic contents and hazards of the transuranic waste that may cross their boundaries. This information includes radiation exposure rates and long-term effects expressed in probabilities of developing cancer.

C.2.2.3 **Capabilities**

The number of resources (and thus capabilities) available to State, Tribal, and local governments depends on the types of industry located within their boundaries. States with operating commercial reactors have more resources, because a demonstrated emergency-response capability must be established in order for a reactor to be licensed by the NRC. All States have functionally oriented radiological health and emergency management organizations. These organizations include trained staff and specialized equipment.

Most first responders do not maintain the capability to measure or detect radiation or radioactive material at the scene of an accident. However, the Committee on Emergency Response Planning of the Conference of Radiation Control Program Directors advised the Federal committee that revised the FEMA guidance document (FEMA, 1988) that a radiation detection instrument is not necessary in first response to a transportation accident. The role of the first responders is to deal with preservation of life, health, and property. This generally means extinguishing or preventing fires and saving lives. First responders, therefore, should arrive at the scene with adequate protective clothing. For example, bunker clothes or turnout gear and self-contained breathing apparatus are typically used by responding firefighters and some rescue personnel. This type of gear will give sufficient protection against the inhalation of radioactive material such as would be transported to the WIPP and would prevent external contamination. Protection is also provided by the surgical gloves (or their equivalent) and masks that have been issued to most ambulance, rescue, and law- enforcement personnel.

State-level radiological health personnel would respond with protective clothing (shoe covers, gloves, coveralls, and respirators) and portable instruments for detecting and measuring radiation. Many States have mobile laboratories for analyzing environmental samples. Information generated by State radiological field teams would be provided to the decision makers responsible for recommending protective actions to nearby residents.

In addition, if the State and local resources need to be supplemented, the resources of the Federal government, primarily the DOE, can be requested to support radiation monitoring and assessment.

C.2.3 **FEDERAL ASSISTANCE IN EMERGENCY RESPONSE**

C.2.3.1 **The Federal Emergency Management Agency and the Framework for Federal Assistance**

Until 1979, several Federal agencies had responsibilities related to emergency response, and no single agency was charged with coordinating their efforts. To consolidate resources and capabilities, the Federal Emergency Management Agency (FEMA) was created in April 1979 by Presidential order. The FEMA was created from the following five agencies: the Defense Civil Preparedness Agency (Department of Defense), the Federal Disaster Assistance Administration (Department of Housing and Urban Development), the Federal Preparedness Agency (General Services Administration), the U.S. Fire Administration (Department of

Commerce), and the Federal Insurance Administration (Department of Housing and Urban Development). In addition, the FEMA took over the responsibilities of the NRC for planning the activities of State and local governments in emergency response for radiation-related accidents. The NRC, however, provides technical assistance and expertise to the FEMA.

The FEMA was subsequently made responsible for establishing policies for and coordinating all Federal functions in civil-defense and civil-emergency planning, management, mitigation, and assistance. The Director of the FEMA represents the President in working with State and local governments and the private sector to stimulate active participation in planning and implementing programs for civil-emergency response and recovery. Civil emergencies include transportation accidents involving radioactive materials. The FEMA has entered into cooperative agreements with each of the States, and under these agreements it provides financial assistance to the States to support planning, preparedness, and response activities (see Subsection C.2.3.6).

In 1985, the FEMA, in cooperation with several other Federal agencies, including the DOE, developed the Federal Radiological Emergency Response Plan. This document was released as an interim document in 1984; in 1985, after receiving the concurrence of the above-listed agencies, it was released as its final operational plan (FEMA, 1985).

The Federal Radiological Emergency Response Plan (FRERP) assigned to the FEMA the responsibility of coordinating overall Federal assistance for radiological-emergency preparedness. The DOE was assigned the specific responsibility of providing Federal assistance for radiological monitoring and accident assessment. To facilitate this task, the DOE developed the Federal Radiological Monitoring and Assessment Plan (FRMAP). Under the FRMAP, the DOE has the primary responsibility (if assistance is requested by State or local governments) to provide technical personnel and equipment for radiation monitoring and assessment for any radiological emergency including a transportation accident involving radioactive waste. The DOE resources that are available for emergency response are discussed in Subsection C.2.3.2.

The FRERP recognizes that a transportation accident involving radioactive waste may represent a lesser hazard or serious threat to the public than other radioactive material accident scenarios, such as reactors, weapons, etc., and States that "in most cases, State resources or a limited Federal Response will suffice." In accordance with the practice established under the Civil Defense Act of 1950, the plan makes two basic assumptions about the role of the Federal Government in responding to radiological emergencies:

- State and local governments are responsible for protecting the health and safety of their citizens.

- An agency of the Federal Government will respond only if requested by the State, except in situations where the Federal agency has statutory or other authority. The availability of Federal resources is subject to prior statutory commitments to fulfill other operational requirements.

In order to assist State, Tribal, and local governments in planning emergency preparedness for transportation accidents involving radioactive material and to coordinate this Federal assistance, the FEMA promulgated regulations as Title 44 to the Code of Federal Regulations, Part 351 (44 CFR Part 351). In these regulations the FEMA assigned to various Federal agencies responsibilities for assisting State, Tribal, and local governments in planning for radiological emergencies. To this end it created the Federal Radiological Preparedness Coordinating Committee and 10 separate Regional Assistance Committees.

The Federal Radiological Preparedness Coordinating Committee (FRPCC) is composed of nine Federal agencies:

- Federal Emergency Management Agency
- U.S. Department of Energy
- U.S. Department of Commerce
- U.S. Department of Defense
- U.S. Department of Health and Human Services
- U.S. Department of the Interior
- U.S. Department of Transportation
- U.S. Environmental Protection Agency
- U.S. Nuclear Regulatory Commission.

The Federal Radiological Preparedness Coordinating Committee provides the FEMA with policy direction for the program of Federal assistance to State and local governments in their planning and preparedness activities for radiological emergencies. The Committee has established several subcommittees, one of which is the Subcommittee on Transportation Accidents. The DOE is one of the agencies represented on this subcommittee.

While the Federal Radiological Preparedness Coordinating Committee has coordination responsibilities at the national level, the Regional Assistance Committees provide coordinated Federal assistance directly to State and local governments. In general, the agencies involved in the FEMA are also involved in the Regional Assistance Committees. The committees have been given the responsibility of assisting State and local government officials in developing and reviewing their radiological emergency plans and in observing exercises to evaluate the adequacy of the plans. On specific requests from State and local governments, Federal assistance is provided, to the extent that resources permit, through the integrated efforts of the Regional Assistance Committees. The DOE has been active in all 10 Regional Assistance Committees, primarily in the area of radiation monitoring and assessment.

C.2.3.2 **The Emergency-Response Resources of the DOE**

The DOE has a wide variety of resources available for response to radiological emergencies; these resources are briefly described in this subsection. A more comprehensive discussion of these resources can be found in a recently published report (DOE, 1989b). In addition, this subsection discusses the various levels at which the DOE can provide assistance in emergency response and a typical sequence for DOE response to a transportation emergency.

The DOE organizations providing emergency radiological assistance are guided by the Regional Radiological Assistance Plan (see Subsection C.2.3.2.1) and the Federal Radiological Monitoring and Assessment Plan (see Subsection C.2.3.1).

C.2.3.2.1 Radiological Assistance Program. The DOE maintains an active emergency-response program through its Radiological Assistance Program, which is implemented through eight Regional Coordinating Offices in various parts of the United States (see Figure C.2.1). These offices, supported as necessary by other DOE offices, DOE contractors, and Federal agencies in their regions, have the capability to respond to transportation and nontransportation radiological emergencies. They usually respond directly to incidents involving materials (e.g., TRU waste) owned by the DOE or its contractors, and they will respond to requests for assistance from State, Tribal, or local governments. The guidelines for providing assistance under the Radiological Assistance Program are given in a Regional Radiological Assistance Plan. When a DOE Regional Coordinating office responds to a request for assistance, the authority of State and local jurisdictions as on-scene directors prevails, except in cases involving nuclear weapons.

Each Regional Coordinating office maintains a 24-hour per day point-of-contact, where calls for assistance are received.

C.2.3.2.2 Levels of Emergency Response. A DOE response to a request for radiological assistance will vary, depending on the incident. As discussed below, it can be as simple as advice by telephone or a full Federal response. Unless the Federal Radiological Monitoring and Assessment Plan (FRMAP) is activated, all forms of response are conducted under the DOE's Regional Radiological Assistance Plan. Transportation emergencies, however, are not likely to be serious enough to activate the FRMAP.

For minor incidents, the DOE's response may be limited to advice given by telephone. The point-of-contact at the Regional Coordinating Office in whose area the accident occurred requests from the party reporting the accident essential information, including a telephone number where the first responders can be reached and a description of the accident. This information is then provided to a designated health physicist. The health physicist then calls the first responders at the scene of the accident and provides all advice necessary for mitigation, including recommendations to expand the response, if necessary. The Regional Coordinating Office also coordinates an exchange of information with the appropriate State and Tribal agency or agencies.

When the caller asks for assistance in radiological monitoring or assessment, the Regional Coordinating Office coordinates with and receives approval from the State or Indian Tribe prior to dispatching a Radiological Assistance Program (RAP) team to the scene of the accident. This team consists of specialized personnel, such as health physicists, industrial hygienists, and medical specialists, chosen from DOE and contractor personnel. The size and composition of the team will depend on the severity of the accident.

The mission of the team is to help State, Tribal, and local authorities identify and mitigate the radiological effects of the accident. Specific activities include identifying vehicles or property that is contaminated with radioactive materials, providing advice on decontamination, and arranging for medical advice on the treatment of personal injuries that may be complicated by exposure to radiation and/or contaminated with radioactive material. A designated spokesperson of the RAP team also coordinates with the local or State authorities to provide prompt information to the public about DOE shipments and the DOE's response assistance.

In the event of a major emergency requiring response by several Federal agencies, the FRMAP is activated, and the activities of the RAP team are incorporated into the general Federal response. In such an event, the DOE's management and staff would initiate and maintain effective coordination of their radiological monitoring and assessment efforts with State and local agencies and Tribal governments. The DOE would provide all necessary resources to fully integrate Federal activities with the response efforts of the State, Tribal, and local authorities. It should be noted, however, that an emergency of such severity is not likely in transportation accidents involving radioactive materials.

C.2.3.2.3 Sequence of Events in an Emergency Response. The basic activities of a DOE Regional Coordinating Office in response to a transportation accident are likely to proceed in the sequence given below. However, because each Regional Coordinating Office has its own response plans and procedures, some variations may occur.

1) The Regional Coordinating Office receives a call for assistance.

2) The appropriate State, Tribal, or local authorities are immediately notified to verify the request.

3) A health physicist may give advice over the telephone and determine the proper level of response.

4) If the emergency requires emergency-response personnel or equipment, the Regional Coordinating Office will contact State, Tribal, and local authorities to determine their capabilities. If the State, Tribal, or local resources are adequate, the participation of the DOE is terminated unless additional assistance is

specifically requested. However, if the DOE is the owner, shipper, or receiver of the shipment, the Regional Coordinating Office will respond automatically.

5) The Regional Coordinating Office notifies the Emergency Operations Center at DOE Headquarters in Washington, D.C., about the incident and the resources requested. If the Office needs additional support, such as the Atmospheric Release Advisory Capability, it will request DOE Headquarters to facilitate that request.

6) On arriving at the scene of the accident, the RAP team assesses the situation to determine whether additional assistance is needed. If an emergency requires additional resources, the leader of the RAP team contacts the Regional Coordinating Office, which requests the Emergency Operations Center in Washington to activate additional DOE resources. If no other assistance is required, the leader of the RAP team ensures that the response proceeds appropriately until it is terminated.

7) In the unlikely event that the resources needed for radiological monitoring assessment exceed those of the DOE, the Federal Radiological Monitoring and Assessment Plan will be activated. When this happens, the manager of the DOE's Nevada Operations Office, (responsible for managing DOE resources during responses to major radiological emergencies), will select a director to coordinate monitoring and assessment assistance and to establish the liaison with the cognizant Federal agency (the shipper or owner) and State, Tribal, and local officials.

8) The appointed director selects a site near the incident to establish a Federal Radiological Monitoring and Assessment Center. The appropriate procedures from the Federal Radiological Monitoring and Assessment Plan are then executed until the emergency phase of the accident is over.

9) Once the initial emergency is over, the EPA assumes the DOE's duties of radiological monitoring and assessment. The time for this transfer will be determined by consultation among the DOE, the EPA, and the State or Indian Tribe. The EPA designates who assumes the DOE's responsibilities.

C.2.3.2.4 Resources Available to Regional Coordinating Offices. Each of the Regional Coordinating Offices has a wide range of resources for responding to a transportation accident involving radioactive materials, including both personnel and equipment. These resources are drawn from the staffs and facilities of the DOE and the DOE contractors.

The equipment available at most of the Offices includes the following:

1) Radiation monitors
   a. Alpha detectors
   b. Beta and gamma detectors
   c. Neutron detectors
   d. Tritium detectors

2) Whole-body dosimeters

3) Spectrometers (instruments capable of identifying specific radioisotopes)

4) Sampling equipment
   a. Air-sampling equipment for particulates and gases
   b. Environmental sampling equipment (plastic bags, etc.)

5) Decontamination equipment

6) Aerial-survey instruments

7) Protective clothing
   a. Gloves, boots, etc.
   b. Anticontamination clothing
   c. Breathing apparatus, including respirators and self-contained breathing apparatus

8) Dedicated response vehicles

9) Mobile laboratories

10) Electric power generators

11) Communications equipment (RAP radio frequencies).

The personnel available for response include experts in health physics, medicine, security, legal counsel, public information, and industrial hygiene.

C.2.3.2.5 Other DOE Resources. In responding to a major radiological emergency, the Regional Coordinating Offices can request assistance from various other DOE resources. The magnitude of resources available is extensive. However, for scenarios considered credible for transportation accidents, only a portion of the DOE's full cadre of resources would be called upon. These resources, which are described in more detail in the above-cited report on the DOE's emergency preparedness (DOE, 1989b), include the following:

- Atmospheric Release Advisory Capability. This resource is operated by the Lawrence Livermore National Laboratory in Livermore, California. It provides estimates, using computer modeling techniques, of atmospheric diffusion, deposition of radioactive material on the ground, and radiation doses.

- Aerial Measurement System. This system, based in Las Vegas, Nevada and Washington, D.C., consists of airplanes and helicopters with extensive equipment for radiation detection, data management, location mapping, and photography. It can be used for aerial monitoring to determine the extent of lost or diverted radioactive materials.

- Mobile Accident Response Group. This unit consists of two trucks and two trailers designed to support a military response and can be transported by U.S. Air Force C-141 aircraft. One of the trailers is a personnel-decontamination unit equipped with a shower, sink, a 30-gallon hot-water tank, and anticontamination equipment and supplies, while the trucks carry an electric generator, a 250-gallon water tank, and a workshop.

- Mobile Manipulator. The mobile manipulator is used as an emergency or standby system for toxic or radioactive environments. It is attached to a control console and can operate at a distance of up to 700 feet from the console. The mechanical hand on the manipulator can lift up to 160 pounds and drag up to 500 pounds. Two television cameras mounted behind the arm transmit pictures to monitors on the control console. This equipment is located at the Oak Ridge National Laboratory in Tennessee.

- Radiation Emergency Assistance Center/Training Site. This facility in Oak Ridge, Tennessee provides the most modern multipurpose facilities available for handling victims of radiological emergencies and is designed to handle any type of incident involving exposure to radiation (see Subsection C.3.4.2).

C.2.3.2.6 The TRANSCOM Vehicle-Tracking and Communication System. As described in Appendices D and M, a satellite-based communications system will be used to track vehicles carrying TRU waste. Based in Oak Ridge, Tennessee, it has several features that can be useful during a transportation emergency. For example, the monitoring screens at the TRANSCOM Control Center will indicate the occurrence of an accident to an operator who is on duty 24 hours a day, 7 days a week. In addition, the system can be used to obtain information about the type of radioactive material carried in a shipment, it provides information from the Emergency Response Guidebook (DOT, 1987), and it provides a means for communication between the drivers of the vehicle involved in the accident and the Central Coordination Center at the WIPP.

C.2.3.3 **Guidance to State, Tribal, and Local Governments for Emergency Response to Transportation Accidents**

The Subcommittee on Transportation Accidents (Subsection C.2.3.1), of which the DOE is a member, has been charged with coordinating activities associated with transportation accidents involving radioactive materials. One of the major activities of this subcommittee has been to prepare emergency planning guidance for State, Tribal, and local governments so that they may safely and appropriately respond to a transportation accident involving radioactive material. The subcommittee has coordinated the development of a document entitled Guidance for Developing State and Local Radiological Emergency Response Plans and Preparedness for Transportation Accidents (FEMA, 1988). This document, which is referred to as FEMA Rep-5, was initially released in 1983 and was revised in 1988. In addition to general information on transportation systems and casks, the document provides planning objectives and guidance.

Included in the revised document is guidance for ensuring that State, Tribal, and local organizations have established procedures for contacting the proper emergency-response personnel, establishing methods for communicating to the general public when an accident occurs, ensuring the availability of means for limiting radiation exposures, making arrangements for medical services, providing for clean-up after the accident, and training. The document also describes the FEMA program for assisting States, Tribal, and local governments in their planning if they request assistance.

C.2.3.4 **Federal Emergency-Response Training**

Training in emergency response is offered by several Federal agencies, including the FEMA, the DOT, the EPA, and the DOE. Information on the training courses that are available is given in the Digest of Federal Training in Hazardous Materials (FEMA-134, Washington, D.C., July 1987), which includes a summary of Federal training courses for emergency response to accidents involving radioactive materials. (The digest can be obtained from the FEMA Publications Office, 500 C Street S.W., Washington, D.C. 20472.)

The FEMA operates the National Emergency Training Center in Emmitsburg, Maryland. Training courses are offered at this center by the Emergency Management Institute. They address such topics as the assessment of radiological accidents, planning for radiological emergency-response teams. Information on the Emergency Management Institute and a schedule of courses can be obtained by writing to the FEMA National Emergency Training Center, Emmitsburg, MD 20727.

In addition, the FEMA sponsors a radiological-emergency-response course at the Nevada Test Site. This course consists of 8-1/2 days of instruction on such topics as accident assessment and procedures for response. This course is targeted for individuals in State governments who must respond to radiological emergencies, including those initiated by transportation scenarios.

The DOT supports the Transportation Safety Institute in Oklahoma City, Oklahoma. In addition, the DOT has recently published and distributed the 1987 Emergency Response Guidebook: Guidebook for Hazardous Material Incidents (DOT/P-5800.4, Washington, D.C., 1987). The guidebook contains an inventory of hazardous materials, including radioactive materials, and a series of 76 one-page guides listing potential hazards and recommended emergency actions. It is intended to be carried, for immediate use, in every emergency-service vehicle (fire, police, first aid, civil defense) in the United States. Copies can be obtained by writing to the U.S. Department of Transportation, Research, and Special Programs Administration, Attention: DHM-51, Washington, D.C. 20590.

The DOE has created the Radiation Emergency Assistance Center/Training Site (REAC/TS) at Oak Ridge, Tennessee. This multipurpose facility, operated by the Oak Ridge Associated Universities, is designed to treat victims of radiological accidents and to train medical and health-physics personnel. It is designed to handle any type of radiation-exposure accident that might occur at Oak Ridge or elsewhere (see Subsection C.3.4.2).

The DOE's Transportation Management Division sponsors a series of workshops on radiation-related emergency response. These one-day introductory courses cover basic emergency-response issues related to hazardous materials transportation incidents, with emphasis on accidents. Designed for regulatory and enforcement personnel as well as first responders to transportation incidents, the workshops cover four major topics: hazardous materials in general radioactive materials, shipments of radioactive materials, and response to incidents involving radioactive materials.

The DOE has also instituted a special training program for the transportation of TRU waste to the WIPP. This program is discussed in Subsection C.3.4.

C.2.3.5 **Federal Information Services for Radiological Emergencies**

The DOE operates, in conjunction with the Defense Nuclear Agency, the Joint Nuclear

Accident Coordinating Center (JNACC). The purpose of the JNACC, which is headquartered at the Kirtland Air Force Base in Albuquerque, New Mexico, is to exchange and maintain information related to radiological-assistance capabilities within Federal government agencies and the military. The JNACC also functions as a point of coordination for requesting military assistance in connection with radiological accidents.

The DOE also has eight regional centers of emergency-response experts to provide information and assist in responding to accidents. The teams are located in Upton, New York; Oak Ridge, Tennessee; Aiken, South Carolina; Albuquerque, New Mexico; Argonne, Illinois; Idaho Falls, Idaho; Oakland, California; and Richland, Washington.

Information is also available from the National Response Center in Washington, D.C. This center is maintained by the DOT through the Coast Guard and in cooperation with the EPA. It provides information and advice to all interested parties for meeting emergencies involving spills of hazardous substances, including radioactive materials. The Chemical Manufactures Association maintains CHEMTREC, a similar information resource, also located in Washington, D.C. Both the National Response Center and CHEMTREC can be accessed using a toll free 800 telephone number, 24 hours per day.

C.2.3.6 **Financial Responsibility for Transportation Accidents**

To provide a high level of financial protection for the public in the event of a nuclear incident, Congress enacted the Price-Anderson Act, 42 USC 2014 and 2210 (Act). The Act provides a system of financial protection for public liability for a nuclear incident or a precautionary evacuation arising out of or in connection with DOE contractor activity by providing Government indemnity to pay claims up to approximately $7.3 billion per incident. (Certain NRC-licensed activities are also covered by the Price-Anderson system through insurance and a pooling of utility funds, but those provisions are not applicable to the WIPP.)

In the event that claims exceed the statutory dollar limit, the President is required to submit a compensation plan to the Congress providing for prompt and full compensation for all valid claims, and Congress has promised to "take whatever action is determined to be necessary (including approval of appropriate compensation plans and appropriation of funds) to provide full and prompt compensation to the public for all public liability claims resulting from a disaster of such magnitude" (42 USC 2210 [e]).

Price-Anderson coverage applies to all DOE fixed facilities shipping waste to the WIPP, the WIPP itself, and transportation to or from these covered facilities. All transportation modes are covered, and the protection applies not only to the named party in the indemnity agreement, but to any person (except DOE and NRC) who may be liable for public liability.

In addition to the Price-Anderson coverage, all motor vehicles carrying TRU waste to the WIPP are required by the Motor Carrier Act of 1980, 42 USC 10927, and implementing regulations, 49 CFR 387, to maintain financial responsibility of at least $5 million, which would be available to cover public liability from a non-nuclear incident and for environmental restoration.

C.3 **EMERGENCY-RESPONSE PLAN FOR WASTE TRANSPORTATION TO THE WIPP**

This subsection specifically addresses emergency preparedness for accidents occurring during the transportation of TRU waste to the WIPP. It outlines the general responsibilities, illustrates the responses that might be expected by describing a hypothetical accident scenario, and then gives detailed procedures to be followed by the various cognizant organizations or persons.

In transportation accidents involving shipments of TRU waste, the responsibilities will be as follows:

1) The carrier will be responsible for notifying designated authorities of the accident (see Subsection C.3.1).

2) State, Tribal, and local authorities will be the first responders at the scene of the accident. They will have command and control authority for emergency response, and they will be responsible for implementing measures necessary to protect life, property, and the environment.

3) The DOE, as owner and shipper, will be present at the scene to assess the damage, to verify the level of any release of radioactive material or that no release of radioactive material has occurred, and to help the State and local authorities promptly inform the public about the situation. In the unlikely event that a release of radioactive material has occurred, the DOE or its contractors will collect the TRU waste and any debris; decontaminate soil, vehicles, and persons as needed; reload the TRU waste into new shipping containers; and return the site of the accident to normal use.

These responsibilities are illustrated in Figure C.3.1 and outlined in the sections that follow, which discuss the procedures to be followed by the carrier; State, Tribal, and local governments; and the DOE.

Each of the responsible parties must make various notifications of the accident. The organizations to be called by each party are cited in the text that follows, and the notifications that are to be made are summed up in Figure C.3.2.

C.3.1 **EMERGENCY-RESPONSE PROCEDURES FOR THE CARRIER**

The trucking contractor (the carrier) for the WIPP has prepared an emergency-response plan, including an itemized list of the emergency equipment carried on the vehicle, and has submitted it to the DOE for approval. The trucking contractor has provided the tractors transporting the TRU waste with equipment to be used in the event of a

Figure C.3.1

Activities performed by State, Tribal, and local authorities and the DOE in response to a transportation emergency involving TRU waste.

Figure C.3.2

Typical notifications that would be made after a transportation accident involving a shipment to the WIPP.

transportation accident. This equipment includes a citizens' band radio, a mobile telephone, an antenna for the TRANSCOM satellite-based vehicle tracking system, and instruments for detecting and measuring alpha, beta, and gamma radiation. The drivers of the tractor-trailers are to receive training in radioactive waste transportation and emergency response, including procedures for obtaining local, State, or Federal assistance, if technical advice or emergency assistance is needed. (As explained in Appendix M, two drivers will be used for each shipment in order to provide constant surveillance of the tractor-trailer at all times.) The drivers will be trained in the use of the radiation survey meters. They will be supplied with complete procedures for responding to the accident, including the telephone numbers of the Central Coordination Center at the WIPP, the cognizant State or Tribal agencies, and the telephone numbers of the DOE's Regional Coordinating Offices where the Radiological Assistance Program teams are located (Figure C.2.1). The drivers will be given telephone numbers that can be called collect if the mobile telephone does not operate. For communication with the dispatcher of the trucking contractor, the drivers will be given 800-numbers.

C.3.1.1 **Procedures for the Drivers of the Vehicles**

If a transportation accident occurs, the drivers of the vehicle will take the following actions, in addition to the usual actions (e.g., extinguishing fires, placing caution devices on the road) necessary to control an accident situation:

1) Isolate the immediate area around the vehicle.

2) Prevent unauthorized personnel from entering the affected area.

3) Notify local authorities.

4) If there is a possibility that one of the shipping containers has been breached and radioactive materials have been released, the drivers will perform a preliminary radiation survey with the radiation monitoring instruments provided in the cab of the tractor.

5) Notify the WIPP Central Coordination Center and report as much of the following information as is available at the time:

   a. Date, time, and location of accident
   b. Severity of accident
   c. Telephone number where the drivers can be reached
   d. Shipment number and description of waste from shipping papers
   e. Extent of property damage and/or personnel injuries
   f. Results of the radiation survey made by the drivers
   g. The authorities in charge at the scene
   h. The civil agencies that have been notified
   i. What assistance is required.

   If all of the information listed above is not known, the drivers must not delay calling. To facilitate this reporting, the driver will be provided with a form that lists all of the items to be reported. This form should be filled out before the trip is completed.

6) The drivers are to identify the persons who might have been in the immediate area of the vehicle to the on-scene commander. If an on-scene commander is not

present, request the persons who have been identified to remain at the scene.

7) Stand by until assistance arrives.

8) Notify the dispatcher of the trucking contractor.

9) Follow any site-specific instructions that have been given to the drivers by the dispatcher.

10) Notify TRANSCOM Operator (as shown on Figure C.3.2).

While the above-listed activities are performed, constant surveillance of the tractor-trailer must be provided by one of the drivers. The drivers are not to move any vehicles, containers, or wreckage unless directed to do so by the on-scene commander, or unless it is in the interest of public health and safety. Before moving vehicles, containers, or wreckage, the drivers must obtain permission from WIPP Transportation Operations or the cognizant DOE regional office of the Radiological Assistance Program. The drivers must obtain the name of the person or persons approving the movement.

In addition, the drivers may not remove any seals from the shipping containers. And unless they have the specific approval of the WIPP Transportation Operations, the drivers shall not permit the removal of seals by anyone other than the authorized WIPP representative.

C.3.1.2 **Procedures for the Dispatcher of the Trucking Contractor**

The dispatcher of the trucking contractor will take the following actions:

1) In conjunction with the WIPP Central Coordination Center, notify the following, in order of priority:

   a. The WIPP Project Office
   b. The DOE Albuquerque Operations Office
   c. Appropriate State, Tribal, and local law-enforcement agencies
   d. Generator facility.

2) In the event of breakage of the shipping containers, spillage of TRU waste, or suspected contamination with radioactive material, notify the DOT.

3) Have the vehicle repaired or dispatch a replacement tractor.

4) Send replacement drivers, if necessary.

5) Authorize the shipment of replacement parts, if necessary.

6) Maintain a log of actions taken during the emergency, including the time of each action, and send a copy of the record to the WIPP.

C.3.1.3 **Insurance**

The trucking contractor will be responsible for maintaining up to $5 million liability insurance for nonradiation-related property damage, injury, or death. Radiation-related liabilities will be covered by the Federal Government under the provisions of the Price-Anderson Amendment Act (see Subsection C.2.3.6).

C.3.2 **EMERGENCY-RESPONSE PROCEDURES FOR THE STATE, TRIBAL, AND LOCAL GOVERNMENTS**

As explained in Subsection C.2, State, Tribal, and local governments have primary responsibility for implementing measures at the scene of the accident to protect life, property, and the environment. These measures may include such activities as extinguishing fires, excluding people from the scene of the accident, giving first aid to the injured, and evacuating the nearby residents. The same responsibility applies to the governments of Indian Tribes having response capabilities in the case of emergencies on Indian reservations.

The DOE has developed a program for training police and emergency-response personnel of State, Tribal, and local governments in the proper procedures to be followed in the event of a transportation accident. The training course includes an 8-hour course for personnel selected by the States to be the first responders. The personnel who were trained first were 2,417 firemen, policemen, and emergency medical personnel from the States involved in shipments from the Idaho National Engineering Laboratory and the Rocky Flats Plant; that is, Idaho, Utah, Wyoming, Colorado, and New Mexico. Personnel from the other States will be trained before any TRU waste is transported through their State. The training course is described in detail in Subsection C.3.4.

C.3.3 **PROCEDURES FOR RESPONSES BY THE DOE AND ITS CONTRACTORS**

C.3.3.1 **Procedures for the Central Coordination Center at the WIPP**

The Central Coordination Center (CCC) at the WIPP will be responsible for coordinating the emergency-response actions of the DOE. This center will be linked to the Control Center of the TRANSCOM satellite-based vehicle tracking and communication system at Oak Ridge, Tennessee. (The TRANSCOM system is described in Subsection D.2.)

To increase public confidence and maintain a high level of coordination, a CCC operator will monitor incoming and outgoing shipments 24 hours per day, 7 days a week. The duties of the CCC operator will include the following:

1) Monitor the transport of the TRUPACT-II containers and the shipping casks for RH TRU waste, both loaded and empty.

2) Coordinate, as necessary, the activities of the DOE, the trucking contractor, and the drivers, in the event of breakdown or driver emergency.

3) Provide a means of emergency notification.

4) Coordinate, as necessary, with the State and local personnel who are designated first responders and with law-enforcement agencies.

5) Coordinate between the drivers and the Joint Nuclear Accident Coordinating Center for a safe haven for the shipment if necessary. (A "safe haven" is a parking

area, for example, at military installations that can be used, by agreement with the Department of Defense, for TRU waste shipments.)

6) Function as a central tracking point in the event the TRANSCOM satellite-based system does not function properly.

To facilitate CCC responses during and after a transportation accident, check sheets will be provided. The CCC operator will maintain a log of events as they occur, citing all actions taken, if appropriate.

In the event that the CCC operator is notified or becomes aware of an emergency situation, he or she will follow a prescribed procedure, using an Accident Response Checklist. An emergency situation requiring this response from the CCC operator is defined to be one of the following: a vehicle accident, a breach of a shipping container (a TRUPACT-II for CH TRU waste or a NuPac 72B for RH TRU waste), or a security problem (an attempt to impede the progress of the vehicle to the WIPP site).

The procedure is as follows:

1) The CCC operator will attempt to establish contact with the driver and gather as much information as possible about the cause of the accident.

2) In the event of an accident, the CCC operator will notify the organizations listed on the Accident Response Checklist.

C.3.3.2 **Procedures to Be Followed in the TRANSCOM Control Center**

The operator of the TRANSCOM Control Center will update or correct, as appropriate, the data bases for the list of emergency contacts and the emergency checklist. This operator is the only user that may update these data bases.

The MESSAGE option of the TRANSCOM system provides a means of communication that links the Central Coordination Center at the WIPP, the TRANSCOM Control Center in Oak Ridge, Tennessee, other selected users, and the vehicles used to transport TRU waste. Messages are assigned one of four priority categories. All messages from vehicle drivers are routed to the CCC operator at the WIPP; messages to drivers are sent by the operator of the TRANSCOM Control Center or the CCC operator.

Priority 1 messages are information only and do not require responses. Priority 2 messages signify minor problems and must be acknowledged in 5 minutes. All messages from vehicle drivers will be automatically assigned a priority ranking of 3. Such a message must be read and acknowledged within 2 minutes, or an alarm will be generated at the TRANSCOM Control Center.

Priority 4 will be reserved for emergency messages. If such a message is not acknowledged within 1 minute or if the addressee is not logged onto the system, an alarm will sound at the TRANSCOM Control Center. In such a case, the TRANSCOM operator will attempt to contact the CCC Operator at the WIPP. If necessary, the message will be routed to a back-up WIPP computer.

C.3.3.3 **Emergency-Response Responsibilities of Other DOE and DOE-Contractor**

Organizations

This subsection reviews the emergency-response responsibilities of the WIPP Transportation Operations, the DOE's Albuquerque Operations Office, and the WIPP Project Office.

Transportation Operations is a group in the Waste Isolation Division of the Westinghouse Electric Corporation (WEC). WEC is the operations contractor for the WIPP, and it is responsible for ensuring that the transportation of TRU waste to the WIPP is safe, cost-effective, and legal. WEC provides maintenance for the shipping containers for TRU waste, and ensures that WIPP transportation activities are properly documented.

Transportation Operations personnel must demonstrate an understanding and knowledge of emergency-response procedures. The qualification program for these personnel includes formal training, on-the-job training and retraining, performance checklists, and written and oral examinations. Specific emergency-response topics covered in the examinations include fire, nuclear criticality, evacuation, and the use of radiation-dose meters in accidents involving radioactive materials.

The specific emergency-response responsibility of Transportation Operations personnel is the timely notification of WIPP management of accidents or incidents involving TRU waste shipments to the WIPP. When an accident occurs, Transportation Operations will receive information on the details of the accident from the CCC operator. Transportation Operations will notify the DOE's Albuquerque Operations Office, and the DOE Operations Manager will permit the carrier to remove the shipment from the scene of the accident, if necessary. This decision will be relayed to Transportation Operations, who will notify the carrier (driver) through the CCC operator and the TRANSCOM system. The traffic manager of the shipping site and the WIPP Transportation Operations will decide whether the shipment should proceed to the WIPP or return to the point of origin. This decision will be relayed to the carrier (driver) by WIPP Transportation Operations through the CCC operator and the TRANSCOM system.

The DOE's Albuquerque Operations Office (DOE/AL) will be responsible for notifying the Radiological Assistance Teams of the Radiological Assistance Program (see Subsection C.2.3) if their assistance is needed. DOE/AL will identify the Regional Office of the Radiological Assistance Program that is closest to the scene of the accident (there are eight regional offices) and notify it (through the established DOE Headquarters notification system) that a Radiological Assistance Team should be dispatched. In the event that the accident is a Type A accident as defined in DOE Order 5484.1, "Environmental Protection, Safety, and Health Protection Information Reporting Requirements"), DOE/AL will notify DOE Headquarters.

DOE/AL and the DOE's WIPP Project Office will coordinate the deployment of assistance resources for the accident. These resources may include a public information officer, Radiological Assistance Teams for making radiation surveys, and other technical and management personnel, as may be required by the conditions of the accident to support the on-scene command and control maintained by the State, Tribal, and local agencies involved in the response.

C.3.4 EMERGENCY-RESPONSE TRAINING

C.3.4.1 Introduction

In late 1987, the State of New Mexico agreed to provide training for responding to WIPP-related emergencies to the States traversed by WIPP transportation routes. This led to the creation of the States Training and Education Program (STEP). As a result, the TRU System Integration and Transportation office of DOE's WIPP Program office, developed and conducted ER training to transport-corridor States and Indian Tribes. The purposes of this training are 1) to provide accurate information regarding the WIPP in order to enhance hazardous material response capabilities along the transport corridor routes, 2) to provide specific response protocols to responders along TRU waste routes, 3) to provide States and local jurisdictions with the framework to build radiological materials response programs, 4) to provide responders with the skills necessary to assess impacts of an accident involving a WIPP shipment, and 5) to provide States and Tribes with independent response capabilities.

Five separate training programs have been developed for the first responders to enable them to respond to a maximum credible emergency involving a WIPP shipment.

- First Responder Course. A 1-day, 8-hour class that provides an overview of the WIPP and basic radiation and radiation protection principles. This course is intended to train fire, law enforcement, and emergency medical personnel to ascertain accident severity before a command center can be established. These courses are available to local responders at approximately 60-mile intervals along transportation routes. As of November 30, 1989, this course was offered 123 times, and attended by approximately 3,500 personnel in the States of Idaho, Wyoming, Utah, Colorado, New Mexico, Texas, Louisiana, Mississippi, Alabama, Georgia and South Carolina.

- First Responder Refresher Course. This is a 4-hour course offered to those personnel in the States of Idaho, Utah, Wyoming and New Mexico who have attended the First Responders course over 1 year ago. This course presents updated information and reviews radiological protection techniques and health effects and response protocol. As of November 30, 1989, this course has been offered in 19 different locations.

- Command and Control Course. This is a 2-day course intended for individuals who may be in command at the scene of a transportation accident involving TRU waste. In most cases these are law-enforcement or firefighting officers. In either case the DOE works with State training contacts to identify those organizations assigned this responsibility either in a written plan/procedure or by legislation. State, Tribal, and local authorities are responsible for identifying and inviting those individuals who have command and control responsibility. As of November 30, 1989, this course was offered 35 times, and 998 people were trained in the States of Idaho, Wyoming, Utah, Colorado, New Mexico, Texas, and Alabama.

This course discusses the same topics as the First Responders course, but in much greater detail. The incident command system is used to explain the roles of each response organization in mitigating overall impacts of accidents. Topics covered include scene and crowd control, fire fighting practices, medical and rescue protocols, equipment necessary to respond to a TRUPACT or RH cask accident, activities of radiological monitoring teams, the use of the TRANSCOM satellite tracking system for obtaining specific information about WIPP shipments, and media interaction techniques. The course stresses that use of protective equipment normally carried to any accident and the application of techniques taught in class will be sufficient to protect responders. Personnel are instructed in basic radiological protection principles to assist in decision making. The scope of this instruction does not include the use of radiation monitoring and detection equipment.

Personnel being trained are provided handout materials to supplement the learning experiments. Additional teaching aids include videotapes and scale models of the TRUPACT and the 55-gal drum packaging. Table-top exercises using 4 ft by 6 ft models of rural and urban environments are included to challenge the personnel and ascertain their ability to respond correctly to postulated accidents.

- Mitigation Course. This is a 4-hour course intended for State radiological health and environmental professionals who may perform radiological monitoring, make protective action decisions or perform environmental restoration activities associated with a transportation accident involving transuranics. States are responsible for inviting class participants. Individuals able to perform activities previously described would be invited. This course is offered in one location, usually the capital city, in each State where analysis has indicated that the target audience lives. As of November 30, 1989, this course was offered 11 times and taught to 231 people in the States of Idaho, Utah, Wyoming, Colorado, New Mexico, Texas, Alabama, Georgia, and South Carolina.

The course assumes all attendees have a basic knowledge of health physics. Specific information is presented on the unique properties of transuranic elements. Specific topics include a detailed discussion of the WIPP Waste Acceptance Criteria, detection techniques for alpha emitters, decontamination procedures, and methods of reducing uptake of radioisotopes following ingestion or inhalation.

Participants are provided handout materials of visual slides used in the training to reinforce the learning experience and to be used as a reference, if required during an actual response to a TRU waste transportation accident.

- Train-the-Trainer Program. This is a 12-hour course intended for individuals currently certified or otherwise authorized to train law-enforcement, fire or emergency medical personnel within the State, Tribal, or local jurisdiction. Attendees sit in on an in-depth presentation of the First Responders course. Each section is expanded upon so that future instructors will be prepared to answer potential questions. In addition, response protocols are discussed in greater detail.

Each attendee receives a copy of the First Responders course lesson plan and sample handouts. Each organization attending the course will be provided with a set of 35mm slides for use in their own training programs. These points-of-contact will be maintained and updated information will be provided when changes have occurred.

As of November 30, 1989, this course has been offered 14 times and taught to 150 potential trainers in the States of Wyoming, Utah, Alabama, Georgia, South Carolina, Mississippi, Louisiana, and Texas.

Training in the first transportation corridor, between the Idaho National Engineering Laboratory and Rocky Flats Plant to the WIPP, was completed in October 1988. A total of 2,451 persons attended 75 courses. Refresher training in the five first corridor States (Wyoming, Idaho, Colorado, New Mexico, and Utah) started in June 1989 and was completed in November, 1989.

Training for State personnel along the Southern Transportation Corridor, between the Savannah River Site and the WIPP (South Carolina, Georgia, Alabama, Mississippi, Louisiana, and Texas) started in April 1989 and finished in October 1989; approximately 1,700 people attended 64 courses.

C.3.4.2 **Medical Response Training**

The DOE has contracted with the Radiation Emergency Assistance Center/Training Site (REAC/TS) in Oak Ridge, Tennessee, to conduct an 8-hour course entitled "Medical Management in Radiation Accidents." This 8-hour presentation is a compressed version of the 3-day course offered at the REAC/TS facility. The course is being offered along the transportation route from the Idaho National Engineering Laboratory to the WIPP and Rocky Flats Plant to the WIPP, in the States of New Mexico, Colorado, Wyoming, Utah, and Idaho. Twelve different locations, usually hospitals with trauma centers, were designated to receive training based on feedback from a State point-of-contact, usually an emergency management or radiological health representative. As of November 30, 1989, 370 people have been trained in this program. The 8-hour on-location course is a generic presentation for physicians, nurses, health/medical physicists, lab technicians, etc. about how to treat traumatized individuals who may be exposed to radiation and/or contaminated with radioactive materials. Health physicists in nearby areas are also invited to attend. The techniques presented are also applicable to TRU waste. The instructors stress that normal disease control and germ prevention techniques practiced in all hospitals are the techniques that are recommended to prevent the spread of contamination in the hospital environment. Normal surgical apparel is adequate in protecting hospital staff from contamination. In addition, instrumentation available in hospitals that use radioisotopes is also shown to be effective for responding to TRU accidents.

This course is designed to initiate further dialogue between community hospital staffs in order to prepare written response procedures and to schedule transportation scenario exercises. The REAC/TS staff also discusses the availability and use of chelating drugs used on individuals who have ingested or inhaled radioisotopes similar to those transported in WIPP shipments.

The REAC/TS staff is recognized by the DOE as the source of instruction for courses related to the handling of radiation accident cases. As part of the Oak Ridge Associated Universities, REAC/TS is accredited by the Accreditation Council of Continuing Medical Education, the American College of Emergency Physicians and the American Board of Health Physics. In addition to their training activities, REAC/TS maintains a research program on human radiation exposure, and provides 24-hour direct or consultative assistance regarding medical and health physics problems associated with radiation accidents in local, national and international incidents. REAC/TS has played an active role in medical responses for actual incidents in Goiania, Brazil (1987); Juarez, Mexico (1983); and Houston, Texas (1983).

REAC/TS is recognized by the Food and Drug Administration (FDA) as the principal investigator for two types of chelating agents that are considered to be Investigational New Drugs. These drugs are calcium and zinc DTPA (diethylenetriaminepentaacetic acid). These drugs are for use in radiation accidents where actinide contamination has occurred. Since 1951, REAC/TS has monitored approximately 3,000 doses administered to about 600 persons.

The DOE has also funded the State of New Mexico for a full-time WIPP trainer in the Environmental Improvement Division. The purpose of this individual is to further train emergency room and hospital staff in the State of New Mexico to deal with traumatized patients involved in WIPP transportation accidents. Training activities began in October 1989.

C.3.5 **ASSISTANCE TO MEDICAL FACILITIES**

C.3.5.1 **Hospital Planning and Capabilities**

All hospitals accredited by the Joint Commission on the Accreditation of Healthcare Organizations (JCAHO) must develop written emergency plans and conduct periodic disaster drills to manage the consequences of community-wide emergency situations that disrupt the hospital's ability to provide care and treatment. Emergency situations include transportation accidents involving radioactive waste and commercial aircraft disasters such as the recent crash in Sioux City, Iowa. Written guidance for these activities exists in documents such as the National Council on Radiation Protection and Measurement's (NCRP) Report 65 entitled, "Management of Persons Accidentally Contaminated with Radionuclides." The NCRP document includes specific guidance for preparing medical response, treatment, and decontamination protocols. This detailed planning is normally found in hospitals with major trauma centers. However, many rural hospitals which are based along major transportation routes or near commercial nuclear power reactors have been active at varied levels of participation. It cannot be overemphasized that the above planning activities are required of each hospital as a condition of accreditation. The certification of the hospital's readiness to respond to radiological emergencies is the responsibility of the JCAHO, not the DOE.

As part of the planning process, each accredited hospital is also responsible for maintaining the proper equipment and facilities for responding to emergencies involving radioactive materials. This includes radiation monitoring equipment. Hospitals with nuclear medicine departments normally have the equipment and staff to handle contamination incidents from internal misuse of radioisotopes, as well as contamination incidents resulting from transportation accidents involving radioactive materials. Normal disease control and germ prevention techniques are also effective in preventing the spread of contamination in a hospital situation. Normal surgical apparel is adequate in protecting hospital staff from contamination.

In the event that a traumatized individual has been exposed to radiation and/or is contaminated with radioactive material, several forms of assistance are available from the DOE. First, the REAC/TS maintains a 24-hour per day assistance telephone line regarding medical and health physics problems associated with radiation accidents. Zinc and calcium DTPA (chelating drugs) are also available from 42 different locations within the United States, 14 of which are DOE plutonium handling facilities that are in close proximity to the WIPP routes. In addition, radiation monitoring and decontamination support is available from the Radiological Assistance Program teams (previously described in Subsection C.2.3.2.1). It is not a medical standard to stockpile chelating drugs in places where plutonium exposure is a possibility. In fact, a study funded by the Department of Defense concluded that DTPA was not required at bases that stored nuclear weapons. The availability of DTPA from the DOE network was satisfactory to provide adequate medical care.

Radiological monitoring instruments, assorted decontamination supplies, and training have been provided by the DOE to the Carlsbad and Hobbs Medical Centers for the purpose of dealing with a major incident at the WIPP site.

C.3.5.2 **Specialty Drugs**

In the event that an individual ingests plutonium into his or her body, chelator drugs (i.e., Ca and Zn DTPA) are available to 42 U.S. physicians as an Investigational New Drug. Fourteen locations are in close proximity to WIPP transportation routes; most are located at DOE facilities that handle plutonium. Through the medical training provided along transportation routes, it is anticipated that the interest in maintaining an inventory of this drug will be sparked. If requested, DOE will evaluate each request for the drug and provide an inventory and training in its use.

One of the drawbacks to using chelator drugs is that the side effects are often more harmful than the preventive efforts. Decisions on administering the drug must be made by a physician who is aware of the risks to the patient balanced by the potential benefits. Oak Ridge Associated Universities has the Food and Drug Administration Investigational New Drug permit to act as principal investigator in monitoring the use of this drug.

C.3.6 **FUNDING**

The FEMA currently provides financial assistance to the States to support planning, preparedness, and response activities for a wide range of emergencies, including those related to accidents involving radioactive materials. The purpose is to assist State and local governments in the development and enhancement of emergency-management systems to cope with all types of disasters and emergencies. The funding is made available through the comprehensive cooperative agreements (CCAs) that the FEMA has entered into with each of the States. These agreements are individually reviewed and renewed every year, and State requests for funding are handled during the agreement-renewal process. Although priority for funding is given to planning for a nuclear attack, the resources provided through the CCA programs may be used to plan for response to peacetime disasters and emergencies, including transportation accidents involving radioactive materials. Such planning must be conducted in the context of emergency operating plans addressing all hazards.

Under the State-specific agreements, the following CCA programs may be funded:

CCA Program	Federal share (percent)
Emergency-management assistance (staff) salaries and administrative costs)	50
Radiation-instrument inspection, maintenance, and calibration	100
Radiation protection (generic planning and exercise)	50
Population protection planning (generic evacuation planning for all hazards)	100
Disaster-preparedness improvement ($25,000 per State)	50
Emergency-management training and education	100

Financial assistance provided for training and education may be used to support the following training and education activities:

- Emergency-response training conducted by State and local governments (up to 100 percent funding by the FEMA).
- Training at the FEMA's own training center.
- Procuring equipment necessary for State and local training courses (up to 50 percent funding by the FEMA if approved by the FEMA).

The DOE has agreed to support approved State and Tribal activities related to WIPP transportation. This funding will be administered through Cooperative Agreements with representative organizations (e.g., the Western Governors' Association will administer funding to the Western States).

There are, however, provisions in a draft piece of legislation entitled "The WIPP Land Withdrawal Act" which call for funding under certain conditions.

C.4 **EMERGENCY-RESPONSE SCENARIOS**

This section presents a scenario for a hypothetical severe transportation emergency and examples of emergency response in two accident situations that have actually occurred. The purpose is to illustrate how response proceeds in a given situation and how the various resources available for emergency response are used.

C.4.1 **SCENARIO FOR A HYPOTHETICAL SEVERE TRANSPORTATION EMERGENCY**

To provide the reader with a graphic example of emergency response and to illustrate the content of the training courses given by the DOE to the States involved in TRU-waste transportation to the WIPP, this section describes in detail a scenario for a hypothetical severe transportation emergency. An emergency as severe as that described in this scenario has a low probability of occurrence because, as described in Appendix L, the TRUPACT-II container in which the TRU waste will be transported is designed to withstand the conditions of accidents that can be expected to occur on the basis of accident experience.

C.4.2 **RADIOLOGICAL ASSISTANCE RESPONSE: BURLEY, IDAHO—OCTOBER 12, 1986**

At 5:25 p.m. on October 12, 1986, the Idaho Warning Communications Center arranged a conference call between the Idaho Department of Health and Welfare and the DOE Idaho Operations Office, Region 6 Radiological Assistance Coordinator. It was reported that a tractor-trailer containing radioactive materials had been involved in an accident with other vehicles on Interstate 84 near Burley, Idaho, and had plunged into the Snake River. The radioactive shipment was en route from the RMI Company Ashtabula Extrusion Plant, Ashtabula, Ohio, to the United Nuclear Company, Hanford Site, Richland, Washington. The IHW staff was preparing to proceed to the scene of the accident (about 130 miles from Boise, Idaho, and about 120 miles from Idaho Falls, Idaho) to assist law enforcement personnel at the scene.

A Radiological Assistance Team (RAT) of the DOE Idaho Operations was placed on alert, pending a request for services. Following a review of the accident, the five-man RAT (with eight RAT kits and special survey instruments) was dispatched to the Burley airport by helicopter. The Idaho State Police and the Cassia County Sheriff's Office provided ground transportation to the accident scene. Following an inspection of the accident scene by the RAT and a determination that the radioactive shipment posed no immediate threat to public health and safety, State and local authorities held a meeting in Twin Falls to formulate a plan of action to be implemented as soon as daylight permitted.

The Idaho National Engineering Laboratory Emergency Response Van arrived at Burley to function as a mobile command post, communications center, and health physics laboratory. A second RAT arrived by helicopter at the Burley airport to provide assistance at the accident scene. The truck driver and relief driver had sustained injuries that required their hospitalization. The tractor-trailer cargo consisted of: 1) 20 wooden packages loaded with 3-5 billets of low enrichment uranium metal weighing 250-285 pounds each, and 2) 73 empty wooden packages. Radiological surveys of the cargo, vehicle, handling personnel, and the environment by the State of Idaho health and physics personnel and the RAT demonstrated that no detectable radioactivity was released form the radioactive shipment.

Local, regional, and national news media coverage of this accident was intense. Local and State authorities at the scene requested that the DOE coordinate radio commentaries, television coverage, and newspaper articles. This action ensured that information about the accident was timely, factual, and consistent among the various reports of the accident. Timely notifications, with appropriate updates, were made throughout the response to DOE management, the DOE Headquarters Emergency Operations Center, the shipper, the receiver, State and local officials, and other officials.

By October 13, 1986, a firm in Twin Falls, Idaho, commenced salvage operations to retrieve the tractor-trailer cargo of loaded and empty containers. The last package was lifted out by crane that afternoon. The salvaged containers were placed in large water-tight containers. In addition, a structural engineer provided guidance to the RAT and the salvage operator, relative to the effects of water (river) pressure on the trailer during load recovery operations. Late on October 13, 1986, two trucks, one loaded with empty containers and the other carrying loaded containers, arrived at the Idaho National Engineering Laboratory.

On October 14, 1986, the tractor-trailer was towed from the Snake River and transported to a salvage yard in Twin Falls, Idaho. Later on October 14, 1986, following radiological surveys of the vehicle and the environment, the area was released for unrestricted use. A close-out was held at the accident scene by participants in the response. A critique of the accident response was held at the Idaho State Police Office in Twin Falls.

C.4.3 RADIOLOGICAL ASSISTANCE RESPONSE: POCATELLO, IDAHO—OCTOBER 10, 1985

On October 7, 1985, the Union Pacific Railroad Operations Division, Omaha, Nebraska, contacted the Idaho Warning Communications Center (WCC), regarding a radioactive placarded ATMX railcar observed to be leaking at the Union Pacific Rail Terminal in Pocatello, Idaho.

The WCC was provided details by the Region 6, Radiological Assistance Coordinators with a follow-up call to the shipping department at Rocky Flats Plant, Colorado, from which the shipment originated. It was confirmed that the subject car was carrying plutonium-contaminated waste. The Union Pacific Railroad formally requested a radiological assistance team (RAT).

At the site, the five personnel who had visited the railcar in question were monitored (frisked) immediately. No contamination was detected during personnel alpha surveys. Then, samples were taken in the immediate railcar area. Weather conditions prior to and during the surveys were generally windy with a mixture of snow and rain. Samples were taken of the dripping liquid and smears taken in the immediate railcar area. The smears were allowed to dry, scanned (frisked) with alpha and beta gamma instruments, and then counted. Follow-up smears were taken of the observed wide crack on the underside of the ATMX car. No contamination was detected, and it was concluded that leakage was weather-related without any radioactive release.

REFERENCES FOR APPENDIX C

DOE (U.S. Department of Energy), 1989a. Radioactive Material Transportation Emergency Response Orientation.

DOE (U.S. Department of Energy), 1989b. Emergency Preparedness for Transportation Incidents Involving Radioactive Materials, SAIC 89/1354, Science Application International Corporation, Oak Ridge, Tennessee.

DOT (U.S. Department of Transportation), September 1, 1987. Emergency Response Guidebook, DOT P 5800.4.

FEMA (U.S. Federal Emergency Management Agency), November 8, 1985. Federal Radiological Emergency Response Plan.

FEMA (U.S. Federal Emergency Management Agency), 1988. Guidance for Developing State and Local Radiological Emergency Response Plans and Preparedness for Transportation Accidents, FEMA-REP-5, Draft Revision, August, 1988.

Mitter et al. (E. L. Mitter, R. D. Hume, F. J. Vilardo, E. Feigenbaum, H. Briggs and R. Piercy), 1980. Survey of Current State Radiological Emergency Response Capabilities for Transportation Related Incidents, School of Public and Environmental Affairs, Indiana University, NUREG/CR-1620, prepared for U.S. Nuclear Regulatory Commission, Washington, D.C.

Wolff, T. A. 1984. The Transportation of Nuclear Materials, SAND84-0062, Sandia National Laboratories, Albuquerque, New Mexico.

APPENDIX D

TRANSPORTATION AND TRANSPORTATION-RELATED
RISK ASSESSMENTS

TABLE OF CONTENTS

Section | Page

LIST OF FIGURES

LIST OF TABLES

D.1 **INTRODUCTION**

This appendix provides information that supports the discussions in Subsections 3.1 and 5.2. It discusses plans for transporting TRU waste to the WIPP and the risks associated with transportation. It has been expanded and revised in response to comments on the draft SEIS. In particular, the assessment of transportation risks has been revised and expanded to include more State-specific transportation data, along with comparative risk data from independent risk models.

Since the DOE prepared the FEIS in 1980, changes have been made in the plans and systems required to transport TRU waste to the WIPP. In addition, substantially more development work has been completed on the required components, systems, and facilities for transporting TRU waste to the WIPP.

The major changes between the 1980 FEIS and this SEIS fall into four general categories.

First, where the FEIS assessed the impacts of only TRU waste shipped from the Idaho National Engineering Laboratory in Idaho and the Rocky Flats Plant near Denver, Colorado, the SEIS analyzes waste transportation from 10 facilities located across the nation. A comprehensive analysis is provided for transportation from each of these facilities.

Second, the U.S. Nuclear Regulatory Commission (NRC) has certified the design of the TRUPACT-II shipping container for CH TRU waste. The TRUPACT-II container is the result of 7 years of an intense, iterative process of design, testing, and certification. Design, testing, and certification of the RH waste shipping cask will be completed in advance of RH TRU shipments.

Third, fulfilling the intent and spirit of the law establishing the WIPP (Public Law 96-164), the DOE held substantive discussions with the State of New Mexico on a wide variety of subjects, including the transportation of TRU waste across the State. The DOE has also conducted discussions with the other States through which TRU waste will be transported.

Fourth, because of better definition and information, the volume, quantities, and characteristics of waste to be transported are more detailed than reported in the 1980 FEIS. This improved data permits a more thorough analysis of the risks associated with transporting waste (see Appendix B).

This appendix should be read in conjunction with the appendices describing the design, testing, and certification of the shipping containers and casks for TRU waste (Appendix L); emergency-response training and capabilities (Appendix C); and the management plan of the trucking contractor (Appendix M). Appendix M has been added in response to comments; it discusses trucking company safety procedures and equipment and maintenance, in addition to the qualifications and training of drivers and the routine and emergency procedures to be followed during waste shipments. When reviewed together, Appendices C, D, L, and M provide a good understanding of how the entire transportation system is organized to ensure that the shipments will be safe.

D.2 TRANSPORTING TRU WASTE TO THE WIPP

D.2.1 TRANSPORTATION MODES

D.2.1.1 Truck Transport

Although the WIPP can receive TRU waste shipments by truck or train, current plans call for all shipments during the approximate 5-year Test Phase to be made by truck. During the Test Phase, the DOE proposes to transport to and emplace in the WIPP limited quantities of waste; the specific quantities of waste emplaced would be limited to that deemed necessary to achieve the objectives of the Test Phase. For purposes of bounding the potential impacts of the Test Phase in this SEIS, the DOE assumes that up to 10 percent of the volume of TRU waste that could ultimately be permanently emplaced at the WIPP would be emplaced during the Test Phase. The actual amount of waste proposed for the Test Phase is likely to be less than that assumed for purposes of analysis in this SEIS. For purposes of bounding the impacts it is also assumed that waste would be shipped from all 10 facilities, although it is now likely that only waste from the Rocky Flats Plant and the Idaho National Engineering Laboratory would be used during the intial phases of the proposed Test Phase. The subsequent Disposal Phase is scheduled to last 20 years.

To ensure that the transportation operations proceed safely and efficiently, the DOE has developed detailed operating plans and provided various facilities for communication, including a satellite-based vehicle tracking system. This system, called TRANSCOM, is discussed in Subsection D.2.4. In addition, the DOE has awarded a contract to a commercial carrier for the truck transport of TRU waste to the WIPP. This contract, which runs for 3 years and has options for two 1-year extensions, contains numerous provisions for the safe and efficient transport of TRU waste and for response to transportation emergencies. The key provisions of the contract include, but are not limited to, the following:

- The contractor will provide tractors wholly dedicated to contract requirements and provide technically qualified and experienced drivers for the life of the contract period. Tractors are to be domiciled and maintained within 50 miles of the WIPP and will be dispatched with a DOE-owned trailer and empty shipping containers for CH TRU waste.

- The DOE will operate a transportation operations control center called the Central Coordination Center (CCC) 24 hours a day, 7 days a week. This center will maintain day-to-day contact with the contractor carrier and the drivers.

- The contractor will be required to meet Federal regulatory requirements for the transportation of radioactive and hazardous materials, including driver training in accordance with 49 CFR, as amended, and the Commercial Motor Vehicle Safety Act of 1986 and subsequent amendments, and manifesting requirements for mixed waste specified in 40 CFR.

- At facilities with high volumes of waste, the driver will drop the trailer and packaging at the loading location designated by the facility and will be provided a return loaded trailer for the shipment back to the WIPP. At facilities with low volumes of waste, the driver will drop the trailer and packaging at the loading location designated by the facility and wait for facility personnel to load the container and trailer and release it to the driver for the return trip to the WIPP.

- On reaching the WIPP, the driver will drop the trailer and the loaded containers at a designated location and return to the terminal.

- The contractor will be required to perform verifiable routine maintenance and inspections on the tractors and trailers before and after each movement.

- The DOE will be responsible for any maintenance and repairs to the shipping containers. If the containers need repair while en route, the contractor will take appropriate corrective steps after receiving approval from the Central Coordination Center.

- The contractor is required to provide a traffic manager (dispatcher) who will act as a single point of contact for the DOE's Technical Representative in dealing with the dispatching and scheduling of shipments and coordinating and resolving problems associated with shipments.

One of the provisions of the contract was the requirement that the carrier prepare a management plan. The plan has been prepared and is summarized in Appendix M.

D.2.1.2 **Rail Transport**

Since current plans call for waste transport by truck for at least 5 years, details and specifications for rail transport have yet to be completed. For example, the design of a railcar for the transportation of TRU waste has not been agreed upon by the rail companies and the DOE, and it is unknown when a certifiable shipping container would be available for use. The present design of the TRUPACT-II container may have to be modified for proper tie-down on a railcar. It may be possible that the tractor-trailer with TRUPACT-II containers could be placed on flatbed cars with only additional supports. The decisions to pursue NRC certification, design modifications, other feature modifications, and safety specification for rail transport will be made in the future as necessary.

D.2.2 **TRANSPORTATION ROUTES**

D.2.2.1 **Truck Transport**

D.2.2.1.1 Applicable Regulations - The Department of Transportation. The DOT regulations in 49 CFR 177.825 provide a routing rule for highway-route-controlled quantities of radioactive materials (WIPP shipments fall into this category). The routing rule permits States and Indian Tribes to designate routes in accordance with DOT guidelines or an equivalent routing analysis. Interstate highways must be used in the absence of a State- or Tribal-designated route, unless a deviation is necessary.

The DOT defines a "state-designated route" as a preferred route selected in accordance with the DOT "Guidelines for Selecting Preferred Highway Routes for Highway Route Controlled Quantity Shipments of Radioactive Materials," or an equivalent routing analysis that adequately considers the overall risk to the public. The designation of routes must be preceded by substantive consultation with affected local jurisdictions and with any other involved States to ensure the consideration of impacts and continuity of designated routes.

"State routing agency" means an entity (including a common agency of more than one State such as one established by Interstate compact) that is authorized to use a State legal process pursuant to 49 CFR 177.825 to impose routing requirements, enforceable by State agencies, on carriers of radioactive materials without regard to intrastate jurisdictional boundaries. This term also includes Indian Tribal authorities that have police power to regulate and enforce highway routing requirements within their lands.

The DOT regulations in 49 CFR 177.825 provide routing and training requirements for carriers of radioactive materials, which are excerpted for the reader as follows:

> (a) The carrier shall ensure that any motor vehicle which contains a radioactive material for which placarding is required is operated on routes that minimize radiological risk. The carrier shall consider available information on accident rates, transit time, population density and activities, time of day, and day of week during which transportation will occur. In performance of this requirement, the carrier shall tell the driver that the motor vehicle contains radioactive materials and shall indicate the general route to be taken. This requirement does not apply when--
>
> 1) There is only one practicable highway route available, considering operating necessity and safety, or
>
> 2) The motor vehicle is operating on a preferred highway under conditions described in paragraph (b) of this section.
>
> (b) Unless otherwise permitted by this section, a carrier and any person who operates a motor vehicle containing a package of highway route controlled quantity radioactive materials as defined in Part 173.403(l) of this subchapter shall ensure that the vehicle operates over preferred routes selected to reduce time in transit, except that an Interstate System bypass or beltway around a city shall be used when available.
>
> 1) A preferred route consists of:
>
> (i) An Interstate System highway for which an alternative route is not designated by a

State routing agency as provided in this section, and

(ii) A State-designated route selected by a State routing agency (see Part 171.8 of this subchapter) in accordance with the DOT "Guidelines for Selecting Preferred Highway Routes for Highway Route Controlled Quantity Shipments of Radioactive Materials" and amended by HM164a (May 12, 1988) as, "an equivalent routing analysis which adequately considers overall risk to the public. Designations must have been preceded by substantive consultation with affected local jurisdictions and with any other affected States to ensure consideration of all impacts and continuity of designated routes. A State designated route is not effective until written notice has been given by the State, by certified mail, return receipt requested, to, and receipt acknowledged by, the Dockets Unit (DHM-30), Research and Special Programs Administration, U.S. Department of Transportation, Washington, D.C. 20590."

2) When a deviation from a preferred route is necessary (including emergency deviation, to the extent time permits), routes shall be selected in accordance with paragraph (a) of this section. A motor vehicle may deviate from a preferred route under any of the following circumstances:

(i) Emergency conditions that would make continued use of the preferred route unsafe.

(ii) To make necessary rest, fuel, and vehicle repair stops.

(iii) To the extent necessary to pick up, deliver, or transfer a highway route controlled quantity package of radioactive materials.

(c) A carrier who operates a motor vehicle which contains a package of highway route controlled quantity radioactive materials as defined in Part 173.403(l) of this subchapter shall prepare a written route plan and supply a copy before departure to the Research and Special Programs Administration (RSPA) as well as to the motor vehicle driver and a copy to the shipper (before departure for exclusive use shipments, or otherwise within 15 working days following departure). Any variation between the route planned and routes actually used, and the reason for it, shall be reported in an amendment to the route plan delivered to the RSPA and to the shipper as soon as practicable but within 30 days following the deviation.

D.2.2.1.2 Proposed Routes. The proposed routes for transporting TRU waste to the WIPP are shown in Figure D.2.1. The various Indian Tribes along the proposed routes are shown in Figure D.2.2, and Figures D.2.3 through D.2.5 provide additional details on the routes. The route selection was based on the use of interstate highways and other criteria presented.

To ensure that road segments of concern to the State were identified, corridor States were contacted and asked to provide a qualitative assessment of hazardous road conditions that may be present along the proposed routes (Table D.2.1). The concerns about particular segments were found to be primarily related to winter driving conditions in the mountains, bridges icing up in the winter, and interchanges in urban areas. These road segments and potential problems will be noted on logs provided to the carrier. Weather conditions will be constantly monitored and drivers will be alerted to possible severe weather conditions; no shipments will be allowed during severe weather. All truck drivers will follow the DOT requirements in 49 CFR 397.7b for identifying parking areas to use in emergency situations. The DOT requirement for motor vehicles transporting hazardous waste materials other than Class A or Class B explosives is that vehicles must not be parked on or within 5 feet of the traveled portion of a public street or road except for brief periods when the necessities of operation require the vehicle to be parked and make it impracticable to park the vehicle in any other place. In addition, the DOE is investigating the use of the 50 Department of Defense facilities along the TRU waste routes for emergency parking and is working with States to identify other emergency parking facilities. The DOE welcomes any State recommendations. The following text provides additional details.

State of New Mexico. As shown in Figure D.2.3, all transportation routes converge in New Mexico and for that reason, New Mexico is addressed separately. Transportation and routing within the State have been identified in several agreements with the State of New Mexico. The most relevant of these is the "Supplemental Stipulated Agreement Resolving Certain State Off-Site Concerns Over WIPP," which was entered into by the State of New Mexico and the DOE in December 1982.

Based on a decision made in September 1989, the State of New Mexico will hold public hearings and initiate a formal process to designate alternate routes in New Mexico for transuranic shipments to the WIPP. The formal State recommendation is expected to be complete in Spring 1990.

The specifications of the agreement recognized that movements between incoming interstates and the relatively remote WIPP would involve local highways and that, because New Mexico is the host State, these highways would see relatively concentrated service. Therefore, the DOE agreed to support the State in efforts to obtain from Congress the funds necessary to repair and upgrade various highway segments that are designated in the agreement.

Figure D.2.1
Proposed TRU Waste
Truck Transportation Routes

Figure D.2.2
Indian Tribes Along Proposed TRU Waste
Transportation Routes

Figure D.2.3
Proposed TRU Waste Truck Transportation Routes
In New Mexico

Figure D.2.4
Proposed TRU Waste Truck Transportation Routes
From Eastern DOE Facilities to the New Mexico Border

Figure D.2.4 (Continued)

Figure D.2.4 (Continued)

Figure D.2.4 (Concluded)

Figure D.2.5
Proposed TRU Waste Truck Transportation
Routes from Western DOE Facilities to the New Mexico Border

Figure D.2.5 Concluded

Two highway segments in New Mexico were identified by the DOE as potential routes in 1981, but are no longer expected to be used as State-preferred routes. However, it is likely that they would be used to carry limited shipments when circumstances (such as inclement weather) prevent transport over more direct routes as shown in Figure D.2.3. East-bound trucks on I-40 would interchange onto I-25 in Albuquerque and continue north on I-25 to the US-285 interchange, just west of Glorietta, New Mexico. They would then continue south on US-285 to the WIPP. West-bound trucks on I-40 would remain on I-40 to Clines Corners and then continue south on US-285.

Route from the Mound Laboratory, Ohio. Figure D.2.4 shows the route WIPP trucks would take from the Mound Laboratory, Ohio, to the New Mexico border. The proposed route is as follows:

Mound Avenue (W)
First Street (N), 0.5 mile
State 725 (E), 0.4 mile
I-75 (N) to I-70
I-70 (W) to I-74/465 (S) Indianapolis, Indiana
I-74/465 (S) to I-70 (W)
I-70/55 (W) to I-255 (S)
I-255 (S) to I-270 (W)
I-270 (W) to I-44 St. Louis, Missouri
I-44 (W) to I-40 Oklahoma City, Oklahoma
I-40 to US-54 (S) Santa Rosa, New Mexico
US-54 (S) to Vaughn, New Mexico
US 285 (S) to Carlsbad, New Mexico
US 62/180 (E), 29 miles
WIPP North Access Road, 13 miles

Between the Mound Laboratory and New Mexico, several highway segments of concern in Indiana have been identified; these are shown on Figure D.2.4 and further described in Table D.2.1. Major populated areas with their populations are also shown in Figure D.2.4. WIPP traffic would use the beltway around St. Louis, Missouri.

Route from the Argonne National Laboratory, Illinois. Figure D.2.4 shows the proposed WIPP transportation route from the Argonne National Laboratory south of Chicago to the New Mexico border. From the Argonne National Laboratory, trucks would take the Northgate Entrance Road (NE) for 0.25 mile to Cass Avenue and go north 0.1 mi to I-55. Once on I-55, they would continue south until they intersected with I-70, east of St. Louis.

Route from the Oak Ridge National Laboratory, Tennessee. Figure D.2.4 shows the proposed transportation route from the Oak Ridge National Laboratory, southwest of Knoxville, Tennessee, to the New Mexico border. A more detailed route description is as follows:

Bethel Valley Road (W), 1.1 miles
Tennessee State Route 95 (S), 3.3 miles
I-40 to I-240 (southern bypass) Oklahoma City, Oklahoma
I-240 to I-44 (N)
I-44 to I-40
I-40 to US-54 (S) Santa Rosa, New Mexico
US-54 (S) to US 285 Vaughn, New Mexico
US-285 (S) to US-62/180, Carlsbad, New Mexico

US-62/180 (E), 29 miles
WIPP North Access Road, 13 miles

Several hazardous road segments of concern were identified in Tennessee and Arkansas; they are shown in Figure D.2.4 and explained in Table D.2.1. WIPP traffic would use established bypasses around major cities as shown in Figure D.2.4.

Route from the Savannah River Site, South Carolina. The Savannah River Site is southwest of Columbia, South Carolina, just east of the Georgia border. The proposed TRU waste transportation route from the Savannah River Site to the WIPP follows I-20 for most of the route. Figure D.2.4 shows the proposed route with major cities, bypasses, and segments of concern. The local route from the Savannah River Site to I-20 has not yet been determined. The rest of the route can be described as follows:

I-20 (W) to I-285 (southern bypass) Atlanta, Georgia
I-285 to I-20 (W)
I-20 to I-459 (W) Birmingham Bypass
I-459 (W) to I-20 (W)
I-20 to US-285 (N), Pecos, Texas
US-285 (N) to US-62/180, Carlsbad, New Mexico
US-62/180 (E), 29 miles
WIPP North Access Road, 13 miles

Route from the Hanford Reservation, Washington. The DOE's Hanford Reservation is north of the Tri-Cities area in south-central Washington. Figure D.2.5 shows the proposed route from Hanford to the New Mexico border, including major cities and road segments of concern. The route would pass through mountainous areas of Oregon, Idaho, Utah, Wyoming, and Colorado. A brief description of the route follows:

SR-240 (S), 3.4 miles
I-182 (E), 5-10 miles
I-82 (E) to I-84 (Oregon, Idaho, Utah)
I-84 to I-80 (Utah)
I-80 to I-25 (Wyoming, Colorado, New Mexico)
NM US-285 to NM US-62/180
US-62/180 (E), 29 miles
WIPP North Access Road, 13 miles

Route from the Idaho National Engineering Laboratory, Idaho. The Idaho National Engineering Laboratory is west of Idaho Falls in southeastern Idaho. Figure D.2.5 shows the proposed highway route from Idaho National Engineering Laboratory to where it will intersect with the Hanford Reservation transportation corridor. US-26 will be used to access I-15.

Route from the Rocky Flats Plant, Colorado. The Rocky Flats Plant is between Golden and Boulder, Colorado, west of Denver. TRU waste shipments to the WIPP would follow the transportation corridor in Colorado shown for the Hanford Reservation in Figure D.2.5. Access from the plant to I-25 would be as follows:

Exit RFP by State Highway 93 (N) to State Highway 128
State Highway 128 (E) to US Highway 36
US Highway 36 (S) to I-25

Route from the Los Alamos National Laboratory, New Mexico. The Los Alamos National Laboratory is shown in Figure D.2.3. At the time of this writing, approximately one-third of a relief route to the west of Santa Fe is under construction. A second bypass, known as the Los Alamos-Santa Fe Corridor, is planned for future construction, although funding commitments have not yet been made. Shipments from Los Alamos would use the relief route or the Los Alamos-Santa Fe Corridor to access I-25, which would be used to access US-285.

Route from the Lawrence Livermore National Laboratory, California. The Lawrence Livermore National Laboratory is located just west of Stockton, California. Figure D.2.5 shows the proposed route for transporting TRU waste to the New Mexico border. The State of California is in the process of evaluating additional routes in California and plans to propose an alternate route in 1990 for WIPP-related use. No TRU waste shipments are planned from the Lawrence Livermore National Laboratory during the first 5 years of the WIPP program. The following describes in more detail the proposed route:

South Exit 0.5 mile on East Avenue
Right on Vasco 3.0 miles on I-580
I-580 South 35 miles to I-5
I-5 to I-210
I-210 to I-10
I-10 to I-15
I-15 to I-40
I-40 to NM US-285
NM US-285 to NM US-62/180, Carlsbad, New Mexico
US-62/180 (E), 29 miles
WIPP North Access Road, 13 miles

Route from the Nevada Test Site, Nevada. Highway access from the Nevada Test Site is northwest of Las Vegas. TRU waste will be transported on US-95 to I-40; Figure D.2.5 shows the proposed route to the New Mexico border.

D.2.2.2 **Rail Transport**

There are no regulatory requirements related to the selection of routes to be used for rail shipment of TRU waste (or any other material). However, the Federal Railroad Administration (FRA), which is the delegated enforcement arm of the DOT, does request to be informed of any hazardous materials shipments and will provide an evaluation of the proposed rail route for the shipper. In addition, the FRA will provide regular (e.g. each 6 months) safety inspections of the route.

Six mainline rail companies have rail lines that would provide access to eight waste facilities. These are the Atchison-Topeka Santa Fe (now known as the Santa Fe Railroad), the Union Pacific (which also owns the Missouri Pacific), Mid-South, CSX Transportation, Norfolk-Southern, and Denver, Rio Grande. The two facilities that are not readily accessible by mainline railroads or that would require truck transportation to a railspur are the Nevada Test Site and Los Alamos National Laboratory. Figure D.2.6 shows the proposed rail routes and mainline companies. As noted in the figure, only the Argonne National Laboratory would be able to transport directly to the WIPP without changing rail companies during shipment; between one and five transfers would be required for transporting TRU waste from the other waste facilities.

D.2.3 **TRANSPORTATION PLANNING AND RESPONSIBILITIES**

D.2.3.1 **General**

The truck transportation system will consist of the shippers (the waste facilities), the carrier (the trucking contractor), and the receiver (the WIPP). Overall management of the transportation system will be conducted at the WIPP at the Central Coordination Center.

Transportation planning tasks such as the development of transportation strategies and plans and the implementation of TRU waste shipments will be coordinated by DOE personnel.

An overall schedule will be developed by WIPP Transportation Operations in cooperation with the TRU Waste and Integration Department of the Westinghouse Electric Corporation, the operating contractor for the WIPP. A strategy will be developed for the optimum employment of available TRUPACT-II containers. The schedule will be revised at the end of each fiscal year to reflect the current operating experience of the transportation system and updated waste projections. A midyear update will be provided. A short-range schedule reflecting a 6-week projection will be developed in close cooperation with the waste shipper traffic managers. This schedule will be developed to implement the long-term schedule.

With respect to transportation, each of the waste facilities will be responsible for the following transportation activities:

- Interacting with the WIPP and involved States on institutional issues
- Certifying TRU waste to meet the WIPP Waste Acceptance Criteria (WAC)
- Meeting the shipment schedule developed by WIPP Transportation Operations and the waste facilities

Figure D.2.6
Proposed Mainline Railroad Routes For
TRU Waste Transportation to the WIPP

- Reporting the status of the TRUPACT-II containers and NuPac 72B casks to the Central Coordination Center

- Loading TRU waste into TRUPACT-II containers and NuPac 72B casks

- Meeting DOT and RCRA shipping paper requirements

- Dispatching loaded TRUPACT-II containers and NuPac 72B casks

- Notifying the Central Coordination Center of shipments

- Following on-site emergency response procedures for TRU waste loading accidents.

The trucking contractor will be responsible for the actual physical movement of the TRUPACT-II containers and NuPac 72B casks between the waste sites and the WIPP. The contractor will provide a dedicated tractor fleet, dedicated drivers, and a dedicated manager for this contract. The responsibilities of the contractor are outlined in the summary of the management plan in Appendix M.

The DOE will be responsible for the following transportation tasks:

- Interfacing on institutional issues with other Federal, State, and local agencies in conjunction with TRU waste facilities and local DOE field offices

- Coordinating with the waste facilities

- Planning TRU waste transportation

- Translating DOE policies into operating procedures

- Establishing and operating the Central Coordination Center

- Administering the contract of the trucking contractor

- Budgeting transportation operations

- Procuring transport packaging and trailers with placard holders

- Scheduling shipments in coordination with the traffic managers at the waste facilities

- Receiving shipments

- Maintaining communications equipment

- Complying with procedures and reporting requirements

- Reporting routine activities and nonroutine incidents to appropriate authorities
- Monitoring and evaluating the performance of the trucking contractor.

D.2.3.2 **Preoperational Checkout**

Before shipment of TRU waste, as part of an overall integrated operations demonstration, multiple dry runs from each waste facility to the WIPP will be conducted as a part of a series of preoperational checks designed to provide experience and hands-on training for the drivers of the trucking contractor and the operations personnel of the waste facility and the WIPP. A summary of the preoperational checkout plan is provided here to describe the types of testing and training procedures used by the WIPP, the waste facilities, affected States, and the trucking contractor. The checkout will provide a review of the completeness of the facility readiness review procedures, will determine the adequacy of facility readiness, and will allow the review process to track incomplete items to closure. The checkout is designated to:

- Validate the facility's ability to load and ship a TRUPACT container
- Provide experience in using the TRANSCOM tracking and communication system
- Evaluate the responsibilities of States and Indian Tribes
- Evaluate the procedures for waste receipt and emplacement at the WIPP.

The intent of these dry runs is to incorporate as many realistic conditions and procedural checks as possible into a training exercise and to incorporate any changes into the existing procedures before actual shipment. At least two dry-run preoperational checkouts will be conducted at each facility before any actual shipments. If requested by appropriate authorities, additional dry-run preoperational checkouts will be scheduled to ensure readiness of all participants for actual shipments.

It is expected that the products of the preoperational checkouts would include:

- Final shipment procedures for waste facilities and the WIPP, including the WIPP Waste Information System
- Final procedures for interactions with States and Indian Tribes regarding TRU waste shipments
- Final procedures for TRU waste receipt, unloading, and emplacement
- Driver training and familiarization with the preferred routes
- Operational readiness reviews for each waste facility confirming readiness to ship TRU waste.

A typical dry run will begin with the receipt of the empty TRUPACT-II container at the waste facility and end with receipt, unloading, and emplacement at the WIPP. The latter will be done at the discretion of the WIPP waste-handling operations manager. There is no mandatory requirement for the underground emplacement of drums for every checkout. During each dry run, various scenarios for en route events will be initiated by WIPP personnel or by the driver to test systems on the truck or at the WIPP. The locations of each event will be modified for each preoperational checkout to fit the participating waste facility. The dry runs will be tracked with the TRANSCOM system and monitored by WIPP personnel at the CCC; digital communication will be established with the driver on a periodic basis, following established TRANSCOM procedures. As a minimum, on the return trip, drivers will input simulated "shipment problems" via the TRANSCOM to test the CCC operator responsiveness. These may include, but are not limited to, mechanical problems, protestors, sabotage, vehicle accidents, severe weather conditions, or the need to deviate from the preferred route. The CCC operator, following approved procedures, will provide the appropriate direction. On at least one occasion, the operator will ignore a message from the driver to verify that the Transportation Control Center in Oak Ridge, Tennessee, is monitoring the shipment.

Dry runs provide an opportunity to test various shipment scenarios. Data obtained regarding travel times to and from each facility will be used to establish a baseline for future shipments. All routes used during the dry runs will be those contained in the DOE-approved trucking management plan. On occasion, the driver will be instructed to deviate from these routes to test the alertness of the shipment monitoring agencies.

Summaries of various dry-run test scenarios are provided below with the expected response. Those summaries marked with an asterisk were used on dry runs in January and June 1989. These dry runs used an "engineering model" of the TRUPACT-II on a prototype WIPP trailer. These initial dry runs were made to determine shipment time, and to give the driver experience in using the TRANSCOM keyboard, in interacting with the TRANSCOM operator, in using the mobile phone, in using the KAVOURAS weather forecast system, and in responding to a variety of simulated accident scenarios.

*1) Evaluator-induced scenario: National weather channel indicates severe storm approaching the shipper's area. KAVOURAS system indicates temperatures below zero and 15-mph winds.

The operator contacts the facility traffic manager and Transportation Operations personnel to make a coordinated decision of appropriate action. The trucking contractor should be notified of delay if not alerted by driver.

*2) Evaluator-induced scenario: No communication capability with driver through TRANSCOM.

The operator attempts to call the driver via the mobile phone. Instructs driver to call in every 2 hours or when crossing a State border. The operator provides the Transportation Control Center with location provided by driver for manual input to TRANSCOM.

3) Driver-induced scenario: Tractor placed out of service because of excessive play on the right front axle. Vehicle cannot be repaired locally and must be replaced. The gross vehicle weight at weigh station was 79,748 pounds. The driver will notify the Transportation Control Center, and secure approval for the Proposed Action.

The operator should notify the trucking contractor of replacement requirement, as well as the receiver (WIPP Transportation Operations) and the shipper. Weight was specified, as it will require a special tractor not to exceed the 80,000-pound limit. Operator should be aware of weight limitations.

*4) Driver-induced scenario: Broken radiator hose. Driver can arrange repair. Estimated 2-hour delay.

The operator will notify the trucking contractor and WIPP Transportation Operations.

5) Driver-induced scenario: Protesters harassing shipment. Path blocked by protestor vehicles. Carrier tractor damaged by thrown objects. Demonstrators becoming more and more violent.

The operator notifies WIPP Transportation Operations, the waste facility, local law enforcement agency, and trucking contractor. Tractor replacement may be required. The operator stays in contact with driver.

*6) Evaluator-induced scenario: Information provided by the State Highway Patrol: on the downhill slope of the pass, the tractor brakes failed; the driver attempted to keep control but the vehicle overturned. All three TRUPACT-II containers have broken loose and are scattered within 100 yards of the trailer. The drivers have been seriously injured. Not known whether there was any spread of contamination. No further information available at this time.

The operator follows the notification plan given in Appendix C.

*7) Evaluator-induced scenario: Two vehicle accident. Collision between carrier vehicle and auto which entered interstate from on ramp, cutting off tractor-trailer. The auto was totalled. The tractor driver was injured seriously. TRUPACT-II containers are undamaged. The tractor is inoperable (right front fender and frame crushed). Damage to car--$12,000; to tractor--$7,000. Local authorities at the scene; the ambulance has departed.

The operator notifies the WIPP Project Office, WIPP Transportation Operations, trucking contractor, and the facility traffic manager. The trucking contractor will arrange for replacement tractor and driver replacement.

*8) Driver-induced scenario: 100-mile check shows broken U-bolt in the third rearmost container, right rear corner.

The operator notifies the WIPP Transportation Operations, which arranges for installation of a replacement by a qualified individual. Appropriate staff at the WIPP Project Office would be notified of the event.

9) An evaluator-induced scenario that is yet to be used is as follows: At some point while a dry run shipment is traversing a State, the State police and highway patrol will be notified that TRANSCOM contact with the shipment has been lost and their assistance is requested in locating the vehicle. The State will use its resources to locate the vehicle and pull it over. Once located, the driver will contact the Central Coordination Center and notify the operator of his location. The State police or highway patrol representative will also notify his headquarters that the vehicle has been located, and they, in turn, will notify the CCC operator. This will exercise both lines of communication. This may be implemented in each State the vehicle passes through.

D.2.4 **VEHICLE TRACKING SYSTEM**

The CCC at the WIPP will use the Transportation Tracking and Communication System (TRANSCOM) to track TRU waste shipments. This system is operated by the DOE's Oak Ridge Operations Office and is linked to the WIPP at the CCC via a dedicated telephone line. TRANSCOM will use a land-based LORAN-C positioning system to obtain longitude/latitude information. This information is calculated by a LORAN-C receiver and transporter antenna attached to the trailer. Signals will be transmitted via satellite to a commercial ground station and then to the TRANSCOM Control Center (TCC). The satellite communications system allows digital communication between the driver and the CCC at the WIPP. The CCC is able to communicate directly with the en route driver by mobile telephone. The TCC will provide access to the tracking system to those Indian Tribes, States, and facilities that need to monitor TRU waste shipments.

The location of the tracked vehicle will be monitored by the CCC so as to detect any deviation from the preferred route. Frequency of detection is limited by the frequency of vehicle location transmissions to the TCC. For TRU waste shipments, the frequency will be approximately every 15 minutes.

In New Mexico, as elsewhere, the officials of the State and the Indian Tribes will also have access to limited functions of TRANSCOM. The appropriate software training will be provided to enable them to receive data regarding TRU waste shipments passing through their jurisdictions.

Integrated with the TRU waste shipment system will be a set of activities that function to deter, protect, detect, and respond to unauthorized possession, use, or sabotage of TRU waste shipments. These activities will include:

1) Close, continued surveillance of the en route shipment by means of the TRANSCOM vehicle tracking and two-way communications system.

2) Efforts to minimize intermediate stops for each shipment.

3) Constant surveillance of the vehicle and cargo during transit. One of the drivers in the two-person truck crew will remain with the unit at all times, including refueling, food, and relief stops. A vehicle will be considered to be under surveillance when one driver is in the vehicle, awake, and not in the sleeper berth, or is within 100 feet of the vehicle and has the vehicle within an unobstructed field of view.

4) Use of a tamper-proof fifth wheel locking device.

5) The use of an escort vehicle would be a decision made by the appropriate State agency, with due consideration for DOT regulations. The DOE does not plan to use any escorts because with real-time tracking of shipments, accident situations would be identified and communications with the vehicle would take place almost immediately.

TABLE D.2.1 Road segments of concern[a]

Defense facility/route	Milepost[b]	Geographic description	Description of concern
Mound Laboratory, Ohio[c]			
I-70 (Indiana)	34-33	Approximately seven miles west of Cloverdale, Indiana	Water in east-bound lane may puddle causing trucks to hydroplane
I-70 (Indiana)	18-17	Near Terre Haute, Indiana	Overpass bridge on curve will ice
I-70 (Indiana)	11	State Highway 46 interchange, near Terre Haute, Indiana	High-volume interchange
I-70 (Indiana)	7-6	US-41 Interchange, near Terre Haute, Indiana	High-volume interchange
Oak Ridge National Laboratory, Tennessee			
I-40 (Tennessee)	330-287	Highway segment between Crab Orchard and Cookeville, Tennessee	Mountain driving, ice on highway during winter storms, may be impassable
I-40 (Tennessee)	210 0.00	Nashville and Memphis, Tennessee	Interchanges are busy during rush hour traffic
I-40 (Arkansas)	69.61	Just east of Clarksville, Arkansas	Flat curve in west-bound lane
I-40 (Arkansas)	125.11	Just west of Conway, Arkansas	Flat curve in east-bound lane
Oak Ridge National Laboratory, Tennessee (continued)			
I-40 (New Mexico)	373-358	Just east of San Jon, New Mexico	Pavement is concrete and will freeze first

TABLE D.2.1 Continued

Defense facility/route	Milepost[b]	Geographic description	Description of concern
I-40 (New Mexico)	326-324	Palma Hill, west of Tucumcari, New Mexico	Ices in winter
I-40 (New Mexico)	310	West of Tucumcari, New Mexico	Bad curve, accident area
I-40 (New Mexico)	291-281	East of Santa Rosa, New Mexico	Windy, drifting snow conditions
Savannah River Site, South Carolina			
I-20 (Mississippi)	100-92	Approximately 30 mi west of Meridian, Mississippi	Ground shifting breaks up pavement, road under construction
I-20 (Mississippi)	100	Approximately 30 mi west of Meridian, Mississippi	Long, gradual curve
Hanford Reservation, Washington			
I-82 (Washington)	96.6 - 132.6	Interstate from Richland, Washington south to Oregon border	Subject to freezing rain late fall to early spring

TABLE D.2.1 Continued

Defense facility/route	Milepost[b]	Geographic description	Description of concern
I-84 (Oregon)	208.00 - 378.00	Pendleton to Ontario, Oregon Oregon has hazardous winter driving	Majority of I-84 in northeastern conditions; mountainous driving
I-84 (Oregon)	213.00 - 225.00	Approximately five miles east of Pendleton, Oregon	Fog in winter and steep grades on hill
I-84 (Oregon)	268.00 - 280.00	Approximately eight miles east of La Grande, Oregon (Ladd Canyon)	Mountain driving; snow and ice; winter driving conditions in canyon
I-84 (Idaho)[d]	0.00 - 25.00	Western Idaho border to Caldwell, Idaho	"Black ice" conditions in winter
I-84 (Idaho)	50.00 - 90.00	Boise to Mountain Home, Idaho	"Black ice" conditions in winter
I-84 (Idaho)	100.00 - 121.00	East of Mountain Home, to Glenns Ferry, Idaho	When wet, concrete paving may cause trucks to jackknife
I-84 (Idaho)	222.00 - 275.65	I-84/I-86 interchange in Idaho to Utah border	Low visibility due to blowing snow or dust in early spring and winter; in general, subject to poor weather conditions
Hanford Reservation, Washington (continued)			
I-84 (Utah)	87.70 - 111.70	Nine miles east of Ogden, Utah	Mountain driving; Wever Canyon is subject to high winds and blowing snow

TABLE D.2.1 Continued

Defense facility/route	Milepost[b]	Geographic description	Description of concern
I-84/I-80 (Utah)	168.00	Interchange in Utah	High speed on curve can cause trucks to overturn
I-80 (Utah)	168.00 - 180.00	From interchange east-bound on I-80, Utah	Mountain driving; curves and shady areas with ice in winter; history of vehicles sliding off road
I-80 (Utah)	186.00 - 198.00	I-80 in Utah to Wyoming border	Pavement changes to concrete and freezes in winter; problems with vehicles sliding off road
I-80 (Wyoming)	68.97 - 212.54	Between Little America and Rawlins, Wyoming	Icy roads and strong cross winds; may have concurrent ground blizzard conditions
I-80 (Wyoming)	235.00 - 300.00	Elk Mountain area, Wyoming	Many long and steep grades may have ice, blowing snow and blizzard conditions
I-80 (Wyoming)	323.05 - 359.98	Happy Jack Summit to Cheyenne, Wyoming	Icy roads and strong cross winds; have concurrent ground blizzard conditions
I-25 (Colorado)	298.9 - 272.4	Southern Wyoming border to Fort Collins, Colorado	Hazardous storms with high winds, ground blizzards, and ice conditions
Hanford Reservation Washington (continued)			
I-25 (Colorado)	221 - 197.2	Between 104th St. to Arapahoe Rd., Denver, Colorado	Morning rush hour traffic (6:00 a.m. to 9:00 a.m.); and evening rush hour traffic (4:00 p.m. to 6:00 p.m.)

TABLE D.2.1 Continued

Defense facility/route	Milepost[b]	Geographic description	Description of concern
I-25 (Colorado)	221 - 197.2	38th St. exit and the Alameda exit south to University Ave., Denver, Colorado	During heavy rain storms, segments segments may flood several feet
I-25 (Colorado)	221 - 197.2	Broadway viaduct near Santa Fe Dr. in Denver, Colorado	Has restricted access because it is elevated and may be subject to ice conditions
I-25 (Colorado)	174	Monument Hill, 17 miles north of Colorado Springs, Colorado	Subject to severe weather with high winds, heavy rain, icy conditions, and snow blizzards. I-25 in this location often closed for weather
I-25 (Colorado)	157.1	Colorado Springs, Colorado	Rush hour traffic conditions
I-25 (Colorado)	141.8	Bijou St. exit, Colorado Springs, Colorado	Unique curves and turns may be hazardous during weather or high-speed conditions
I-25 (Colorado)	103.5	Pueblo, Colorado	Rush hour traffic conditions
Hanford Reservation, Washington (continued)			
I-25 (Colorado)	100	Near Colorado Fuel & Iron Plant, Pueblo, Colorado	Unique curves and turns that may be dangerous during weather or high-speed conditions
I-25 (Colorado)	15.6	Elevated portion of I-25 in	Has restricted access because it

TABLE D.2.1 Continued

Defense facility/route	Milepost[b]	Geographic description	Description of concern
	Trinidad, Colorado		is elevated and may be subject to ice conditions
I-25 (Colorado)	0.0	Entire I-25 corridor in Colorado	Severe weather conditions may result in white-outs and heavy winds
I-25 (New Mexico)	454-460	Raton Pass, New Mexico	Moutain pass area, may be closed because of weather conditions
I-25 (New Mexico)	434	North of Maxwell, New Mexico	Curves and overpass may ice up
I-25 (New Mexico)	426-413	Between Maxwell and Springer, New Mexico	Winter ski traffic packs snow on road
I-25 (New Mexico)	374-369	South of Wagon Mound, New Mexico	Hill ices up in winter
I-25 (New Mexico)	323-307	South of Las Vegas, New Mexico	Icy hills with snow drifts in winter
Hanford Reservation, Washington (concluded)			
I-25 (New Mexico)	300-284	Glorietta Pass to Lamy area, New Mexico	Ices up with drifting snow Interchange to US-285 can be dangerous; US-285 is two-lane with old pavement
U.S.-285 (New Mexico)	276	White Lakes area, New Mexico	Hills, icy

TABLE D.2.1 Continued

Defense facility/route	Milepost[b]	Geographic description	Description of concern
U.S.-285 (New Mexico)	264-250	Clines Corners area, New Mexico	May have drifting snow with zero visibility, high winds
U.S.-285 (New Mexico)	239-238	South of Clines Corners, New Mexico	Hills, icy
U.S.-285 (New Mexico)	205-175	South of Vaughn, New Mexico	Snow pack, icy, windy
U.S.-285 (New Mexico)	135	20-Mile Hill, 30 mi north of Roswell, New Mexico	Long hill, weather change area
Rocky Flats Plant, Colorado			
SH-128;	48 - 57.2	Segment from Rocky Flats	Area is subject to high winds and
U.S.-36 (Colorado)		Plant to I-25, Colorado	severe snow blizzards; portions of road are two-lane

TABLE D.2.1 Continued

Defense facility/route	Milepost[b]	Geographic description	Description of concern
Idaho National Engineering Laboratory, Idaho			
SH 26/20 (Idaho)	272.00 - 306.00	Entire length of State road to I-15 interchange	Road is two-lane with old pavement. Severe weather may close road. Blowing snow and wind gusts exceeding 40-60 mph are not uncommon in winter
I-15 (Idaho)	92.50 - 00.00	Entire I-15 segment from Blackfoot, Idaho to Utah border	Mountain driving; winter closures for weather of blowing snow and high wind gusts of 40-60 mph. Also, segments 3 to 10 miles in length on I-15 will be under construction until 1995
I-15 (Utah)	397.5 - 381	Plymouth to Tremonton, Utah	Mountain driving, two lanes; under construction until 1992 or 1993
Lawrence Livermore National Laboratory, California			
I-580 (California)	1.48 - 8.29	Altamont Pass, San Joaquin County	Steep grades
I-5 (California)	[e]	From I-580 to Tejon Pass, in San Joaquin Valley	Subject to heavy fog, particularly months of Dec., Jan., and Feb., clearing by 10:00 a.m. Steep grades on Tejon Pass, may close for ice or snow conditions

Lawrence Livermore National Laboratory, California (concluded)

TABLE D.2.1 Continued

Defense facility/route	Milepost[b]	Geographic description	Description of concern
I-5, I-210, I-10 (California)	[f]	Freeway interchanges in Los Angeles area	Extremely hazardous, multiple freeway interchanges
I-10, I-15 (California)	[g]	I-10 to I-15 interchange	Hazardous freeway interchange. May have high winds
I-15 (California)	[h]	Cajon Pass in San Bernadino Mountains	Steep mountain road grades, may have ice and snow road closures
I-15 (California)	[i]	Near Victorville, California	Steep downhill grade with curve
Nevada Test Site, Nevada			
U.S.-95 (Nevada)	86.65 - 70	Las Vegas area, Nevada	Dangerous intersections; I-15 - US-95 interchange capacity problem; capacity/ safety problems from I-15 to Rainbow Blvd.
U.S.-95 (Nevada)	64.89 - 11.37	Junction of U.S.-93 in Las Vegas to Henderson	High speeds at intersections, construction until 1995
I-40 (New Mexico)	36-47; 63-68	Between Gallup and Grants, New Mexico	Rapid snow, ice accumulation
I-40 (New Mexico)	80-100	Between Grants and Laguna, New Mexico	Rapid snow and ice accumulation

Nevada Test Site, Nevada (concluded)

TABLE D.2.1 Continued

Defense facility/route	Milepost[b]	Geographic description	Description of concern
I-40 (New Mexico)	104	West of Laguna, New Mexico	Wind, rapid snow and ice accumulations, steep hill
I-40 (New Mexico)	114	Near Laguna, New Mexico	Curves, high accident area
I-40 (New Mexico)	115	Laguna area, New Mexico	Wind, steep hill, accident area
I-40 (New Mexico)	126-128	East of Laguna area, New Mexico	Interchanges
I-40 (New Mexico)	149	Nine Mile Hill, west of Albuquerque, New Mexico	Sharp exit
I-40 (New Mexico)	160	Albuquerque, New Mexico "Big I" interchange	All ramps ice quickly
I-40 (New Mexico)	170-184	Tijeras Canyon, east of Albuquerque, New Mexico	Icy, winds, poor visibility; road may be closed for weather
I-40 (New Mexico)	179-183	West of Edgewood, New Mexico	Long hill can ice up
I-40 (New Mexico)	194	West end of Moriarity, New Mexico	Bad curve, accident area, fog will settle over area

TABLE D.2.1 Concluded

[a] The States of Arizona, Illinois, Louisiana, Missouri, Ohio, Oklahoma, South Carolina and Texas did not report segments of concern. This table should be used in conjuction with Figure D.2.1 and the more detailed route figures. The DOE facilities are presented in order from Northeast to Southeast and Northwest to Southwest.

[b] Estimated.

[c] Segments of concern reported along interstate by defense facility until routes merge (e.g., I-40 in Oklahoma is confluence of Mound Laboratory route I-70 and Oak Ridge National Laboratory route I-40).

[d] Also route from Idaho National Engineering Laboratory to the WIPP.

[e] From milepost 28.06 in Stanislaus County to milepost 11.0 in Kern County, California.

[f] From milepost 46.58 to 42.44 (on I-10) in Los Angeles County, California.

[g] Milepost 9.95 on I-10 in San Bernadino County, California.

[h] Milepost 15 to 34 in San Bernadino County, California.

[i] Milepost 55 in San Bernadino County, California.

D.3 TRANSPORTATION RISKS

D.3.1 **INTRODUCTION**

This section presents an analysis of the risks involved in shipping CH and RH TRU waste to the WIPP. These risks fall into two general categories: radiological risks and nonradiological risks, and each of these categories can be further divided into risks incurred from transportation under normal conditions and from transportation accidents.

This analysis of transportation risks was conducted in a manner similar to other risk assessments, including the WIPP FEIS, using the methodology established by the NRC in studies done in the late 1970s. Although computer models and basic assumptions have been refined since these studies, the basic approach to assessing risk remains essentially the same. The primary reason for this stability of research methods is that this approach has proved to be accurate and reliable.

The analytical models or codes used in this analysis have been extensively documented elsewhere (Peterson, 1984; Joy et al., 1982; NRC, 1977; Taylor and Daniel, 1977; AEC, 1972). The code used to calculate radiological risks was RADTRAN II (Taylor and Daniel, 1982), a revision of the RADTRAN code (Taylor and Daniel, 1977). This code is the product of almost 15 years of development and is a flexible analytical tool for calculating the impacts of both normal transportation and transportation accidents.

The initial RADTRAN code and its subsequent versions have been used to prepare a number of key risk assessment documents, including the environmental assessment used in hearings held by the Interstate Commerce Commission on the issue of shipping radioactive materials by special-use trains; the Final Environmental Impact Statement on the Transportation of Radioactive Material by Air and Other Modes (NRC, 1977); the shipping risk analysis presented in the WIPP FEIS; and subsequent environmental and technical documentation for shipping TRU waste to the WIPP.

The RADTRAN model continues to be modified and refined; even at the present time changes are being made to the code. However, the versions of RADTRAN used in this SEIS have been validated by extensive use and assessment.

The major revisions to RADTRAN II from the earlier RADTRAN version used in the FEIS include the following:

Incident-Free Model (Transportation Under Normal Conditions)

- Shielding options in urban and suburban areas
- Checks for regulatory consistency
- Addition of rail crew doses
- Inclusion of rail travel through urban areas
- Revision of dose-while-stopped model
- Three package-size discriminators for handlers

- Pedestrian dose evaluated in cities

Accident Model

- Groundshine dose evaluated
- Cloudshine dose evaluated
- Economic impacts included
- Early morbidities evaluated
- Genetic effects evaluated
- Building dose factors included
- Inclusion of urban pedestrian inhalation dose
- Addition of Pasquill stability category option
- Expanded material dispersibility classes

General

- Redesign of input and output

Incident-free radiological risks occur during routine transportation and are the result of public and worker exposures to direct radiation at levels allowed by transportation regulations. While radiation shielding is incorporated into package designs where needed in accordance with DOT and NRC regulations, workers, vehicle crew members, and the public along the transportation routes will be exposed to very low dose rates of direct radiation from the packages during incident-free transportation. These low doses usually fall below the threshold of natural background radiation.

In the case of transportation accidents, radiological risks could be incurred if any radioactive material is released into the environment and is spread by winds or possibly through the plume of a fire that occurs during the accident. Since TRU waste emits primarily nonpenetrating (i.e., will not penetrate the skin) radiation, the released material must be either inhaled or ingested in order to present an immediate health hazard.

In order to evaluate the radiological risks of accidents, it is necessary to do a probabilistic analysis--that is, to consider the probability of an accident occurring and the potential consequences of that accident. This analysis includes the following steps:

1) a description of the physical, chemical, and radiological characteristics of the waste
2) a system description (types of shipping containers, number of containers per shipment, etc.)
3) an identification of potential accident scenarios in which radioactive material may be released
4) a probability to be assigned to the release scenarios
5) an estimate of the amount and type of material released in each scenario (the release fraction)
6) an evaluation of consequences, most often in terms of radiation exposure to the worker and the public.

In addition, a credible probabilistic evaluation of the radiological risks of accidents must include variations in transportation routes, population density along the routes and weather characteristics that could affect the results.

In the RADTRAN transportation accident model, the consequences of accidents are apportioned among eight severity categories and calculated for truck and rail transport (see Tables D.3.15 and D.3.16). Each severity category is associated with a release fraction and probability of occurrence. These categories are related to fire and mechanical forces expected in an accident, but specific accident scenarios are not described for the severity categories. The model for calculating release combines the fraction of material that is released from the shipping container with the fraction of material that becomes airborne and the fraction of the released material that is of respirable size. These latter fractions are based on the characteristics of the waste and the mechanisms by which the release occurs.

For this analysis, an average release fraction for each severity category was estimated, and the shipping containers were assumed to respond the same way in an accident regardless of the waste contents or waste form. It was further assumed that there would be no release for accidents assigned to severity category one or two, which a Type B shipping container or cask (e.g., TRUPACT-II or RH cask) must survive intact in order to be certified by the NRC.

Releases from crush impacts were expected to be limited to the Type A containers (55-gal drums/standard waste boxes) only and those to be limited to the interior of the TRUPACT-II containers with no subsequent release for accidents below severity category six. Releases from the TRUPACT-II were assumed to be possible during accidents involving fires in category three or above. The release fractions were increased for each succeeding severity category. The release fractions for each severity category were combined with the accident rates for each category, the probability of a fire or impact event, the travel distance per shipment, and the fraction of travel through each population density zone to determine a cumulative, probability-weighted consequence for each shipment in terms of radiation doses.

To complement the radiological incident-free and probabilistic accident risk analysis, bounding case accidents were postulated and their radiological consequences analyzed. These accidents were assumed to occur under conditions which maximized, within reasonable bounds, the consequences to exposed population groups.

In addition to the analyses of transportation radiological risks, an analysis was conducted of the nonradiological risks associated with projected shipments of TRU waste. These risks include potential injuries and fatalities along the truck and rail routes from accidents that are unrelated to the cargo and are based on historical injury and fatality rates for truck and rail traffic. These risks also include the exposure of populations along the routes to vehicle emissions from the TRU truck and rail shipments.

Although the transportation of TRU waste cannot be made entirely risk free, with reasonable planning and control, risks can be reduced to a level usually below that of comparable shipments (e.g., commercial shipments of hazardous materials such as gasoline) on the nation's transportation routes.

A more complete picture of how various components of the transportation system fit together to provide reliability and ensure the safety of the TRU waste shipping campaign is provided when Appendix C, Appendix L, and Appendix M are reviewed in conjunction with this appendix.

- Appendix C discusses emergency response training, procedures, and plans for the WIPP shipping campaign.

- Appendix L discusses the design, certification, and operation of the TRUPACT-II shipping container for CH TRU waste and the NuPac 72B shipping cask for RH TRU waste.

- Appendix M summarizes the trucking contract, including qualifications standards and training requirements for drivers, and quality assurance standards applicable to operational activities.

The approach to the transportation of TRU waste continues to be based on proven and safe practices established in transporting this waste to retrievable storage facilities at several sites over the last 20 years. These transportation practices are enhanced by the training, certification, regulatory compliance, safety, and quality assurance procedures discussed in the above-cited appendices.

D.3.2 **INCIDENT-FREE RISKS**

D.3.2.1 **Method for Calculating Radiological Risks from Normal Transportation**

The analysis of incident-free radiological risks began with an estimate of the volumes and characteristics of the waste to be transported. As discussed in more detail in Appendix B, the volumes of waste currently in storage and projected to be generated through the year 2013 were estimated from the 1987 Integrated Data Base (ORNL, 1987). These volumes were scaled-up to the maximum amount of waste that could be emplaced at the WIPP (approximately 6.45 million ft^3) and are shown in Table D.3.1. The analysis assumed that for truck shipments CH TRU waste would be packaged in Type A 55-gallon drums and transported in TRUPACT-II shipping containers, with each TRUPACT-II carrying two 7-packs of drums and 3 TRUPACT-II containers or 42 drums, per shipment. RH TRU waste was assumed to be transported in RH casks (one cask per shipment). For these conditions, the number of shipments to the WIPP was calculated as shown in Table D.3.2. For rail shipments, six TRUPACT-II containers on a single railcar constitute a CH shipment, and two RH casks on a railcar constitute an RH shipment.

For incident-free shipments, important waste characteristics include the radionuclide composition of the waste and the total amount (curies) of each radionuclide transported

TABLE D.3.2 Projected number of CH TRU and RH TRU waste shipments from generator and storage facilities to the WIPP

Facility	Number of shipments 100% Truck	Maximum rail
Contact-Handled[a,b]		
Idaho National Engineering Laboratory	4046	2023
Rocky Flats Plant	7608	3804
Hanford Reservation	3103	1552
Savannah River Site	2640	1320
Los Alamos National Laboratory	2065	2065[c]
Oak Ridge National Laboratory	228	114
Nevada Test Site	80	80[c]
Argonne National Laboratory-East	14	7
Lawrence Livermore National Laboratory	969	485
Mound Laboratory	150	75
TOTAL	20903	11525
Remote-Handled[d]		
Idaho National Engineering Laboratory	487	244
Hanford Reservation	2470	1235
Los Alamos National Laboratory	101	101[c]
Oak Ridge National Laboratory	4605	2303
Argonne National Laboratory-East	300	150
TOTAL	7963	4033

[a] Shipments based on 3 TRUPACT-IIs per truck shipment and 6 TRUPACT-IIs per railcar shipment.

[b] Truck shipments calculated from a drum volume of 0.2 m^3/drum x 14 drums/TRUPACT-IIs x 3 TRUPACT-IIs/Truck.

Rail shipments from a drum volume of 0.2 m^3/drum x 14 drums/TRUPACT-IIs x 6 TRUPACT-IIs /Railcar.

[c] Los Alamos National Laboratory and Nevada Test Site do not have access to rail, thus truck shipments are included in the maximum rail case.

[d] Truck shipments calculated from a NuPac 72B volume of 0.89 m^3/NuPac 72B x 1 NuPac 72B/Truck.

Rail shipments calculated from a NuPac 72B volume of 0.89 m^3/NuPac 72B x 2 NuPac 72B/Railcar.

per shipment. Using the waste volumes presented in the 1987 Integrated Data Base, and the information on waste characteristics provided by the facilities, the radioactivity characteristics of average truck or rail shipments of TRU waste from each of the sites were determined and are shown in Table D.3.3 for CH TRU waste and Table D.3.4 for RH TRU waste. Site-specific values of the Transport Index (TI) for a typical shipment of CH and RH TRU waste were developed by the WIPP and generator/storage site personnel. The TI represents the radiation dose rate at 1 meter (3.28 ft) from the surface of the shipping container (TRUPACT-II with a load of 14 drums of waste or an RH cask) and depends on waste density, distribution of radionuclides, quantity of radionuclides per shipment, mix of waste types, self-shielding provided by the waste, and shielding provided by the TRUPACT-II container or RH cask. The TI is very sensitive to small quantities of gamma-emitting fission products such as Cobalt-60 and Cesium-137. TI values for typical shipments from each facility are shown in Table D.3.5. The radiation dose rate represented by the TI was used to calculate radiation exposures of occupational populations (i.e., crew, shipment inspectors, waste handlers) and nonoccupational populations (people living or traveling along shipment routes, and people in the vicinity of the shipment while it is stopped). These TI values are very conservative (see Appendix B) in that they were based on two key assumptions: 1) the maximum drum surface dose rates as measured by the facilities and 2) a drum source term and energy of 1 MeV. A more typical source term energy would be 0.06 to 0.1 MeV_E for CH TRU waste.

In the RADTRAN model, the people living along shipment routes were classified into urban, suburban, and rural fractions with respective population densities of 3,861, 719, and 6 persons per square kilometer as specified by the NRC (1977). These population densities are quite typical of urban, suburban, and rural environments. For example, statistics from the Denver Regional Council of Governments show that along Interstate 25 through Denver only a small area around downtown Denver has a population density exceeding the urban figure used in RADTRAN (3,997 persons per square kilometer for Denver versus the 3,861 assumed by RADTRAN). Other segments through Denver have much lower population densities than the RADTRAN urban value. Fifteen miles south of downtown, population densities along I-25 approach the rural value of six persons per square kilometer.

For truck shipments, the HIGHWAY model (Joy et al., 1982) was used to estimate trip lengths from various facilities to the WIPP and the corresponding population density fractions along these routes. The routes selected generally follow interstate highways as specified by the DOT for shipments of route-controlled quantities of radioactive materials. For rail shipments, the INTERLINE model (Peterson, 1984) was used to estimate trip lengths and population density fractions. The selected routes follow Class A/Class B main lines. These distances and population density fractions are summarized in Table D.3.6. Other major input parameters to RADTRAN are summarized in Table D.3.7.

D.3.2.2 **Results of the Analysis**

The radiation exposures that would be received from the normal transportation of CH and RH TRU waste by truck and rail are shown in Tables D.3.8 and D.3.9. These exposures are summarized for both occupational and nonoccupational populations. The radiological exposures are presented on a per-shipment basis for each facility and are given in doses (person-rem) received by the exposed population for each shipment. These per-shipment exposures were used to calculate the total incident-free transportation exposures for the Proposed Action and the two alternatives (see Table

TABLE D.3.4 Average radioactivity in a shipment of RH TRU waste[a]

	Waste facility[b]				
Radionuclide	ANLE	HANF	INEL	LANL	ORNL
Cobalt-60	0.00×10^{0}	2.97×10^{0}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}
Strontium-90	0.00×10^{0}	6.76×10^{0}	4.08×10^{0}	7.99×10^{0}	1.12×10^{0}
Ruthenium-106	0.00×10^{0}	1.89×10^{-3}	0.00×10^{0}	6.31×10^{0}	0.00×10^{0}
Antimony-125	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	1.95×10^{-1}	0.00×10^{0}
Cesium-137	8.83×10^{0}	9.46×10^{0}	5.81×10^{0}	6.18×10^{0}	4.42×10^{-2}
Cerium-144	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	6.22×10^{1}	0.00×10^{0}
Europium-155	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	3.13×10^{-1}	0.00×10^{0}
Thorium-232	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}
Uranium-233	0.00×10^{0}	5.41×10^{-4}	0.00×10^{0}	0.00×10^{0}	4.56×10^{-3}
Uranium-234	0.00×10^{0}	8.11×10^{-5}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}
Uranium-235	1.21×10^{-5}	2.43×10^{-6}	8.68×10^{-2}	9.48×10^{-5}	1.87×10^{-6}
Uranium-238	0.00×10^{0}	5.41×10^{-5}	2.46×10^{-2}	0.00×10^{0}	1.96×10^{-6}
Neptunium-237	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}
Plutonium-238	0.00×10^{0}	9.73×10^{-2}	1.63×10^{-2}	0.00×10^{0}	1.18×10^{-3}
Plutonium-239	2.52×10^{-1}	1.38×10^{0}	8.80×10^{1}	8.29×10^{-1}	3.67×10^{-2}
Plutonium-240	9.27×10^{-2}	4.05×10^{-1}	3.58×10^{1}	2.73×10^{-1}	0.00×10^{0}
Plutonium-241	0.00×10^{0}	8.11×10^{0}	0.00×10^{0}	1.26×10^{1}	0.00×10^{0}
Plutonium-242	0.00×10^{0}	8.65×10^{-5}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}
Americium-241	0.00×10^{0}	5.95×10^{-1}	3.27×10^{-3}	0.00×10^{0}	1.88×10^{-2}
Curium-244	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	1.69×10^{-1}
Californium-252	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	2.91×10^{-1}
TOTAL	9.18×10^{0}	2.98×10^{1}	1.34×10^{2}	9.68×10^{1}	1.68×10^{0}

[a] Radioactivity in curies per shipment for the volumes of waste assumed for the SEIS analyses (i.e., volumes scaled up to correspond to the design capacity of the WIPP--see last column, Table B.2.4). The volume per shipment is 0.89 m^3 (one shipping cask per shipment).

[b] Key: ANLE, Argonne National Laboratory--East; HANF, Hanford Reservation; INEL, Idaho National Engineering Laboratory; LANL, Los Alamos National Laboratory; ORNL, Oak Ridge National Laboratory.

TABLE D.3.5 Transport index values[a]

Facility	CH TRU waste	RH TRU waste
Idaho National Engineering Laboratory	1.0	5.0
Rocky Flats Plant	1.5	[b]
Hanford Reservation	0.7	16.0
Savannah River Site	2.7	[b]
Los Alamos National Laboratory	4.1	8.9
Oak Ridge National Laboratory	11.0	3.2
Nevada Test Site	1.2	[b]
Argonne National Laboratory-East	7.5	2.5
Lawrence Livermore National Laboratory	0.4	[b]
Mound Laboratory	0.4	[b]

[a] mrem/hr at 1 meter from transporter surface.
[b] Blanks = RH TRU waste not stored at facility.

TABLE D.3.6 Average distances to the WIPP and percent of travel in various population zones[a]

	Average distance	Population zone		
	Miles	R	S	U
Truck				
Idaho National Engineering Laboratory	1521	85.0	13.8	1.2
Rocky Flats Plant	874	82.3	15.7	2.0
Hanford Reservation	1913	85.7	13.4	0.9
Savannah River Site	1585	74.3	25.1	0.6
Los Alamos National Laboratory	343	90.1	9.9	0.0
Oak Ridge National Laboratory	1350	78.6	20.7	0.7
Nevada Test Site	1286	86.8	11.2	2.0
Argonne National Laboratory-East	1387	78.1	21.8	0.1
Lawrence Livermore National Laboratory	1458	86.2	10.1	3.7
Mound Laboratory	1472	75.4	24.1	0.5
Rail				
Idaho National Engineering Laboratory	1761	89.5	9.8	0.7
Rocky Flats Plant	1098	86.7	11.6	1.7
Hanford Reservation	2296	87.8	11.5	0.7
Savannah River Site	1915	76.0	22.4	1.6
Oak Ridge National Laboratory	1630	79.8	18.9	1.3
Argonne National Laboratory-East	1469	81.6	17.0	1.4
Lawrence Livermore National laboratory	1873	85.0	14.3	0.8
Mound Laboratory	1677	76.8	21.3	1.9

[a] Mean population densities are utilized and correspond to:
R = Rural (6 persons/km^2)
S = Suburban (719 persons/km^2)
U = Urban (3861 persons/km^2).

Source: Madsen et al., 1983.

TABLE D.3.7 RADTRAN general input data[a]

Parameter	CH TRU waste Truck	CH TRU waste Rail	RH TRU waste Truck	RH TRU waste Rail
Package type	TRUPACT-II		Cask	
Package waste volume, m^3	2.8	2.8	1.0	1.0
Packages/shipment	3	6	1	2
Transport Index (TI), mrem/hr	(Site-specific, see Table D.3.5)			
Package length dimension, m	7.32	7.32	3.61	3.61
Number of crewmen	2	5	2	5
Distance from source to crew, m	4	152	5	152
Speed, km/hr				
Urban population zone	24	24	24	24
Suburban population zone	40	40	40	40
Rural population zone	88	64	88	64
Stop time per kilometer, hr/km	.011	.0036	.011	.0036
No. of people exposed while stopped	50	100	50	100
No. of people per vehicle	2	3	2	3
Population density, people/km^2				
Urban population zone	3861	3861	3861	3861
Suburban population zone	719	719	719	719
Rural population zone	6	6	6	6
Avg. rad./trailer-load of pkgs., Ci	(Site-specific, see Tables D.3.3 and D.3.4)			
Accident release fractions	(See Tables D.3.17 through D.3.22)			

[a] Source: Madsen et al., 1983.

TABLE D.3.8 Radiological exposures per CH TRU shipment (person-rem)[a,b,c]

	Truck		Rail	
Facility	Occupational	Nonoccupational	Occupational[d]	Nonoccupational
Idaho National Engineering Laboratory	5.0×10^{-2}	2.0×10^{-2}	2.9×10^{-4}	3.0×10^{-2}
Rocky Flats Plant	4.0×10^{-2}	1.0×10^{-2}	2.7×10^{-4}	2.0×10^{-2}
Hanford Reservation	3.9×10^{-2}	2.3×10^{-2}	2.6×10^{-4}	4.0×10^{-2}
Savannah River Site	1.4×10^{-1}	7.0×10^{-2}	8.4×10^{-4}	1.2×10^{-1}
Los Alamos National Laboratory	2.8×10^{-2}	8.0×10^{-3}	[e]	[e]
Oak Ridge National Laboratory	1.3×10^{-1}	2.0×10^{-1}	2.1×10^{-3}	2.0×10^{-1}
Nevada Test Site	5.0×10^{-2}	2.0×10^{-2}	[e]	[e]
Argonne National Laboratory-East	1.3×10^{-1}	1.4×10^{-1}	1.8×10^{-3}	1.9×10^{-1}
Lawrence Livermore National Laboratory	1.7×10^{-2}	9.0×10^{-3}	1.2×10^{-4}	1.6×10^{-2}
Mound Laboratory	1.9×10^{-2}	9.0×10^{-3}	1.1×10^{-4}	1.4×10^{-2}

[a] Exposures per waste shipment are expressed in equivalent whole body dose and are tabulated in units of person-rem.

[b] Values for rail are expressed per railcar shipment.

[c] Exposures per waste shipment are presented as a function of the Transport Index (TI) which is defined as the dose rate in mrem/hr at 1 meter from the waste package. Calculations are based on three TRUPACT-IIs per truck and six per railcar.

[d] Rail occupational exposures resulting from normal transportation include the impact of DOT inspection activities (.01 X Total Stop Time (hr) X TI).

[e] No railheads present.

TABLE D.3.9 Radiological exposures per RH TRU shipment (person-rem)[a,b,c]

	Truck		Rail	
Shipment origin facility	Occupational	Nonoccupational	Occupational[d]	Nonoccupational
Idaho National Engineering Laboratory	1.0×10^{-1}	8.0×10^{-2}	1.3×10^{-3}	1.3×10^{-1}
Hanford Reservation	1.7×10^{-1}	3.3×10^{-1}	3.5×10^{-3}	2.9×10^{-1}
Los Alamos National Laboratory	2.8×10^{-2}	1.2×10^{-2}	[e]	[e]
Oak Ridge National Laboratory	6.3×10^{-2}	4.4×10^{-2}	7.7×10^{-4}	7.4×10^{-2}
Argonne National Laboratory-East	5.0×10^{-2}	4.0×10^{-2}	5.5×10^{-4}	5.0×10^{-2}

[a] Exposures per waste shipment are expressed in equivalent whole body dose and are tabulated in units of person-rem.

[b] Values for rail are expressed per railcar shipment.

[c] Exposures per waste shipment are presented as a function of the Transport Index (TI) which is defined as the dose rate in mrem/hr at 1 meter from the waste package. Calculations are based on three TRUPACT-IIs per truck and six per railcar.

[d] Rail occupational exposures resulting from normal transportation include the impact of DOT inspection activities (.01 X Total Stop Time (hr) X TI).

[e] No railheads present.

D.3.10). The Proposed Action corresponds to an approximate 5-year Test Phase period during which up to 10 percent of the waste would be shipped to the WIPP by truck and a subsequent 20-year Disposal Phase during which the remainder of the waste would be shipped by either truck or rail. Cumulative exposures for the entire campaign in the Proposed Action are the sum of the total exposures from the Test Phase (truck shipments) and Disposal Phase (truck or rail shipments). The No Action Alternative does not involve transportation to the WIPP and therefore has no radiological exposures from transportation.

The Alternative Action also includes an approximate 5-year Test Phase during which approximately 300 drums of CH TRU waste would be shipped from the Rocky Flats Plant to the Idaho National Engineering Laboratory for bin storage tests. This would require approximately seven truck shipments with three TRUPACT-II containers per shipment. Assuming a per-shipment incident-free exposure which is the ratioed difference (based on Transport Index) between the per-shipment exposures for the Idaho National Engineering Laboratory to the WIPP and the Rocky Flats Plant to the WIPP (see Table D.3.8), the estimated occupational and nonoccupational incident-free exposures from these shipments are 0.035 person-rem and 0.02 person-rem, respectively.

Tables D.3.11 and D.3.12 summarize the differences between the Proposed Action and the Alternative Action in the radiological exposure to occupational and nonoccupational populations from transporting CH TRU waste under normal conditions.

Table D.3.13 shows the lifetime radiological exposure of transporting RH TRU waste under normal conditions during the Disposal Phase of either the Proposed Action or the Alternative Action. No RH TRU waste would be shipped during the Test Phase for either the Proposed Action or the Alternative Action. However, if RH TRU waste is shipped to the WIPP during the Test Phase, the lifetime radiological exposures would be spread over more than the 20 years assumed for the Disposal Phase.

Doses to maximally exposed individuals in various population groups over the 25-year shipping campaign (Test Phase and Disposal Phase) for the Proposed Action are presented in Table D.3.14. Two sets of dose tabulations are provided: one for 100 percent truck shipments and one for maximum rail. The totals represent the dose expected for an individual whose residence or occupation results in an exposure to all or a large number (depending on exposure group) of waste shipments. For the Alternative Action, these maximum individual doses would be identical, except that they would be received over a 20-year period.

Maximum individual doses were determined using the RADTRAN occupational and hypothetical maximum individual exposure models. The doses were adjusted or supplemented by more detailed models to account for individual doses due to inspections, refueling, food stops, rail operations, and traffic congestion. Estimates of individual doses (e.g., exposure duration, distances) for each of these activities were calculated using line source ($1/r$) or point source ($1/r^2$) approximations. No credit was taken for attenuation of radiation by the air or by any structures between the individual being exposed and the radiation source.

TABLE D.3.11 Summary of lifetime radiological exposures between Proposed Action and Alternative Action: CH TRU incident-free occupational exposures (person-rem)

Facility	Proposed Action Truck	Proposed Action Rail	Alternative Action Truck	Alternative Action Rail
Idaho National Engineering Laboratory	2.0×10^{2}	2.1×10^{1}	2.0×10^{2}	5.9×10^{-1}
Rocky Flats Plant	3.0×10^{2}	3.1×10^{1}	3.0×10^{2}	1.0×10^{0}
Hanford Reservation	1.2×10^{2}	1.2×10^{1}	1.2×10^{2}	4.0×10^{-1}
Savannah River Site	3.7×10^{2}	3.8×10^{1}	3.7×10^{2}	1.1×10^{0}
Los Alamos National Laboratory	5.8×10^{1}	5.8×10^{1}	5.8×10^{1}	5.8×10^{1}
Oak Ridge National Laboratory	3.0×10^{1}	3.2×10^{0}	3.0×10^{1}	2.4×10^{-1}
Nevada Test Site	4.0×10^{0}	4.0×10^{0}	4.0×10^{0}	4.0×10^{0}
Argonne National Laboratory-East	1.8×10^{0}	1.9×10^{-1}	1.8×10^{0}	1.3×10^{-2}
Lawrence Livermore National Laboratory	1.6×10^{1}	1.7×10^{0}	1.6×10^{1}	5.8×10^{-2}
Mound Laboratory	2.8×10^{0}	2.9×10^{-1}	2.8×10^{0}	8.2×10^{-3}
TOTAL	1.1×10^{3}	1.7×10^{2}	1.1×10^{3}	6.5×10^{1}

TABLE D.3.12 Summary of lifetime radiological exposures between Proposed Action and the Alternative Action: CH TRU incident-free nonoccupational exposures (person-rem)

	Proposed Action		Alternative Action	
Facility	Truck	Rail	Truck	Rail
Idaho National Engineering Laboratory	8.1×10^{1}	6.3×10^{1}	8.1×10^{1}	6.1×10^{1}
Rocky Flats Plant	7.6×10^{1}	7.6×10^{1}	7.6×10^{1}	7.6×10^{1}
Hanford Reservation	7.1×10^{1}	6.3×10^{1}	7.1×10^{1}	6.2×10^{1}
Savannah River Site	1.8×10^{2}	1.6×10^{2}	1.8×10^{2}	1.6×10^{2}
Los Alamos National Laboratory	1.6×10^{1}	1.6×10^{1}	1.6×10^{1}	1.6×10^{1}
Oak Ridge National Laboratory	4.6×10^{1}	2.5×10^{1}	4.6×10^{1}	2.3×10^{1}
Nevada Test Site	1.6×10^{0}	1.6×10^{0}	1.6×10^{0}	1.6×10^{0}
Argonne National Laboratory-East	2.0×10^{0}	1.4×10^{0}	2.0×10^{0}	1.3×10^{0}
Lawrence Livermore National Laboratory	8.7×10^{0}	7.9×10^{0}	8.7×10^{0}	7.8×10^{0}
Mound Laboratory	1.4×10^{0}	1.0×10^{0}	1.4×10^{0}	1.0×10^{0}
TOTAL	4.8×10^{2}	4.1×10^{2}	4.8×10^{2}	4.1×10^{2}

TABLE D.3.13 Summary of lifetime radiological exposures for incident-free transportation of RH TRU waste (person-rem): Proposed Action and Alternative Action

	Disposal Phase (20-yr)[a]			
	100% Truck		Maximum Rail	
Facility	Occ[b]	Nonocc[c]	Occ	Nonocc
Idaho National Engineering Laboratory	4.9×10^{1}	3.9×10^{1}	3.2×10^{-1}	3.2×10^{1}
Hanford Reservation	4.2×10^{2}	8.2×10^{2}	4.3×10^{0}	3.6×10^{2}
Los Alamos National Laboratory[e]	2.8×10^{0}	1.2×10^{0}	2.8×10^{0}	1.2×10^{0}
Oak Ridge National Laboratory	2.9×10^{2}	2.0×10^{2}	1.8×10^{0}	1.7×10^{2}
Argonne National Laboratory-East	1.5×10^{1}	1.2×10^{1}	8.2×10^{-2}	7.5×10^{0}
TOTAL	7.8×10^{2}	1.1×10^{3}	9.3×10^{0}	5.7×10^{2}

[a] No RH TRU waste is shipped to the WIPP during the Test Phase for any alternative.

[b] Occupational population-quantifies doses received by transportation crews.

[c] Nonoccupational population.

[d] Population group exposures are calculated by multiplying the exposure/shipment identified in Table D.3.9 by the total number of shipments to WIPP by truck or rail, as determined from the projections in Table D.3.2. Rail occupational exposures resulting from normal transportation include the impact of inspection activities.

[e] Waste shipments from this facility are limited to the truck mode. Rail exposures are thus the same as truck exposures.

Doses to a truck crew member include those received while the shipment is moving and stopped. The RADTRAN model was used to determine the exposure to an individual crew member while the shipment is moving. An exposure distance of 13 ft (4 m) was specified. Doses received while stopped are from inspections every 100 miles, refueling, and food stops. A truck driver, rather than a service attendant, is assumed to refuel the truck. Estimated exposure distances and durations for these activities while stopped are given in Table D.3.14. Depending upon the number of shipments from a facility and the travel time to the WIPP, a truck driver may transport all or only a fraction of the shipments. Hypothetical lifetime maximum crew member exposures are projected to be up to 130 rem for CH TRU waste shipments and up to 180 rem for RH TRU waste shipments. However, any monitored crew member who receives an accumulated dose that approaches 5 rem (the regulatory limit for occupational exposures) in any given year would be reassigned to other duties involving no further exposure.

Exposures to rail crew members while shipments are moving were also calculated using the RADTRAN model, with an exposure distance of approximately 490 ft (150 m). Exposure while stopped for inspections and servicing was estimated assuming a crew member radiation dose rate equal to the Transport Index value received over a duration of 1 percent of the total stop time (.033 hours per kilometer, typical of regular freight shipments).

The maximum individual dose to a railyard handler/serviceman was estimated assuming an average exposure distance of 33 ft (10 m) for a duration of 2 hours and that this person is exposed to approximately 13 percent of CH TRU shipments and 17 percent of RH TRU shipments (allowing for a 10-year career in the same position and three shifts/crew).

Maximum individual occupational exposures resulting from inspecting departing trucks were estimated assuming an exposure distance of approximately 3 ft (1 m) for 30 minutes. As above, it was also assumed that this individual would remain in the same job for 10 years, and that there would be three shifts/crews performing the same tasks. Individual dose commitments were projected to range from 0.0041 to 0.76 rem for CH TRU shipments and 0.063 to 3.3 rem for RH TRU shipments. The lifetime occupational exposure for truck inspections at the WIPP was estimated by summing the individual facility departure values, and resulted in a dose of 2.4 rem for CH TRU shipments and 4.8 rem for RH TRU shipments. The transportation worker performing rail departure inspections would receive the same maximum exposure as the worker inspecting departing truck shipments, since there are only one-half the number of shipments but about twice the inspection effort per shipment.

Estimated doses to an individual performing State safety vehicle inspections were calculated assuming the person would be involved in 20 percent of the inspections with an average exposure distance of approximately 3 ft (1 m). Inspections may occur at the origin facility, upon arrival at the WIPP, or in the corridor States at ports of entry for trucks or classification yards (transfer of railcar to another rail carrier) for rail shipments. To allow for queues, a truck inspection time of 1 hour was used. For individual railcar shipments, an inspection time of 45 minutes was assumed. For truck transportation, maximum lifetime inspection doses of 7.3 and 12 rem were calculated for CH TRU and RH TRU waste shipments. For rail transportation, maximum lifetime exposures of 5.9 rem (CH TRU) and 8.9 rem (RH TRU) were estimated.

The maximum radiation dose to an individual member of the public (off-link) due to waste shipments which travel by his or her residence or workplace was calculated using the RADTRAN model. It was assumed that the individual is exposed to every waste shipment at a distance of approximately 100 ft (30 m). For truck shipments, an additional exposure category (on-link) was evaluated to assess the radiation dose to a person in an adjacent traffic lane for an extended length of time due to traffic congestion. Assuming the individual is present for one 30-minute period in the adjacent traffic lane during the lifetime of the WIPP at an exposure distance of about 3 ft (1 m), individual doses could range from 0.2 to 8 mrem depending on the shipment's origin facility and type of waste (CH TRU or RH TRU).

The maximum individual dose to a member of the public working at a truckstop was calculated to be 480 mrem for CH TRU waste shipments and 980 mrem for RH TRU waste shipments. This assumes a stop duration of 2 hours, with an exposure distance of 65 ft (20 m). This also assumes that the individual is exposed to approximately 13 percent of all CH TRU shipments and 17 percent of all RH TRU shipments arriving at the WIPP (assuming all shipments stop at the same location, that the individual works for 10 years at the truckstop, and there are 3 shifts/crew.). Exposures to individuals employed at truckstops along routes leading from the individual waste origin facilities will be lower, ranging from .83 to 660 mrem, depending on the specific origin facility and type of waste shipped (CH TRU or RH TRU).

The maximum exposure to a member of the public residing near a train terminal was estimated assuming an exposure distance of 660 ft and that the individual is exposed to every railcar shipment for a duration of 20 hours per stop (Wooden, 1986 used for guidance). Lifetime doses of 0.3 rem for CH TRU shipments and 0.42 rem for RH TRU shipments were estimated.

D.3.3 **RADIOLOGICAL RISKS OF TRANSPORTATION ACCIDENTS**

D.3.3.1 **Method for Calculating Radiological Risks of Transportation Accidents**

D.3.3.1.1 Severity Categories. CH TRU and RH TRU shipments to the WIPP will be made in NRC-certified Type B containers (TRUPACT-II and RH cask). The certification standards ensure that these containers will withstand virtually any accident condition without releasing their radioactive contents to the environment. Recently, a 1987 NRC study (Fischer et al., 1987) determined that only 0.6 percent of truck and rail accidents involving Type B containers or casks could cause a radiation hazard to the public. The earlier 1977 NRC study (NRC, 1977) conservatively estimated that approximately 9 percent of all truck accidents and 20 percent of rail accidents involving Type B containers or casks would result in radioactive material releases. Thus, a TRU waste transportation accident that exceeds regulatory criteria and causes the release of a portion of the contents of the shipping container has an extremely small chance of occurring. However, in order to assure bounding estimates of environmental impact, the more conservative accident severity probability statistics from the older 1977 NRC study (NRC, 1977) are considered by RADTRAN to determine the overall, probabilistic transportation radiological risk.

The amount of radioactive material released in an accident depends on the severity of the accident, the characteristics of the waste, and the capabilities of the shipping container. Most accidents are unlikely to cause any release, but very severe accidents (much more severe than conditions represented by NRC certification standards for Type B containers) may cause some of the radioactive materials to be released. Thus, the distribution of accidents according to severity must be determined, in addition to the overall accident rate. In this subsection, the accident severity classification scheme that was used in this assessment is discussed. The distribution of accidents according to severity is presented for truck and rail shipping modes.

Accident severity categories define the seriousness of an accident in terms of mechanical and thermal loads. Many methods can be used to classify accidents in terms of mechanical and thermal parameters. The relevant mechanical parameters may include impact speed, impact force, impact location and orientation, impact surface hardness, and impact puncture characteristics. The thermal characteristics may include flame temperature, fire duration, fire source size and orientation with respect to the container, and heat transfer properties (such as flame emissivity and convection coefficients).

The NRC defined eight accident severity categories for each transportation mode in a study performed to assess the adequacy of regulations for radioactive material transport (NRC, 1977). The first two accident categories were defined to be less serious than the hypothetical accident conditions specified in 10 CFR Part 71 for testing Type B packaging (i.e., shipping containers or casks). These tests simulate very severe transportation accidents, with the packaging sequentially subjected to drop, puncture, thermal, and water immersion tests. Thus, accidents in severity categories 1 and 2 are very unlikely to cause any release to the environment because the shipping containers or casks are designed to withstand them without releasing any of their contents.

The NRC (1977) classification scheme for truck accidents, illustrated in Figure D.3.1, uses crush force and fire duration to determine the seriousness of an accident. The crush force may result from either an internal (e.g., container crushed upon impact by other containers in the load) or static load (e.g., container crushed beneath vehicle). The classification approach used for train accidents is shown in Figure D.3.2. While fire duration is retained as the thermal parameter, the NRC decided to use puncture and impact speed as the mechanical measure of accident severity. This was done because crushing from the impact of other containers in the cargo was considered less relevant for rail shipments.

The assessment used in this SEIS retains the severity classification scheme used by the NRC (1977). In order to place the accident severities into perspective, two accidents representative of categories 1 and 2 are described:

Figure D.3.1
Truck accident severity category classification scheme

Figure D.3.2
Railroad accident severity category classification scheme

In the accident known as the I-80 bridge accident, a tractor-trailer rig was struck by a pickup truck while on an overpass bridge on I-80 near San Francisco, California. The tractor-trailer rig veered into the bridge railing and fell to a soil surface 64 feet below. Fischer et al. (1987) determined that a comparable accident involving a Type B certified container would be within the accident conditions specified for the design of the containers and thus would not be expected to cause any significant release.

A truck accident involving a fire occurred in the Caldecott Tunnel near Oakland, California. The accident resulted from a collision involving a gasoline truck, a bus, and a car. The gasoline truck carried approximately 8,800 gallons of gasoline, which acted as the fire source; a resulting peak flame temperature of 1900°F was estimated. Although it took about 2 hours and 42 minutes to completely extinguish the fire, most of the gasoline burned in less than 40 minutes. Fischer et al. (1987) concluded in that the response of Type B containers to an accident of this type would be within the design capabilities.

For higher accident severities, there is an incremental increase in mechanical and thermal loads. At the highest severity category, impact forces can be 100 times greater than those in category 2, and fire durations can exceed 1.5 to 2 hours. For example, a fire that engulfs a truck shipment in a diameter of 40 feet would require approximately 17,000 gallons of hydrocarbon fuel to burn for 2 hours. This would require the very unlikely event of involving three tanker trucks in the incident because a typical tanker carries approximately 5,000 gallons of hydrocarbons (Wolff, 1984). At a minimum, at least two full 10,000-gallon tanker trucks would need to be involved. For a rail incident, the average fire pool size is 2,000 square feet (50 ft in diameter) (Wolff, 1984); over 27,000 gallons of hydrocarbon fuel would be required to maintain a fire of this magnitude for 2 hours. The large majority of truck (99.90 percent) and rail (99.83 percent) accidents that involve fires, however, last less than 30 minutes (Wolff, 1984). The probability of such accidents diminishes as their severity increases, as already noted.

Table D.3.15 presents the fractional occurrences of truck accidents in each of the eight severity categories. The assessment conducted for this SEIS assumes an overall accident rate of 1.1×10^{-6} accidents per kilometer (NRC, 1977). The fraction of accidents in each population zone relevant to TRU waste shipments to the WIPP is also presented in Table D.3.15.

Table D.3.16 presents the fractional occurrence of train accidents in each of the eight accident severity categories. The overall accident rate is 9.3×10^{-7} railcar accidents per railroad-kilometer, assuming an average train length of 70 cars and an average of 10 cars involved in each accident (NRC, 1977). The more severe accidents are assumed to occur in lower-population-density zones, where travel speeds are higher.

D.3.3.1.2 Release Fractions. The DOE plans to ship TRU waste to the WIPP in Type B shipping containers or casks whose designs are approved and certified by the NRC (see Appendix L). Type B containers or casks are designed and tested to NRC requirements to demonstrate that they are sufficiently strong to withstand very severe accidents, with safety largely independent of the transport vehicle and procedural and other controls on the shipment. Testing as specified by the NRC in 10 CFR 71.73

TABLE D.3.15 Fractional occurrences[a] for truck accidents by accident severity category and population density zone

Accident severity category	Fractional occurrences	Fractional occurrences according to population density zones		
		Low	Medium	High
I	.55	.1	.1	.8
II	.36	.1	.1	.8
III	.07	.3	.4	.3
IV	.016	.3	.4	.3
V	.0028	.5	.3	.2
VI	.0011	.7	.2	.1
VII	8.5×10^{-5}	.8	.1	.1
VIII	1.5×10^{-5}	.9	.05	.05

[a] Overall accident rate = 1.1×10^{-6} accidents/kilometer.

TABLE D.3.16 Fractional occurrences[a] for train accidents by accident severity category and population density zone

Accident severity category	Fractional occurrences	Fractional occurrences according to population density zones: Low	Medium	High
I	.50	.1	.1	.8
II	.30	.1	.1	.8
III	.18	.3	.4	.3
IV	.018	.3	.4	.3
V	.0018	.5	.3	.2
VI	1.3×10^{-4}	.7	.2	.1
VII	6.0×10^{-5}	.8	.1	.1
VIII	1.0×10^{-5}	.9	.05	.05

[a] Overall accident rate = 9.3×10^{-7} railcar accidents/kilometer.

encompasses a range of very severe accident conditions that are applied sequentially to determine cumulative effects; it includes impact (free drop), puncture, thermal, and water-immersion tests.

The 1977 NRC study (NRC, 1977) conservatively estimated that approximately 9 percent of all truck accidents and 20 percent of rail accidents involving Type B containers or casks could result in radioactive material releases. More recently, however, Fischer et al. (1987) determined that only 0.6 percent of truck and rail accidents could cause a radiation hazard to the public. To estimate how much radioactive material could be released to the environment for the very small number of accidents that exceed the containment design capabilities of the Type B containers or casks, a release fraction analysis was performed.

Release Fraction Definition. The release fraction analysis determined how much radioactive material could be released to the environment in a respirable, airborne form after a very severe accident that affects the containment capabilities of the shipping containers or casks. The calculation focused on respirable particle sizes with a mean aerodynamic diameter of less than 10 microns because inhalation is the primary exposure pathway for TRU elements. Particles that are larger will be expelled from the body and consequently are not as significant in estimating health effects. This calculational approach is consistent with existing NRC risk assessments (WASH-1400, NUREG-0170, NUREG/CR-4829).

Method of Calculating Release Fractions. In order to calculate release fractions for very severe accidents, it is necessary to:

- Characterize the radioactive material being transported

- Identify and quantify the response of the shipping containers or casks (loss of containment) to accident conditions

- Identify and quantify the release mechanisms resulting in the escape of radioactive material from the containers or casks to the environment.

This analysis used representative values for parameters where published data and test results are applicable and reasonable, and conservative estimates where uncertainties exist. "Conservative" is used in this discussion to mean using such parameter values that the consequences of potential accidents will be overestimated.

Characterization of the TRU Waste. The radionuclide compositions, quantities, and volumes used in the analysis are based on the waste inventory data and projections presented in Appendix B. As noted in Subsection 2.3.1, the DOE has established criteria and procedures which govern the physical, radiological, and chemical composition of the waste. Physical restrictions require that the waste not be in a free-liquid form and that particulate waste materials be limited to specific levels in accordance with DOE (1989). Transuranic radionuclides are generally present as oxides with concentrations exceeding 100 nanocuries per gram.

Response of Shipping Containers and Casks. If a shipping container or cask is involved in an accident, the extent of damage will depend on the design of the container and the severity of the accident. Accident severity is categorized in terms of mechanical (e.g., impact) and thermal loads. Many methods can be used to classify accidents in terms of mechanical and thermal parameters. The relevant mechanical parameters may include impact speed, impact force, impact location and orientation, impact surface hardness, and impact puncture characteristics. The thermal parameters may include flame temperature, fire duration, fire source size and orientation with respect to the containers, and heat transfer properties (e.g., flame emissivity and convection coefficients).

The analysis conducted for the SEIS used the accident severity model developed by the NRC (1977) as discussed in the preceding subsection. This model conservatively predicts the frequency of accidents whose severity exceeds Type B package test requirements (accident severity category three through eight).

Because NRC regulations do not require Type B containers to be tested to failure, and because there are no historical data on the response of containers to very severe accidents, certain assumptions were required to estimate the extent of damage sustained by the TRUPACT-II container and the RH cask from accidents in severity categories three through eight. Guidance was obtained from the analysis and test data presented in NRC (1977), Fischer et al. (1987), and Jefferson (1978). The data indicate that a catastrophic failure (e.g., gaping hole, container severed in half) of a Type B container or cask would not be expected for accidents more severe than those in severity category two. Because of margins in the materials of construction (e.g., minimum versus actual rupture stress) and structural design (e.g., absorption of energy by plastic deformation), more likely failures would include the formation of cracks in the side of the container or cask, the failure of the closure seals, or the failure of any valves or penetrations.

To define the response of Type B containers or casks to transportation accidents, the following conservative assumptions were made:

- For shipments of several Type B containers on one transport vehicle, it was assumed that all containers would sustain the same damage. No credit was taken for the mitigating effects of one container shielding the others from impact forces or thermal loadings.

- Two package response states were defined for the shipping container or cask:

  1) No leak path and no release of radioactive material

  2) A leak path is present, allowing the release of <u>all</u> respirable airborne radioactive material present inside the containers.

The second state was postulated even though catastrophic failures are very unlikely. This state is consistent with NRC's position (Fischer et al., 1987) and does not take credit for any processes that will tend to reduce radioactive material releases (e.g., particle settlement, vapor plate-out on interior surfaces, filtration effects along leak path) from the containers.

The response states are influenced by both the mechanical and thermal conditions of the accident. The response to the impact conditions will be largely independent of the thermal conditions, with impact effects immediate and thermal effects delayed. Consequently, the analysts elected to use two components for the response state (one for the impact event and one for the thermal event) for each accident severity category. Both components have two accident response states as defined above.

Once the potential response states for the shipping containers or casks have been defined, it is necessary to assign the appropriate response state components to each accident severity category. As previously noted, there are few data that can be used to determine failure thresholds for transport containers involved in accidents with conditions more severe than NRC certification test requirements. NRC (1977) Model II release fractions (Table 5-8 of reference) were used as a primary guide. From impact test data, the NRC (1977) projected Type B shipping containers for plutonium to have a failure threshold at accident severity category six. With current development programs, more recent container designs (1985) were projected to have an increased failure threshold, corresponding to accident severity category seven. The NRC (1977) also projected Type B casks to have a failure threshold at accident severity category three, with more significant releases occurring at accident severity category five. These projections included effects from both impact and thermal events.

For response to an impact event, a failure threshold corresponding to severity category five was assigned; it corresponds to the more significant release state projected by the NRC (1977) for Type B casks. For response to a thermal event, a failure threshold corresponding to severity category three (an accident with conditions slightly exceeding the NRC's test requirements) was conservatively assigned.

Release Mechanisms. Any release of radioactive material due to a transportation accident would normally progress in two stages: release inside the shipping containers or casks, followed by release to the environment. Releases from the container to the environment were addressed in the preceding discussion of accident response states. The discussion that follows evaluates how much radioactive material would be released into the cavities of the shipping containers or casks.

There are multiple release mechanisms and pathways that may lead to the release of respirable radioactive material into container cavities. Impact release mechanisms include waste container (e.g., a 55-gallon drum or standard waste box) failure, fragmentation of solid waste, particulate suspension, and aerodynamic entrainment of particles. Thermal release mechanisms include heat-induced failures of the waste containers; aerosolization of particles by combustion, gas generation, or the heating of contaminated surfaces; and potential volatilization of radionuclides. Impact and thermal release mechanisms were evaluated by using applicable test data and analyses available in the published literature, as supplemented by conservative assumptions where only limited data exist. It was assumed that all failed waste containers, without regard to waste form or type, release an average amount of material for each accident severity category.

In assessing releases from impact events for each severity category, the following procedure was used:

- Identification of the fraction of failed waste containers inside the shipping container or cask

- Determination of the fraction of radioactive material released from the failed waste containers

- Calculation of the fraction of radioactive material released from the failed waste containers that is aerosolized in a respirable form by the mechanical stress of impact

- Calculation of the fraction of radioactive material released from the failed waste containers that becomes aerodynamically entrained in a respirable form after the loss of containment by the shipping containers and any subsequent depressurization (e.g., TRUPACT-II design pressure of 50 psig).

Studies by Huerta (1983) and Shirley (1983) were used to determine the fraction of failed waste containers. The fractions of radioactive material released from the failed waste containers were conservatively estimated using reports by Huerta (1983) and the NRC (1977) for guidance. The fraction of radioactive material converted to a respirable aerosol from impact stresses was calculated by using a resuspension factor approach. This is an accepted analytical method for predicting airborne concentrations of material above contaminated surfaces. The mechanical action of vigorous sweeping was used to represent the respirable airborne contamination fraction, using data taken from an NRC report (NRC, 1980), for the resuspension factor.

It was judged that this approach would be at least representative, if not conservative, in estimating the release of respirable contaminants by impact stresses.

The aerodynamic entrainment of respirable particulates was determined by using data from wind tunnel tests for uranium dioxide power (Mishima and Schwendiman, 1973a). This release mechanism will occur only to the extent that the shipping container is pressurized by the release of gases from the waste containers. The analysis conservatively assumed that maximum pressurization of the container cavity will always occur for every shipment. Based upon the nature of potential container damage previously described, and the void volume space within the container cavity, a depressurization duration of approximately 30 minutes at an average velocity of about 2.5 mph was calculated. For these conditions, the average entrainment value given by Mishima and Schwendiman (1973a) for four surfaces (asphalt, sand, vegetation, and stainless steel) was conservatively assigned.

The algorithm used to calculate the release fraction of respirable radioactive material from impact stresses is summarized in Table D.3.17. Values for specific algorithm parameters are presented in Table D.3.18.

TABLE D.3.17 Estimate of potential accident release fractions for CH and RH TRU waste shipments due to impact events

Impact release fraction (IRF) = (FFC x FMRC) (FMAI + FMEI) (FMRPI)

Where:

FFC = Fraction of failed waste containers

FMRC = Fraction of material released from failed containers into package cavity

FMAI = Fraction of material aerosolized from impact

FMEI = Fraction of material entrained to environment during

FMRPI = Fraction of material released from package cavity during impact event

					TRUPACT-II[a]		RH Cask [a,b]	
Severity category	FMRC	FMAI	FMEI	FMRPI	FFC	IRF	FFC	IRF
1	0×10^{0}	0×10^{0}	0.0×10^{0}	0×10^{0}	0×10^{0}	0×10^{0}	0×10^{0}	0×10^{0}
2	0×10^{0}	0×10^{0}	0.0×10^{0}	0×10^{0}	0×10^{0}	0×10^{0}	0×10^{0}	0×10^{0}
3	1×10^{-1}	8×10^{-5}	0.0×10^{0}	0×10^{0}	3×10^{-1}	0×10^{0}	3×10^{-1}	0×10^{0}
4	3×10^{-1}	8×10^{-5}	0.0×10^{0}	0×10^{0}	5×10^{-1}	0×10^{0}	7×10^{-1}	0×10^{0}
5	5×10^{-1}	8×10^{-5}	1.5×10^{-4}	1×10^{0}	7×10^{-1}	8×10^{-5}	1×10^{0}	1×10^{-4}
6	7×10^{-1}	8×10^{-5}	1.5×10^{-4}	1×10^{0}	1×10^{0}	2×10^{-4}	1×10^{0}	1×10^{-4}
7	1×10^{0}	8×10^{-5}	1.5×10^{-4}	1×10^{0}	1×10^{0}	2×10^{-4}	1×10^{0}	2×10^{-4}
8	1×10^{0}	8×10^{-5}	1.5×10^{-4}	1×10^{0}	1×10^{0}	2×10^{-4}	1×10^{0}	2×10^{-4}

[a] Respirable release fractions.

[b] Release fractions are the same for truck and rail transportation modes.

TABLE D.3.18 Impact release algorithm parameters for CH and RH TRU waste shipments

Parameters	Value	Basis/reference
FFC	.2728 lnF -2.814	Huerta (1983); Shirley (1983). Where F is NRC (1977) accident severity breach force (Newtons)
FMRC	Table D.3.17	Huerta (1983) and NRC (1977) used as guidance
FMAI	Table D.3.17	NRC (1980) resuspension factor of 2.00×10^{-2} m^{-1} used (mechanical stress of vigorous sweeping)
FMEI	1.50×10^{-4}	Mishima and Schwendiman (1973a) average entrainment value for 4 surfaces used with airflow of 2.5 mph for 30 minutes
FMRPI	Accident severity 1-4: 0.0 Accident severity 5-8: 1.0	Type B package design and NRC (1977) used as guidance

Fischer et al. (1987) estimated that 1.7 percent of truck accidents and 6.8 percent of rail accidents will involve fires. For fire events, the following method was used for each accident severity category:

- Identification of the fraction of radioactive material subject to thermal release mechanisms

- Calculation of the fraction of radioactive material released by combustion in a respirable form

- Calculation of the fraction of radioactive material released in a respirable form by the release of gases and the heating of contaminated surfaces

- Determination of the fraction of radioactive material released in a respirable form from any volatilization of radionuclides.

In the absence of detailed knowledge about the responses of shipping containers and waste containers to fires more severe than those specified in regulatory test requirements for Type B packagings, it was conservatively assumed that all radioactive material was available for release for all accidents exceeding severity category two, as limited by the specific release mechanisms.

For combustion related releases, it was assumed that combustible materials could be ignited in all accident severity categories exceeding category two. To maximize the amount of combustible waste burned for a given amount of oxygen, incomplete combustion, producing carbon monoxide (CO), was assumed. The amount of oxygen present to support combustion was calculated by assuming an 85 percent void volume for a loaded shipping container and observing that there would be no external sources of air or oxygen (no major breach of container). From a review of the inorganic compound tables in the Handbook of Chemistry and Physics, it was concluded that any decomposition of metal hydroxides (e.g., $Ca(OH)_2$, $Al(OH)_3$) present in cemented sludges would not act as an internal source of additional oxygen. Finally, the results of experiments conducted by Mishima and Schwendiman (1973b) were used to assess the fraction of radioactive material released in a respirable form from the burning of combustible material.

For accident severity categories four through eight, the fire event may last longer than 1.5 hours. For these more severe conditions, it was assumed that more radioactive material could be converted to an aerosol form because of the release of gases from the waste at elevated temperatures. Potential gas generation was assumed to be comparable for all five accident severity categories and was calculated by assuming a graphite/steam reaction as the off-gassing source. For an upper bound gas generation estimate, it was further assumed that all waste containers within the shipping container were loaded with solidified process waste (water/steam source) and that there was adequate graphite (e.g., molds) present to react with all of the steam.

With these assumptions, gas generation was calculated to be in excess of 600 TRUPACT-II void volumes and 700 RH cask void volumes, at atmospheric pressure. The fraction of respirable radioactive material present in the gases released from the waste containers and subsequently to the environment was calculated by using a resuspension factor approach. A resuspension factor value corresponding to a vigorous and continued surface stress of people walking on a surface contaminated with plutonium dioxide (at a rate of 36 steps per minute) was used in the analysis.

Vaporization was reviewed as another thermal release mechanism. As previously noted, TRU radionuclides are generally present in an oxide form. They are highly stable at elevated temperatures. Alexander et al. (1986) report that volatile releases of transuranic radionuclides are not of any significance until temperatures of 3140°F are reached. The volitization of uranium oxide (e.g., UO_2) becomes measurable at approximately 2960°F. Flame temperatures for the open burning of hydrocarbon fuels (e.g., JP-4, gasoline, diesel) range from 1400°F to 2400°F, with a median temperature of approximately 1800°F. Consequently, a volatile release of TRU or uranium oxide material is not credible for a transportation accident. This is consistent with the release analysis presented by Fischer et al. (1987), in which the releases of TRU material are quantified in terms of particulates only. In conjunction with waste characterization data, it can be concluded that potential accidents involving CH TRU waste shipments cannot result in radioactive material releases in a vapor form. However, RH TRU waste contains activation/fission products that may volatilize at elevated temperatures. These radionuclides are identified as being present in RH TRU waste. Testing conducted by Lorenz (1980) indicates that cesium, antimony, and ruthenium may volatilize at elevated temperatures. Assuming that volatilization mechanisms for RH TRU waste would be similar to the referenced test conditions at 1290°F, it was concluded that the releases of cesium, antimony, and ruthenium vapors would be comparable to the values estimated for respirable particulate releases.

The algorithm for estimating the respirable release fraction of radioactive material from thermal accident events is illustrated in Table D.3.19. Values for specific algorithm parameters are summarized in Table D.3.20.

Total Respirable Release Fractions. The calculated impact release fractions (Table D.3.17) and thermal release fractions (Table D.3.19) were added to determine the total respirable release fractions due to very severe transportation accidents and are summarized in Table D.3.21 and D.3.22. A maximum release fraction of 0.0002 was estimated for accidents involving both CH and RH TRU waste shipments. This is consistent with or bounding of previous transportation risk studies such as the NRC modal study (Fischer et al., 1987), which estimated particulate releases of 0.000002 and vapor (C_s)releases of 0.0002 due to spent fuel shipments, and the WIPP FEIS (DOE, 1980), which incorporated a release fraction of 0.00018 for CH TRU waste shipments.

D.3.3.1.3 Dispersal Conditions. The dispersion of airborne radioactive material during an accident is controlled by meteorological conditions at the time of the accident. The airborne radioactive material moves downwind from the scene of the accident and its dispersal and transport are affected by the degree of atmospheric turbulence. For this analysis, the materials were assumed to move downwind and disperse. As the radioactive cloud disperses, the people in its path will be exposed to external radiation, internal radiation from inhalation, or internal radiation from ingestion. For inhalation and TABLE D.3.19 Estimate of potential accident release fractions for CH and RH TRU waste shipments due to thermal events

Thermal release fraction (TRF) = FAT [(FMC x FMAC) + FMAT] FMRPT

Where: FAT = Fraction of accidents involving a thermal event

FMC = Fraction of material consumed by combustion

FMAC = Fraction of material aerosolized by combustion

FMAT = Fraction of material aerosolized by thermal event

FMRPT= Fraction of material released from package cavity during thermal event

					Truck[a]		Rail[a]	
Severity Category	FMAC	FMAC	FMAT	FMRPT	FAT	TRF	FAT	TRF
TRUPACT-II								
1	0×10^{0}	0×10^{0}	0×10^{0}	0×10^{0}	1.7×10^{-2}	0×10^{0}	6.8×10^{-2}	0×10^{0}
2	0×10^{0}	0×10^{0}	0×10^{0}	0×10^{0}	1.7×10^{-2}	0×10^{0}	6.8×10^{-2}	0×10^{0}
3	9×10^{-4}	5×10^{-4}	2×10^{-8}	1×10^{0}	1.7×10^{-2}	8×10^{-9}	6.8×10^{-2}	2×10^{-8}
4	9×10^{-4}	5×10^{-4}	1×10^{-5}	1×10^{0}	1.7×10^{-2}	2×10^{-7}	6.8×10^{-2}	7×10^{-7}
5	9×10^{-4}	5×10^{-4}	1×10^{-5}	1×10^{0}	1.7×10^{-2}	2×10^{-7}	6.8×10^{-2}	7×10^{-7}
6	9×10^{-4}	5×10^{-4}	1×10^{-5}	1×10^{0}	1.7×10^{-2}	2×10^{-7}	6.8×10^{-2}	7×10^{-7}
7	9×10^{-4}	5×10^{-4}	1×10^{-5}	1×10^{0}	1.7×10^{-2}	2×10^{-7}	6.8×10^{-2}	7×10^{-7}
8	9×10^{-4}	5×10^{-4}	1×10^{-5}	1×10^{0}	1.7×10^{-2}	2×10^{-7}	6.8×10^{-2}	7×10^{-7}
RH Cask								
1	0×10^{0}	0×10^{0}	0×10^{0}	0×10^{0}	1.7×10^{-2}	0×10^{0}	6.8×10^{-2}	0×10^{0}
2	0×10^{0}	0×10^{0}	0×10^{0}	0×10^{0}	1.7×10^{-2}	0×10^{0}	6.8×10^{-2}	0×10^{0}
3	7×10^{-4}	5×10^{-4}	2×10^{-8}	1×10^{0}	1.7×10^{-2}	6×10^{-9}	6.8×10^{-2}	2×10^{-8}
4	7×10^{-4}	5×10^{-4}	9×10^{-5}	1×10^{0}	1.7×10^{-2}	2×10^{-7}	6.8×10^{-2}	7×10^{-7}
5	7×10^{-4}	5×10^{-4}	9×10^{-5}	1×10^{0}	1.7×10^{-2}	2×10^{-7}	6.8×10^{-2}	7×10^{-7}
6	7×10^{-4}	5×10^{-4}	9×10^{-5}	1×10^{0}	1.7×10^{-2}	2×10^{-7}	6.8×10^{-2}	7×10^{-7}
7	7×10^{-4}	5×10^{-4}	9×10^{-5}	1×10^{0}	1.7×10^{-2}	2×10^{-7}	6.8×10^{-2}	7×10^{-7}
8	7×10^{-4}	5×10^{-4}	9×10^{-5}	1×10^{0}	1.7×10^{-2}	2×10^{-7}	6.8×10^{-2}	7×10^{-7}

[a] Respirable release fractions.

TABLE D.3.20 Thermal release algorithm parameters for CH and RH TRU waste shipments

Parameter	Value	Basis/reference
FAT	1.7×10^{-2} (Truck) 6.8×10^{-2} (Rail)	Fischer et al. (1987)
FMC	Accident severity 1-2: 0×10^{0}	
	Accident severity 3-4: 9×10^{-4} (TRUPACT-II) 7×10^{-4} (RH Cask)	Type B package design Limited internal oxygen source: 3.95 lb O_2 (TRUPACT-II) 0.73 lb O_2 (RH Cask)
FMAC	Accident severity 1-2: 0×10^{0}	Type B package design Mishima and Schwendiman (1973b)
	Accident severity 3-8: 5×10^{-4}	
FMAT	Accident severity 1-2: 0×10^{0}	Type B package design
	Accident severity 3: 2×10^{-8}	Only combustion assumed to occur, with attendant off-gas (combustion) products
	Accident severity 4-8: 1×10^{-5} (TRUPACT-II) 9×10^{-6} (RH Cask)	Off-gasing assuming steam/graphite reaction and resuspension factor of 5.00×10^{-6} m^{-1} corresponding to a surface stress from walking (NRC, 1980)
FMRPT	Accident severity 1-2: 0×10^{0}	Type B package design NRC (1977) used as guidance
	Accident severity 3-8: 1×10^{0}	

TABLE D.3.21 CH TRU waste transportation release fractions

Total respirable release fraction (TRRF) = Impact release fraction (IRF) + Thermal release fraction (TRF)

Accident severity category	Impact release fraction[a]	Thermal release fraction[b]	Total respirable release fraction
Truck			
1	0×10^{0}	0×10^{0}	0×10^{0}
2	0×10^{0}	0×10^{0}	0×10^{0}
3	0×10^{0}	8×10^{-9}	8×10^{-9}
4	0×10^{0}	2×10^{-7}	2×10^{-7}
5	8×10^{-5}	2×10^{-7}	8×10^{-5}
6	2×10^{-4}	2×10^{-7}	2×10^{-4}
7	2×10^{-4}	2×10^{-7}	2×10^{-4}
8	2×10^{-4}	2×10^{-7}	2×10^{-4}
Rail			
1	0×10^{0}	0×10^{0}	0×10^{0}
2	0×10^{0}	0×10^{0}	0×10^{0}
3	0×10^{0}	2×10^{-8}	2×10^{-8}
4	0×10^{0}	7×10^{-7}	7×10^{-7}
5	8×10^{-5}	7×10^{-7}	8×10^{-5}
6	2×10^{-4}	7×10^{-7}	2×10^{-4}
7	2×10^{-4}	7×10^{-7}	2×10^{-4}
8	2×10^{-4}	7×10^{-7}	2×10^{-4}

[a] From Table D.3.17.

[b] From Table D.3.19.

TABLE D.3.22 RH TRU waste transportation release fractions

Total respirable release fraction (TRRF) = Impact release fraction (IRF) + Thermal release fraction (TRF)

Accident severity category	Impact release fraction[a]	Thermal release fraction[b]	Total respirable release fraction
<u>Truck</u>			
1	0×10^{0}	0×10^{0}	0×10^{0}
2	0×10^{0}	0×10^{0}	0×10^{0}
3	0×10^{0}	6×10^{-9}	6×10^{-9}
4	0×10^{0}	2×10^{-7}	2×10^{-7}
5	1×10^{-4}	2×10^{-7}	1×10^{-4}
6	1×10^{-4}	2×10^{-7}	1×10^{-4}
7	2×10^{-4}	2×10^{-7}	2×10^{-4}
8	2×10^{-4}	2×10^{-7}	2×10^{-4}
<u>Rail</u>			
1	0×10^{0}	0×10^{0}	0×10^{0}
2	0×10^{0}	0×10^{0}	0×10^{0}
3	0×10^{0}	2×10^{-8}	2×10^{-8}
4	0×10^{0}	7×10^{-7}	7×10^{-7}
5	1×10^{-4}	7×10^{-7}	1×10^{-4}
6	1×10^{-4}	7×10^{-7}	1×10^{-4}
7	2×10^{-4}	7×10^{-7}	2×10^{-4}
8	2×10^{-4}	7×10^{-7}	2×10^{-4}

[a] From Table D.3.17.

[b] From Table D.3.19.

ingestion, the degree of exposure depends on the amount of material retained in the lungs or other organs of the exposed persons.

Airborne transport and diffusion can disperse radioactive materials over large areas. The degree of dispersion is influenced by many factors, such as season (which influences atmospheric turbulence), time of day, degree of cloud cover, land surface features and characteristics, and other meteorological parameters. Dispersed material can expose people in many ways, as shown in Figure D.3.3. The principal effect of gamma-emitting materials is a direct external or internal dose. Material that emits alpha or beta radiation if it is converted to an aerosol and inhaled by people produces the largest consequence. Figure D.3.3 illustrates that radioactive materials can also be incorporated in the food chain. Radiation doses received by the population through the food chain pathway are usually more significant if a continuous release exists.

One of the pathways of note is resuspension. This occurs when deposited particulate material becomes airborne through the action of pedestrians, vehicles, plowing, the wind, etc. The resuspended material then becomes available for inhalation and can deliver an additional dose that accumulates with time.

D.3.3.1.4 Pathways and Exposed Populations. RADTRAN or similar analytical tools can be used to evaluate the radiological impacts of transporting radioactive materials under accident conditions. As input to RADTRAN, the exposure pathways must be identified and the size of exposed populations must be estimated. Transportation accidents may be divided into those accidents in which the shipping containers maintain their integrity and there is no release of radioactive materials, and those accidents in which the integrity of the shipping containers is compromised. The exposure pathways and the exposed population subgroups are discussed below.

In an accident that does not compromise the containment of the shipping containers, the exposure pathway is limited to direct exposure by penetrating radiation from the intact package. The dose delivered to any member of an exposed population is evaluated in the same manner as the exposure from normal (incident-free) transportation, with adjustments made for the duration of exposure and the distance between the shipment and the exposed individuals. The exposed populations include the truck or rail crew, the occupants of the other vehicle(s) involved in the accident, bystanders and pedestrians, the occupants of nearby buildings, and the members of emergency response crews.

In an accident that results in a failure of the shipping containers and possible release of radioactive material, exposures may result from both nondispersible and dispersible materials.

The exposure pathway from accidents involving shipping containers with nondispersible materials is direct exposure resulting from the loss of shielding of the contents of the containers. Certain radioactive materials are not dispersible because of their chemical or physical form, such as irradiated steel hardware; these materials may nevertheless result in exposure by penetrating radiation. The doses received by exposed individuals are evaluated in the same manner as other direct exposures, with adjustments made

Figure D.3.3
Possible pathways to man from radionuclide release

for increased dose rates resulting from shielding loss as well as exposure time and distance adjustments. The exposed populations are the same as identified above.

Four exposure pathways may result from accidents that cause a release of dispersible radioactive materials:

- Cloudshine: The exposure from cloudshine is the direct *external* dose from the passing cloud of dispersed material. Dispersion depends on the meteorological conditions at the accident scene, as well as the fraction of failed shipping containers and the fraction of released material that becomes airborne.

- Groundshine: The exposure from groundshine is the direct *external* dose from material that has deposited on the ground after being dispersed from the accident site. The degree of deposition depends on the material being deposited (i.e., the rate at which the dispersed material settles out) and the amount of dispersed material available to settle out (i.e., how much material from the original release has dispersed far enough to deposit on the area of interest).

- Inhalation: The exposure from inhalation is the *internal* exposure that results from breathing aerosolized material. Exposure from inhalation depends on the fraction of failed shipping containers, the fraction of material that becomes airborne, the aerosol fraction of respirable size, the radiation dose delivered per curie of radioactivity inhaled, the dilution factor for radioactive material in the surrounding air, and the breathing rate of the exposed individual.

- Resuspension: The exposure from resuspension is the *internal* exposure that results from the inhalation of material that was dispersed, deposited at a distance from the accident scene and then resuspended as an aerosol and inhaled. Exposure from resuspension requires combining the mechanisms of dispersion, deposition and inhalation described above, as well as estimating the fraction of deposited material that is resuspended. (Resuspension may result from changing weather conditions, such as changes in wind speed or direction, or from disturbing deposited material by other means, such as traffic through a deposition area.) Note that exposure by ingestion is not included in evaluating the radiological impacts of accidents because it is assumed that emergency response and governmental authorities would intervene to impound foodstuffs, provide an alternative water supply, and clean up contaminated land.

The population subgroups that are exposed by an accident that results in dispersion of radioactive material include the individuals who are directly exposed at the scene of the accident and the individuals who are present in the areas over which dispersion occurs.

D.3.3.2 **Results of the Accident Analysis**

The radiological exposures associated with truck or rail accidents involving CH TRU waste are expressed as the exposure per shipment and as a cumulative exposure over the shipping campaign for the alternative being considered. The exposure is the sum of the products of the probability of a given severity accident times the consequences of such an accident for each of the severity categories. The radiological exposures from an accident involving CH TRU waste are expressed in equivalent whole body dose and are tabulated in units of person-rem, and assume three TRUPACT-II containers per truck shipment and six TRUPACT-II containers per rail shipment. Table D.3.23 presents the exposure per shipment for each facility that ships CH TRU waste and the total per shipment exposure for all facilities for truck and rail modes. Table D.3.24 presents the cumulative exposure for all facilities that ship CH TRU waste to the WIPP. This table shows the estimated radiological exposures for transportation accidents in the Proposed Action, which consists of the Test Phase (10 percent of CH TRU waste shipped and all shipments by truck) and the Disposal Phase, in which truck or rail could be used.

No radiological exposures from transportation accidents were calculated for the No Action Alternative because no shipments to the WIPP would be made.

For the Alternative Action, the radiological exposures from truck accidents are the sum of the exposures from the Test Phase and Disposal Phase (Table D.3.24). These exposures would be incurred in a continuous 20-year period after an approximate 5-year Test Phase during which no waste would be shipped to the WIPP but during which approximately seven truck shipments of CH TRU waste would be made from the Rocky Flats Plant to the Idaho National Engineering Laboratory to support bin tests. The accident contribution for these shipments was calculated by subtracting the per-shipment radiological exposure from accidents (Table D.3.23) for a shipment from the Idaho National Engineering Laboratory to the WIPP from that for a shipment from the Rocky Flats Plant to the WIPP. This difference, which represents the Idaho-to-Rocky Flats transportation segment, was multiplied by the number of shipments to arrive at the transportation exposures from the bin tests. Thus, an accident contribution of approximately 5.90×10^{-4} person-rem is expected from the bin test shipments. The radiological exposures from rail accidents for the Proposed Action and the Alternative Action are shown in Table D.3.25.

The radiological exposures from an accident involving a truck or a railcar carrying RH TRU waste are expressed in equivalent whole body dose and are tabulated in units of person-rem, assuming one RH TRU cask per truck shipment and two RH casks per rail shipment. Table D.3.26 presents the per shipment exposure for each facility that ships RH TRU waste by truck or rail and the total exposures for all facilities. Table D.3.27 presents the cumulative exposure for all facilities that ship RH TRU waste to the WIPP. These lifetime radiological exposures from transportation accidents involving RH TRU waste are shown in Table D.3.27 for a 20-year shipping period. No RH TRU waste shipments would occur during the Test Phase of the Proposed Action or the Alternative Action, and therefore no accident exposures result. The radiological exposures of RH TRU shipments are identical for the Proposed Action and the Alternative Action.

TABLE D.3.23 Per shipment accident radiological exposures of CH TRU waste shipments (person-rem)[a,b,c]

Facility	Nonoccupational accident contribution Truck	Rail
Idaho National Engineering Laboratory	7.9×10^{-4}	5.7×10^{-4}
Rocky Flats Plant	2.0×10^{-4}	1.9×10^{-4}
Hanford Reservation	9.9×10^{-4}	8.9×10^{-4}
Savannah River Site	4.2×10^{-2}	4.0×10^{-2}
Los Alamos National Laboratory	1.3×10^{-3}	[d]
Oak Ridge National Laboratory	4.4×10^{-3}	4.22×10^{-3}
Nevada Test Site	8.9×10^{-6}	[d]
Argonne National Laboratory-East	4.9×10^{-4}	3.5×10^{-4}
Lawrence Livermore National Laboratory	1.9×10^{-4}	2.94×10^{-4}
Mound Laboratory	2.8×10^{-5}	5.4×10^{-7}

[a] Population group exposures per waste shipment are expressed in equivalent whole body dose and are tabulated in units of person-rem.

[b] Values for rail are expressed per railcar shipment.

[c] Population group exposures per waste shipment are presented as a function of the Transport Index (TI), which is defined as the dose rate in mrem/hr at 1 m from the waste package.

[d] No railheads present.

TABLE D.3.24 Lifetime radiological exposures for accidents during transportation of CH TRU waste (person-rem): Proposed Action and Alternative Action[a,c]

Facility	Test Phase[b]	Proposed Action Disposal Phase (20-yr) Truck	Proposed Action Disposal Phase (20-yr) Max. rail	Alternative Action Disposal Phase (20-yr) Truck	Alternative Action Disposal Phase (20-yr) Max. rail
Idaho National Engineering Laboratory	3.2×10^{-1}	2.9×10^{0}	1.0×10^{0}	3.2×10^{0}	1.2×10^{0}
Rocky Flats Plant	1.5×10^{-1}	1.4×10^{0}	6.5×10^{-1}	1.5×10^{0}	7.2×10^{-1}
Hanford Reservation	3.1×10^{-1}	2.8×10^{0}	1.2×10^{0}	3.1×10^{0}	1.4×10^{0}
Savannah River Site	1.1×10^{1}	1.0×10^{2}	4.8×10^{1}	1.1×10^{2}	5.3×10^{1}
Los Alamos National Laboratory[d]	2.7×10^{-1}	2.4×10^{0}	2.4×10^{0}	2.7×10^{0}	2.7×10^{0}
Oak Ridge National Laboratory	1.0×10^{-1}	9.0×10^{-1}	4.3×10^{-1}	1.0×10^{0}	4.8×10^{-1}
Nevada Test Site[d]	7.1×10^{-5}	6.4×10^{-4}	6.4×10^{-4}	7.1×10^{-4}	7.1×10^{-4}
Argonne National Laboratory-East	6.9×10^{-4}	6.2×10^{-3}	2.2×10^{-3}	6.9×10^{-3}	2.4×10^{-3}
Lawrence Livermore National Laboratory	1.8×10^{-2}	1.6×10^{-1}	1.3×10^{-1}	1.8×10^{-1}	1.4×10^{-1}
Mound Laboratory	4.2×10^{-4}	3.8×10^{-3}	3.6×10^{-5}	4.2×10^{-3}	4.0×10^{-5}
Total	1.2×10^{1}	1.1×10^{2}	5.4×10^{1}	1.2×10^{2}	6.0×10^{1}

[a] Population group exposures are calculated by multiplying the exposure/shipment identified in Table D.3.23 by the total number of shipments to the WIPP by truck or rail, as determined from the projection in Table D.3.2.

[b] Test Phase assumes 10% of shipment completed by truck.

[c] Nonoccupational population.

[d] Waste shipments from this facility are limited to truck mode, thus rail exposures are the same as truck exposures.

TABLE D.3.25 Summary of lifetime radiological exposure changes between Proposed Action and Alternative Action: CH TRU accident nonoccupational risk (person-rem)

Facility	Proposed Action Truck	Proposed Action Rail	Alternative Action Truck	Alternative Action Rail
Idaho National Engineering Laboratory	3.2×10^{0}	1.3×10^{0}	3.2×10^{0}	1.2×10^{0}
Rocky Flats Plant	1.5×10^{0}	8.0×10^{-1}	1.5×10^{0}	7.2×10^{-1}
Hanford Reservation	3.1×10^{0}	1.5×10^{0}	3.1×10^{0}	1.4×10^{0}
Savannah River Site	1.1×10^{2}	5.9×10^{1}	1.1×10^{2}	5.3×10^{1}
Los Alamos National Laboratory	2.7×10^{0}	2.7×10^{0}	2.7×10^{0}	2.7×10^{0}
Oak Ridge National Laboratory	1.0×10^{0}	5.3×10^{-1}	1.0×10^{0}	4.8×10^{-1}
Nevada Test Site	7.1×10^{-4}	7.1×10^{-4}	7.1×10^{-4}	7.1×10^{-4}
Argonne National Laboratory-East	6.9×10^{-3}	2.9×10^{-3}	6.9×10^{-3}	2.4×10^{-3}
Lawrence Livermore National Laboratory	1.8×10^{-1}	1.5×10^{-1}	1.8×10^{-1}	1.4×10^{-1}
Mound Laboratory	4.2×10^{-3}	4.6×10^{-4}	4.2×10^{-3}	4.0×10^{-5}
Total	1.2×10^{2}	6.6×10^{1}	1.2×10^{2}	6.0×10^{1}

TABLE D.3.26 Per shipment accident radiological exposures of RH TRU shipments (person-rem)[a,b,c]

Facility	Nonoccupational accident contribution	
	Truck	Rail
Idaho National Engineering Laboratory	1.6×10^{-3}	1.3×10^{-3}
Hanford Reservation	4.34×10^{-5}	4.44×10^{-5}
Los Alamos National Laboratory	3.09×10^{-6}	[d]
Oak Ridge National Laboratory	4.84×10^{-6}	5.21×10^{-6}
Argonne National Laboratory-East	6.4×10^{-6}	5.2×10^{-6}

[a] Exposures to the population per waste shipment are expressed in equivalent whole body dose and are tabulated in units of person-rem.

[b] Values for rail are expressed per railcar shipment.

[c] Exposures to the population per waste shipment are presented as a function of the Transport Index (TI) which is defined as the dose rate in mrem/hr at 1 meter from the waste package. Calculations are based on three TRUPACT-II waste packages per truck and six per railcar shipment.

[d] No railheads present.

TABLE D.3.27 Lifetime radiological exposures for accidents during transportation of RH TRU waste (person-rem): Proposed Action and Alternative Action[a,b]

Facility	100% Truck	Maximum rail
Idaho National Engineering Laboratory	7.8×10^{-1}	3.2×10^{-1}
Hanford Reservation	1.1×10^{-1}	5.4×10^{-2}
Los Alamos National Laboratory[c]	3.1×10^{-4}	3.1×10^{-4}
Oak Ridge National Laboratory	2.2×10^{-2}	1.2×10^{-2}
Argonne National Laboratory-East	1.9×10^{-3}	7.8×10^{-4}
Total	9.1×10^{-1}	3.9×10^{-1}

[a] Population group exposures are calculated by multiplying the exposure/shipment identified in Table D.3.26 by the total number of shipments to WIPP by truck or rail, as determined from the projection in Table D.3.22. Rail occupational exposures resulting from normal transportation include the impact of inspection activities.

[b] Nonoccupational populations.

[c] Waste shipments from the facility are limited to truck mode. Rail exposures are thus the same as the truck exposures.

D.3.4 **RADIOLOGICAL CONSEQUENCES OF BOUNDING CASE TRANSPORTATION ACCIDENT**

D.3.4.1 **Assumptions: Bounding Case Accident**

As discussed in Section 5.0, "bounding case" transportation accident scenarios were developed for this SEIS. These scenarios were used to calculate the impact of very severe accidents in higher population areas along the WIPP-preferred transportation routes. Postulated accidents involved both CH and RH truck and rail shipments using TRUPACT-II containers or RH casks. Based on comments received on the draft SEIS, a revised bounding case accident was calculated based on higher curie content CH waste primarily from Los Alamos National Laboratory, the Savannah River Site, and the Idaho National Engineering Laboratory. In the draft SEIS, calculations assuming average CH waste from the Rocky Flats Plant waste were used because these shipments comprise the majority of the total CH waste shipments. Less likelihood of the current bounding case accidents is expected because the number of shipments of maximally loaded containers (WAC or TRUPACT Payload Compliance Plan limits) are smaller than the number of shipments with average waste loadings. Waste compositions from Los Alamos National Laboratory, Savannah River Site, and the Idaho National Engineering Laboratory were analyzed for CH TRU shipments, and from Hanford and the Idaho National Engineering Laboratory for RH TRU shipments. These waste compositions were scaled up to the maximum total curie content of radionuclides allowed by either the WIPP Waste Acceptance Criteria or the TRUPACT Payload Compliance Plan.

During each accident, all TRUPACT-II containers or RH casks were assumed to be equally breached and subsequently engulfed in fire for two hours (it is estimated that at least 17,000 gallons of fuel would be required to provide sufficient fuel to sustain a two-hour fire). External air/oxygen sources were assumed to be limited (internal combustion is limited) because a major breach of the Type B TRUPACT-II containers or RH casks is not credible. Radioactive contamination and hazardous chemicals were assumed to be evenly distributed throughout the waste volume and 0.02 percent of the hazardous and radioactive particulate materials were postulated to be released in a respirable form (less than 10 micron particle size). Each accident was assumed to occur during a period having very stable atmospheric meteorological conditions, so as to limit dispersion or breakup of the plume and maximize radiation doses and hazardous chemical concentrations.

The accident risk analysis method discussed in Subsection D.3.3 relies on the probabilistic approach in RADTRAN to determine cumulative risks of a series of increasingly less probable but more severe accident scenarios. To determine the accident consequences of the "bounding case" accident scenarios, a probability of 100 percent was specified. The specific conditions assumed for these bounding case accidents are summarized in Table D.3.28.

The probability of breaching all Type B containers or casks during truck or rail accidents and engulfing them in a two-hour fire (requiring the fuel equivalent of two fully loaded fuel transports) in an urban area during adverse meteorological conditions

TABLE D.3.28 Bounding case accident scenario assumptions

The waste shipment is assumed to be three fully-loaded TRUPACT-IIs or 1 RH cask on a combination tractor-trailer truck or six fully-loaded TRUPACT-IIs or two RH casks on a railcar. The origin facilities of the waste shipments are those with the greatest likelihood of having a trailer load of waste with a curie content set at the maximum thermal or fissile gram limits specified by the WIPP Waste Acceptance Criteria or WIPP Payload Compliance Plan.
All waste is packaged in Type A drums.
A major breach of any of the Type B TRUPACT-II containers or RH casks that compose a TRU shipment is not credible, limiting external air/oxygen sources.
Loss of packaging containment will result in .0002 fraction of the radioactive waste material in the TRUPACT-II containers or RH casks being released to the environment in a respirable form. These respirable materials are airborne particulates and aerosols, which are all less than 10 microns aerodynamic diameter in size.
Radioactive contamination is evenly distributed throughout the waste volume.
The highest accident severity category, category eight, is assumed, with a fire duration of two hours.
All TRUPACT-II containers or RH casks on the trailer or railcar are equally breached.
The accident occurs in the urban or suburban portion of a nonspecific large (greater than one million population) metropolitan area with a mean population density of 3,861 persons (urban) or 719 persons (suburban) per square kilometer in the subarea immediately surrounding the accident site.
An aerosol cloud of respirable radionuclides is dispersed downwind.

is very small. The probability would be a small fraction of the fraction, $0.05 \times 1.5 \times 10^{-5}$ for a truck shipment or a small fraction of $0.05 \times 1.0 \times 10^{-5}$ for a rail shipment (Tables D.3.15 and D.3.16). Additional conservatism in the analysis included the use of a range of population densities higher than currently exist along most WIPP transportation corridors, including Atlanta, Georgia; Denver, Colorado; and Albuquerque, New Mexico.

These conditions were input to the RADTRAN computer code to determine radiological consequences of these bounding cases. These radiological consequences measure the potential to cause immediate and delayed health effects in the affected population, including early fatalities, early morbidities, latent cancer fatalities, and genetic effects from the inhalation, resuspension, groundshine, and cloudshine of the aerosol cloud of the released radionuclides. As a check on estimated consequences, each bounding case scenario was also analyzed with the AIRDOS model. A comparison or RADTRAN and AIRDOS parameters for CH and RH bounding cases is shown in Tables D.3.29 and D.3.30.

D.3.4.2 **Results: Bounding Case Accident**

The RADTRAN and AIRDOS codes were used to predict the consequences of the bounding case accident scenarios. As previously discussed, health impacts may result from external exposure (e.g., cloudshine, groundshine) and internal exposure (e.g., inhalation, resuspension, and ingestion) to the dispersed radioactive material. Since it was assumed that the accidents occurred in an urban or suburban area, ingestion impacts associated with contamination of agricultural products were not applicable.

The analysis assumed that stable to extremely stable atmospheric conditions predominated. This assumption conservatively predicted high airborne radioactive contaminant concentrations and limited the dispersion of the contaminants to outlying areas. In an urban area, surface irregularities and thermal anomalies will tend to preclude the probability of a prevailing stable atmospheric condition.

The revised results of the bounding case accident analyses are presented in Tables D.3.31 through D.3.34 for CH and RH truck and rail scenarios. Contributions to the total committed effective dose equivalent (CEDE) for the exposed population from various pathways (initial inhalation, inhalation from resuspension processes, groundshine, cloudshine) are shown as calculated by both RADTRAN and AIRDOS. The dose expected for the maximally exposed individual as directly calculated by AIRDOS is also shown for each scenario. Population doses were converted to estimates of health effects (latent cancer fatalities) using a conversion factor of 1 person-rem = 2.8×10^{-4} LCFs.

For all the scenarios analyzed, neither RADTRAN nor AIRDOS estimated any early fatalities or morbidities. The estimated population doses were dominated by inhalation contributions (initial or from resuspension processes). Two values for the resuspended inhalation dose contribution were calculated using RADTRAN. These values were calculated using resuspension particle half-lives of 365 and 60 days and are designated

TABLE D.3.29 CH bounding case accident inputs

Input factor	RADTRAN III	AIRDOS
Curies per TRUPACT-II	Same for each model	Maximum allowed per thermal or fissile grams limits set by WAC or Payload Compliance Plan: LANL 1080 PE-Ci[a] (7170 total Ci) SRS 1100 PE-Ci (3750 total Ci) INEL 1200 PE-Ci (6540 total Ci)
Release fraction	.0002 released of all Ci as airborne, respirable fraction for both models	
Release height	Ground release	Ground release (3.5 meters)
Weather	Same, Stability Class F for both models	
Wind speed	1 meter per second	2 meters per second
Population density	Same for both models (Urban: 3861 people per square kilometer Suburban: 719 people per square kilometer)	
Directly calculated Pathway doses	Inhalation	Inhalation
	Resuspension	-----------
	Groundshine	Groundshine
	Cloudshine	Cloudshine
	Ingestion	----------
Calculation of "Maximum Individual" Directly	No	Yes

[a] PE-Ci is plutonium equivalent curies calculated using weighting factors in Appendix F.

in the tables as Resusp. I and Resusp. II, respectively. The resuspension half-life is the required time for half of the initially deposited material to be removed from the accessible environment (i.e., at this point, half of the initially deposited material is still available for resuspension). Because inhalation of resuspended particles is a major contributor to the estimated population dose, variation of the resuspension half-life can significantly affect the total calculated dose as shown in the tables. A resuspension half-life of 365 days is extremely conservative given washing (rain) and weathering (wind) processes which would serve to remove contaminants from the accessible environment. The assumed population density also affects the total calculated dose and estimated health effects as shown by comparing results of Los Alamos National Laboratory bounding case accidents occurring in either urban or suburban population zones (Table D.3.31).

For CH truck shipments, depending on shipment origin facility and using a resuspension half-life of 365 days, the total population doses as calculated by RADTRAN and AIRDOS ranged from 6,550 person-rem (1.8 LCFs) to 180,000 person-rem (50 LCFs). Using a 60-day resuspension half-life, the population doses ranged from 6,550 person-rem (1.8 LCFs) to 55,800 person-rem (15.6 LCFs). The estimated maximum individual doses ranged from 160 mrem to 180 mrem depending on shipment origin site.

Results for CH rail shipments were twice those calculated for truck shipments for those facilities with rail access (Savannah River Site and the Idaho National Engineering Laboratory) because a rail shipment involves twice the number of TRUPACT-II containers as a truck shipment.

For RH truck shipments, depending on shipment origin facility and assuming a resuspension half-life of 365 days, the total population doses as calculated by RADTRAN or AIRDOS ranged from 899 person-rem (.25 LCFs) to 40,100 person-rem (11.2 LCFs). For a 60-day resuspension half-life, population doses ranged from 899 person-rem (.25 LCFs) to 12,400 person-rem (3.5 LCFs). The estimated maximum individual doses ranged from 4 mrem to 40 mrem depending on shipment origin facility.

As for CH shipments, results for RH rail shipments were twice those estimated for RH truck shipments because a rail shipment involves two RH casks, whereas a truck shipment involves one RH cask.

D.3 TRANSPORTATION RISKS

D.3.1 **INTRODUCTION**

This section presents an analysis of the risks involved in shipping CH and RH TRU waste to the WIPP. These risks fall into two general categories: radiological risks and nonradiological risks, and each of these categories can be further divided into risks incurred from transportation under normal conditions and from transportation accidents.

This analysis of transportation risks was conducted in a manner similar to other risk assessments, including the WIPP FEIS, using the methodology established by the NRC in studies done in the late 1970s. Although computer models and basic assumptions have been refined since these studies, the basic approach to assessing risk remains essentially the same. The primary reason for this stability of research methods is that this approach has proved to be accurate and reliable.

The analytical models or codes used in this analysis have been extensively documented elsewhere (Peterson, 1984; Joy et al., 1982; NRC, 1977; Taylor and Daniel, 1977; AEC, 1972). The code used to calculate radiological risks was RADTRAN II (Taylor and Daniel, 1982), a revision of the RADTRAN code (Taylor and Daniel, 1977). This code is the product of almost 15 years of development and is a flexible analytical tool for calculating the impacts of both normal transportation and transportation accidents.

The initial RADTRAN code and its subsequent versions have been used to prepare a number of key risk assessment documents, including the environmental assessment used in hearings held by the Interstate Commerce Commission on the issue of shipping radioactive materials by special-use trains; the Final Environmental Impact Statement on the Transportation of Radioactive Material by Air and Other Modes (NRC, 1977); the shipping risk analysis presented in the WIPP FEIS; and subsequent environmental and technical documentation for shipping TRU waste to the WIPP.

The RADTRAN model continues to be modified and refined; even at the present time changes are being made to the code. However, the versions of RADTRAN used in this SEIS have been validated by extensive use and assessment.

The major revisions to RADTRAN II from the earlier RADTRAN version used in the FEIS include the following:

Incident-Free Model (Transportation Under Normal Conditions)

- Shielding options in urban and suburban areas
- Checks for regulatory consistency
- Addition of rail crew doses
- Inclusion of rail travel through urban areas
- Revision of dose-while-stopped model
- Three package-size discriminators for handlers

- Pedestrian dose evaluated in cities

Accident Model

- Groundshine dose evaluated
- Cloudshine dose evaluated
- Economic impacts included
- Early morbidities evaluated
- Genetic effects evaluated
- Building dose factors included
- Inclusion of urban pedestrian inhalation dose
- Addition of Pasquill stability category option
- Expanded material dispersibility classes

General

- Redesign of input and output

Incident-free radiological risks occur during routine transportation and are the result of public and worker exposures to direct radiation at levels allowed by transportation regulations. While radiation shielding is incorporated into package designs where needed in accordance with DOT and NRC regulations, workers, vehicle crew members, and the public along the transportation routes will be exposed to very low dose rates of direct radiation from the packages during incident-free transportation. These low doses usually fall below the threshold of natural background radiation.

In the case of transportation accidents, radiological risks could be incurred if any radioactive material is released into the environment and is spread by winds or possibly through the plume of a fire that occurs during the accident. Since TRU waste emits primarily nonpenetrating (i.e., will not penetrate the skin) radiation, the released material must be either inhaled or ingested in order to present an immediate health hazard.

In order to evaluate the radiological risks of accidents, it is necessary to do a probabilistic analysis--that is, to consider the probability of an accident occurring and the potential consequences of that accident. This analysis includes the following steps:

1) a description of the physical, chemical, and radiological characteristics of the waste
2) a system description (types of shipping containers, number of containers per shipment, etc.)
3) an identification of potential accident scenarios in which radioactive material may be released
4) a probability to be assigned to the release scenarios
5) an estimate of the amount and type of material released in each scenario (the release fraction)
6) an evaluation of consequences, most often in terms of radiation exposure to the worker and the public.

In addition, a credible probabilistic evaluation of the radiological risks of accidents must include variations in transportation routes, population density along the routes and weather characteristics that could affect the results.

In the RADTRAN transportation accident model, the consequences of accidents are apportioned among eight severity categories and calculated for truck and rail transport (see Tables D.3.15 and D.3.16). Each severity category is associated with a release fraction and probability of occurrence. These categories are related to fire and mechanical forces expected in an accident, but specific accident scenarios are not described for the severity categories. The model for calculating release combines the fraction of material that is released from the shipping container with the fraction of material that becomes airborne and the fraction of the released material that is of respirable size. These latter fractions are based on the characteristics of the waste and the mechanisms by which the release occurs.

For this analysis, an average release fraction for each severity category was estimated, and the shipping containers were assumed to respond the same way in an accident regardless of the waste contents or waste form. It was further assumed that there would be no release for accidents assigned to severity category one or two, which a Type B shipping container or cask (e.g., TRUPACT-II or RH cask) must survive intact in order to be certified by the NRC.

Releases from crush impacts were expected to be limited to the Type A containers (55-gal drums/standard waste boxes) only and those to be limited to the interior of the TRUPACT-II containers with no subsequent release for accidents below severity category six. Releases from the TRUPACT-II were assumed to be possible during accidents involving fires in category three or above. The release fractions were increased for each succeeding severity category. The release fractions for each severity category were combined with the accident rates for each category, the probability of a fire or impact event, the travel distance per shipment, and the fraction of travel through each population density zone to determine a cumulative, probability-weighted consequence for each shipment in terms of radiation doses.

To complement the radiological incident-free and probabilistic accident risk analysis, bounding case accidents were postulated and their radiological consequences analyzed. These accidents were assumed to occur under conditions which maximized, within reasonable bounds, the consequences to exposed population groups.

In addition to the analyses of transportation radiological risks, an analysis was conducted of the nonradiological risks associated with projected shipments of TRU waste. These risks include potential injuries and fatalities along the truck and rail routes from accidents that are unrelated to the cargo and are based on historical injury and fatality rates for truck and rail traffic. These risks also include the exposure of populations along the routes to vehicle emissions from the TRU truck and rail shipments.

Although the transportation of TRU waste cannot be made entirely risk free, with reasonable planning and control, risks can be reduced to a level usually below that of comparable shipments (e.g., commercial shipments of hazardous materials such as gasoline) on the nation's transportation routes.

A more complete picture of how various components of the transportation system fit together to provide reliability and ensure the safety of the TRU waste shipping campaign is provided when Appendix C, Appendix L, and Appendix M are reviewed in conjunction with this appendix.

- Appendix C discusses emergency response training, procedures, and plans for the WIPP shipping campaign.

- Appendix L discusses the design, certification, and operation of the TRUPACT-II shipping container for CH TRU waste and the NuPac 72B shipping cask for RH TRU waste.

- Appendix M summarizes the trucking contract, including qualifications standards and training requirements for drivers, and quality assurance standards applicable to operational activities.

The approach to the transportation of TRU waste continues to be based on proven and safe practices established in transporting this waste to retrievable storage facilities at several sites over the last 20 years. These transportation practices are enhanced by the training, certification, regulatory compliance, safety, and quality assurance procedures discussed in the above-cited appendices.

D.3.2 **INCIDENT-FREE RISKS**

D.3.2.1 **Method for Calculating Radiological Risks from Normal Transportation**

The analysis of incident-free radiological risks began with an estimate of the volumes and characteristics of the waste to be transported. As discussed in more detail in Appendix B, the volumes of waste currently in storage and projected to be generated through the year 2013 were estimated from the 1987 Integrated Data Base (ORNL, 1987). These volumes were scaled-up to the maximum amount of waste that could be emplaced at the WIPP (approximately 6.45 million ft^3) and are shown in Table D.3.1. The analysis assumed that for truck shipments CH TRU waste would be packaged in Type A 55-gallon drums and transported in TRUPACT-II shipping containers, with each TRUPACT-II carrying two 7-packs of drums and 3 TRUPACT-II containers or 42 drums, per shipment. RH TRU waste was assumed to be transported in RH casks (one cask per shipment). For these conditions, the number of shipments to the WIPP was calculated as shown in Table D.3.2. For rail shipments, six TRUPACT-II containers on a single railcar constitute a CH shipment, and two RH casks on a railcar constitute an RH shipment.

For incident-free shipments, important waste characteristics include the radionuclide composition of the waste and the total amount (curies) of each radionuclide transported

TABLE D.3.2 Projected number of CH TRU and RH TRU waste shipments from generator and storage facilities to the WIPP

Facility	Number of shipments 100% Truck	Maximum rail
Contact-Handled[a,b]		
Idaho National Engineering Laboratory	4046	2023
Rocky Flats Plant	7608	3804
Hanford Reservation	3103	1552
Savannah River Site	2640	1320
Los Alamos National Laboratory	2065	2065[c]
Oak Ridge National Laboratory	228	114
Nevada Test Site	80	80[c]
Argonne National Laboratory-East	14	7
Lawrence Livermore National Laboratory	969	485
Mound Laboratory	150	75
TOTAL	20903	11525
Remote-Handled[d]		
Idaho National Engineering Laboratory	487	244
Hanford Reservation	2470	1235
Los Alamos National Laboratory	101	101[c]
Oak Ridge National Laboratory	4605	2303
Argonne National Laboratory-East	300	150
TOTAL	7963	4033

[a] Shipments based on 3 TRUPACT-IIs per truck shipment and 6 TRUPACT-IIs per railcar shipment.

[b] Truck shipments calculated from a drum volume of 0.2 m^3/drum x 14 drums/TRUPACT-IIs x 3 TRUPACT-IIs/Truck.

Rail shipments from a drum volume of 0.2 m^3/drum x 14 drums/TRUPACT-IIs x 6 TRUPACT-IIs /Railcar.

[c] Los Alamos National Laboratory and Nevada Test Site do not have access to rail, thus truck shipments are included in the maximum rail case.

[d] Truck shipments calculated from a NuPac 72B volume of 0.89 m^3/NuPac 72B x 1 NuPac 72B/Truck.

Rail shipments calculated from a NuPac 72B volume of 0.89 m^3/NuPac 72B x 2 NuPac 72B/Railcar.

per shipment. Using the waste volumes presented in the 1987 Integrated Data Base, and the information on waste characteristics provided by the facilities, the radioactivity characteristics of average truck or rail shipments of TRU waste from each of the sites were determined and are shown in Table D.3.3 for CH TRU waste and Table D.3.4 for RH TRU waste. Site-specific values of the Transport Index (TI) for a typical shipment of CH and RH TRU waste were developed by the WIPP and generator/storage site personnel. The TI represents the radiation dose rate at 1 meter (3.28 ft) from the surface of the shipping container (TRUPACT-II with a load of 14 drums of waste or an RH cask) and depends on waste density, distribution of radionuclides, quantity of radionuclides per shipment, mix of waste types, self-shielding provided by the waste, and shielding provided by the TRUPACT-II container or RH cask. The TI is very sensitive to small quantities of gamma-emitting fission products such as Cobalt-60 and Cesium-137. TI values for typical shipments from each facility are shown in Table D.3.5. The radiation dose rate represented by the TI was used to calculate radiation exposures of occupational populations (i.e., crew, shipment inspectors, waste handlers) and nonoccupational populations (people living or traveling along shipment routes, and people in the vicinity of the shipment while it is stopped). These TI values are very conservative (see Appendix B) in that they were based on two key assumptions: 1) the maximum drum surface dose rates as measured by the facilities and 2) a drum source term and energy of 1 MeV. A more typical source term energy would be 0.06 to 0.1 MeV_E for CH TRU waste.

In the RADTRAN model, the people living along shipment routes were classified into urban, suburban, and rural fractions with respective population densities of 3,861, 719, and 6 persons per square kilometer as specified by the NRC (1977). These population densities are quite typical of urban, suburban, and rural environments. For example, statistics from the Denver Regional Council of Governments show that along Interstate 25 through Denver only a small area around downtown Denver has a population density exceeding the urban figure used in RADTRAN (3,997 persons per square kilometer for Denver versus the 3,861 assumed by RADTRAN). Other segments through Denver have much lower population densities than the RADTRAN urban value. Fifteen miles south of downtown, population densities along I-25 approach the rural value of six persons per square kilometer.

For truck shipments, the HIGHWAY model (Joy et al., 1982) was used to estimate trip lengths from various facilities to the WIPP and the corresponding population density fractions along these routes. The routes selected generally follow interstate highways as specified by the DOT for shipments of route-controlled quantities of radioactive materials. For rail shipments, the INTERLINE model (Peterson, 1984) was used to estimate trip lengths and population density fractions. The selected routes follow Class A/Class B main lines. These distances and population density fractions are summarized in Table D.3.6. Other major input parameters to RADTRAN are summarized in Table D.3.7.

D.3.2.2 **Results of the Analysis**

The radiation exposures that would be received from the normal transportation of CH and RH TRU waste by truck and rail are shown in Tables D.3.8 and D.3.9. These exposures are summarized for both occupational and nonoccupational populations. The radiological exposures are presented on a per-shipment basis for each facility and are given in doses (person-rem) received by the exposed population for each shipment. These per-shipment exposures were used to calculate the total incident-free transportation exposures for the Proposed Action and the two alternatives (see Table

TABLE D.3.4 Average radioactivity in a shipment of RH TRU waste[a]

Radionuclide	Waste facility[b] ANLE	HANF	INEL	LANL	ORNL
Cobalt-60	0.00×10^{0}	2.97×10^{0}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}
Strontium-90	0.00×10^{0}	6.76×10^{0}	4.08×10^{0}	7.99×10^{0}	1.12×10^{0}
Ruthenium-106	0.00×10^{0}	1.89×10^{-3}	0.00×10^{0}	6.31×10^{0}	0.00×10^{0}
Antimony-125	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	1.95×10^{-1}	0.00×10^{0}
Cesium-137	8.83×10^{0}	9.46×10^{0}	5.81×10^{0}	6.18×10^{0}	4.42×10^{-2}
Cerium-144	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	6.22×10^{1}	0.00×10^{0}
Europium-155	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	3.13×10^{-1}	0.00×10^{0}
Thorium-232	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}
Uranium-233	0.00×10^{0}	5.41×10^{-4}	0.00×10^{0}	0.00×10^{0}	4.56×10^{-3}
Uranium-234	0.00×10^{0}	8.11×10^{-5}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}
Uranium-235	1.21×10^{-5}	2.43×10^{-6}	8.68×10^{-2}	9.48×10^{-5}	1.87×10^{-6}
Uranium-238	0.00×10^{0}	5.41×10^{-5}	2.46×10^{-2}	0.00×10^{0}	1.96×10^{-6}
Neptunium-237	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}
Plutonium-238	0.00×10^{0}	9.73×10^{-2}	1.63×10^{-2}	0.00×10^{0}	1.18×10^{-3}
Plutonium-239	2.52×10^{-1}	1.38×10^{0}	8.80×10^{1}	8.29×10^{-1}	3.67×10^{-2}
Plutonium-240	9.27×10^{-2}	4.05×10^{-1}	3.58×10^{1}	2.73×10^{-1}	0.00×10^{0}
Plutonium-241	0.00×10^{0}	8.11×10^{0}	0.00×10^{0}	1.26×10^{1}	0.00×10^{0}
Plutonium-242	0.00×10^{0}	8.65×10^{-5}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}
Americium-241	0.00×10^{0}	5.95×10^{-1}	3.27×10^{-3}	0.00×10^{0}	1.88×10^{-2}
Curium-244	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	1.69×10^{-1}
Californium-252	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	2.91×10^{-1}
TOTAL	9.18×10^{0}	2.98×10^{1}	1.34×10^{2}	9.68×10^{1}	1.68×10^{0}

[a] Radioactivity in curies per shipment for the volumes of waste assumed for the SEIS analyses (i.e., volumes scaled up to correspond to the design capacity of the WIPP--see last column, Table B.2.4). The volume per shipment is 0.89 m^3 (one shipping cask per shipment).

[b] Key: ANLE, Argonne National Laboratory--East; HANF, Hanford Reservation; INEL, Idaho National Engineering Laboratory; LANL, Los Alamos National Laboratory; ORNL, Oak Ridge National Laboratory.

TABLE D.3.5 Transport index values[a]

Facility	CH TRU waste	RH TRU waste
Idaho National Engineering Laboratory	1.0	5.0
Rocky Flats Plant	1.5	[b]
Hanford Reservation	0.7	16.0
Savannah River Site	2.7	[b]
Los Alamos National Laboratory	4.1	8.9
Oak Ridge National Laboratory	11.0	3.2
Nevada Test Site	1.2	[b]
Argonne National Laboratory-East	7.5	2.5
Lawrence Livermore National Laboratory	0.4	[b]
Mound Laboratory	0.4	[b]

[a] mrem/hr at 1 meter from transporter surface.
[b] Blanks = RH TRU waste not stored at facility.

TABLE D.3.6 Average distances to the WIPP and percent of travel in various population zones[a]

	Average distance	Population zone		
	Miles	R	S	U
Truck				
Idaho National Engineering Laboratory	1521	85.0	13.8	1.2
Rocky Flats Plant	874	82.3	15.7	2.0
Hanford Reservation	1913	85.7	13.4	0.9
Savannah River Site	1585	74.3	25.1	0.6
Los Alamos National Laboratory	343	90.1	9.9	0.0
Oak Ridge National Laboratory	1350	78.6	20.7	0.7
Nevada Test Site	1286	86.8	11.2	2.0
Argonne National Laboratory-East	1387	78.1	21.8	0.1
Lawrence Livermore National Laboratory	1458	86.2	10.1	3.7
Mound Laboratory	1472	75.4	24.1	0.5
Rail				
Idaho National Engineering Laboratory	1761	89.5	9.8	0.7
Rocky Flats Plant	1098	86.7	11.6	1.7
Hanford Reservation	2296	87.8	11.5	0.7
Savannah River Site	1915	76.0	22.4	1.6
Oak Ridge National Laboratory	1630	79.8	18.9	1.3
Argonne National Laboratory-East	1469	81.6	17.0	1.4
Lawrence Livermore National laboratory	1873	85.0	14.3	0.8
Mound Laboratory	1677	76.8	21.3	1.9

[a] Mean population densities are utilized and correspond to:
R = Rural (6 persons/km^2)
S = Suburban (719 persons/km^2)
U = Urban (3861 persons/km^2).

Source: Madsen et al., 1983.

TABLE D.3.7 RADTRAN general input data[a]

Parameter	CH TRU waste Truck	CH TRU waste Rail	RH TRU waste Truck	RH TRU waste Rail
Package type	TRUPACT-II		Cask	
Package waste volume, m^3	2.8	2.8	1.0	1.0
Packages/shipment	3	6	1	2
Transport Index (TI), mrem/hr	(Site-specific, see Table D.3.5)			
Package length dimension, m	7.32	7.32	3.61	3.61
Number of crewmen	2	5	2	5
Distance from source to crew, m	4	152	5	152
Speed, km/hr				
Urban population zone	24	24	24	24
Suburban population zone	40	40	40	40
Rural population zone	88	64	88	64
Stop time per kilometer, hr/km	.011	.0036	.011	.0036
No. of people exposed while stopped	50	100	50	100
No. of people per vehicle	2	3	2	3
Population density, people/km^2				
Urban population zone	3861	3861	3861	3861
Suburban population zone	719	719	719	719
Rural population zone	6	6	6	6
Avg. rad./trailer-load of pkgs., Ci	(Site-specific, see Tables D.3.3 and D.3.4)			
Accident release fractions	(See Tables D.3.17 through D.3.22)			

[a] Source: Madsen et al., 1983.

TABLE D.3.8 Radiological exposures per CH TRU shipment (person-rem)[a,b,c]

	Truck		Rail	
Facility	Occupational	Nonoccupational	Occupational[d]	Nonoccupational
Idaho National Engineering Laboratory	5.0×10^{-2}	2.0×10^{-2}	2.9×10^{-4}	3.0×10^{-2}
Rocky Flats Plant	4.0×10^{-2}	1.0×10^{-2}	2.7×10^{-4}	2.0×10^{-2}
Hanford Reservation	3.9×10^{-2}	2.3×10^{-2}	2.6×10^{-4}	4.0×10^{-2}
Savannah River Site	1.4×10^{-1}	7.0×10^{-2}	8.4×10^{-4}	1.2×10^{-1}
Los Alamos National Laboratory	2.8×10^{-2}	8.0×10^{-3}	[e]	[e]
Oak Ridge National Laboratory	1.3×10^{-1}	2.0×10^{-1}	2.1×10^{-3}	2.0×10^{-1}
Nevada Test Site	5.0×10^{-2}	2.0×10^{-2}	[e]	[e]
Argonne National Laboratory-East	1.3×10^{-1}	1.4×10^{-1}	1.8×10^{-3}	1.9×10^{-1}
Lawrence Livermore National Laboratory	1.7×10^{-2}	9.0×10^{-3}	1.2×10^{-4}	1.6×10^{-2}
Mound Laboratory	1.9×10^{-2}	9.0×10^{-3}	1.1×10^{-4}	1.4×10^{-2}

[a] Exposures per waste shipment are expressed in equivalent whole body dose and are tabulated in units of person-rem.

[b] Values for rail are expressed per railcar shipment.

[c] Exposures per waste shipment are presented as a function of the Transport Index (TI) which is defined as the dose rate in mrem/hr at 1 meter from the waste package. Calculations are based on three TRUPACT-IIs per truck and six per railcar.

[d] Rail occupational exposures resulting from normal transportation include the impact of DOT inspection activities (.01 X Total Stop Time (hr) X TI).

[e] No railheads present.

TABLE D.3.9 Radiological exposures per RH TRU shipment (person-rem)[a,b,c]

	Truck		Rail	
Shipment origin facility	Occupational	Nonoccupational	Occupational[d]	Nonoccupational
Idaho National Engineering Laboratory	1.0×10^{-1}	8.0×10^{-2}	1.3×10^{-3}	1.3×10^{-1}
Hanford Reservation	1.7×10^{-1}	3.3×10^{-1}	3.5×10^{-3}	2.9×10^{-1}
Los Alamos National Laboratory	2.8×10^{-2}	1.2×10^{-2}	[e]	[e]
Oak Ridge National Laboratory	6.3×10^{-2}	4.4×10^{-2}	7.7×10^{-4}	7.4×10^{-2}
Argonne National Laboratory-East	5.0×10^{-2}	4.0×10^{-2}	5.5×10^{-4}	5.0×10^{-2}

[a] Exposures per waste shipment are expressed in equivalent whole body dose and are tabulated in units of person-rem.

[b] Values for rail are expressed per railcar shipment.

[c] Exposures per waste shipment are presented as a function of the Transport Index (TI) which is defined as the dose rate in mrem/hr at 1 meter from the waste package. Calculations are based on three TRUPACT-IIs per truck and six per railcar.

[d] Rail occupational exposures resulting from normal transportation include the impact of DOT inspection activities (.01 X Total Stop Time (hr) X TI).

[e] No railheads present.

D.3.10). The Proposed Action corresponds to an approximate 5-year Test Phase period during which up to 10 percent of the waste would be shipped to the WIPP by truck and a subsequent 20-year Disposal Phase during which the remainder of the waste would be shipped by either truck or rail. Cumulative exposures for the entire campaign in the Proposed Action are the sum of the total exposures from the Test Phase (truck shipments) and Disposal Phase (truck or rail shipments). The No Action Alternative does not involve transportation to the WIPP and therefore has no radiological exposures from transportation.

The Alternative Action also includes an approximate 5-year Test Phase during which approximately 300 drums of CH TRU waste would be shipped from the Rocky Flats Plant to the Idaho National Engineering Laboratory for bin storage tests. This would require approximately seven truck shipments with three TRUPACT-II containers per shipment. Assuming a per-shipment incident-free exposure which is the ratioed difference (based on Transport Index) between the per-shipment exposures for the Idaho National Engineering Laboratory to the WIPP and the Rocky Flats Plant to the WIPP (see Table D.3.8), the estimated occupational and nonoccupational incident-free exposures from these shipments are 0.035 person-rem and 0.02 person-rem, respectively.

Tables D.3.11 and D.3.12 summarize the differences between the Proposed Action and the Alternative Action in the radiological exposure to occupational and nonoccupational populations from transporting CH TRU waste under normal conditions.

Table D.3.13 shows the lifetime radiological exposure of transporting RH TRU waste under normal conditions during the Disposal Phase of either the Proposed Action or the Alternative Action. No RH TRU waste would be shipped during the Test Phase for either the Proposed Action or the Alternative Action. However, if RH TRU waste is shipped to the WIPP during the Test Phase, the lifetime radiological exposures would be spread over more than the 20 years assumed for the Disposal Phase.

Doses to maximally exposed individuals in various population groups over the 25-year shipping campaign (Test Phase and Disposal Phase) for the Proposed Action are presented in Table D.3.14. Two sets of dose tabulations are provided: one for 100 percent truck shipments and one for maximum rail. The totals represent the dose expected for an individual whose residence or occupation results in an exposure to all or a large number (depending on exposure group) of waste shipments. For the Alternative Action, these maximum individual doses would be identical, except that they would be received over a 20-year period.

Maximum individual doses were determined using the RADTRAN occupational and hypothetical maximum individual exposure models. The doses were adjusted or supplemented by more detailed models to account for individual doses due to inspections, refueling, food stops, rail operations, and traffic congestion. Estimates of individual doses (e.g., exposure duration, distances) for each of these activities were calculated using line source (1/r) or point source ($1/r^2$) approximations. No credit was taken for attenuation of radiation by the air or by any structures between the individual being exposed and the radiation source.

TABLE D.3.11 Summary of lifetime radiological exposures between Proposed Action and Alternative Action: CH TRU incident-free occupational exposures (person-rem)

Facility	Proposed Action Truck	Proposed Action Rail	Alternative Action Truck	Alternative Action Rail
Idaho National Engineering Laboratory	2.0×10^{2}	2.1×10^{1}	2.0×10^{2}	5.9×10^{-1}
Rocky Flats Plant	3.0×10^{2}	3.1×10^{1}	3.0×10^{2}	1.0×10^{0}
Hanford Reservation	1.2×10^{2}	1.2×10^{1}	1.2×10^{2}	4.0×10^{-1}
Savannah River Site	3.7×10^{2}	3.8×10^{1}	3.7×10^{2}	1.1×10^{0}
Los Alamos National Laboratory	5.8×10^{1}	5.8×10^{1}	5.8×10^{1}	5.8×10^{1}
Oak Ridge National Laboratory	3.0×10^{1}	3.2×10^{0}	3.0×10^{1}	2.4×10^{-1}
Nevada Test Site	4.0×10^{0}	4.0×10^{0}	4.0×10^{0}	4.0×10^{0}
Argonne National Laboratory-East	1.8×10^{0}	1.9×10^{-1}	1.8×10^{0}	1.3×10^{-2}
Lawrence Livermore National Laboratory	1.6×10^{1}	1.7×10^{0}	1.6×10^{1}	5.8×10^{-2}
Mound Laboratory	2.8×10^{0}	2.9×10^{-1}	2.8×10^{0}	8.2×10^{-3}
TOTAL	1.1×10^{3}	1.7×10^{2}	1.1×10^{3}	6.5×10^{1}

TABLE D.3.12 Summary of lifetime radiological exposures between Proposed Action and the Alternative Action: CH TRU incident-free nonoccupational exposures (person-rem)

	Proposed Action		Alternative Action	
Facility	Truck	Rail	Truck	Rail
Idaho National Engineering Laboratory	8.1×10^{1}	6.3×10^{1}	8.1×10^{1}	6.1×10^{1}
Rocky Flats Plant	7.6×10^{1}	7.6×10^{1}	7.6×10^{1}	7.6×10^{1}
Hanford Reservation	7.1×10^{1}	6.3×10^{1}	7.1×10^{1}	6.2×10^{1}
Savannah River Site	1.8×10^{2}	1.6×10^{2}	1.8×10^{2}	1.6×10^{2}
Los Alamos National Laboratory	1.6×10^{1}	1.6×10^{1}	1.6×10^{1}	1.6×10^{1}
Oak Ridge National Laboratory	4.6×10^{1}	2.5×10^{1}	4.6×10^{1}	2.3×10^{1}
Nevada Test Site	1.6×10^{0}	1.6×10^{0}	1.6×10^{0}	1.6×10^{0}
Argonne National Laboratory-East	2.0×10^{0}	1.4×10^{0}	2.0×10^{0}	1.3×10^{0}
Lawrence Livermore National Laboratory	8.7×10^{0}	7.9×10^{0}	8.7×10^{0}	7.8×10^{0}
Mound Laboratory	1.4×10^{0}	1.0×10^{0}	1.4×10^{0}	1.0×10^{0}
TOTAL	4.8×10^{2}	4.1×10^{2}	4.8×10^{2}	4.1×10^{2}

TABLE D.3.13 Summary of lifetime radiological exposures for incident-free transportation of RH TRU waste (person-rem): Proposed Action and Alternative Action

	Disposal Phase (20-yr)[a]			
	100% Truck		Maximum Rail	
Facility	Occ[b]	Nonocc[c]	Occ	Nonocc
Idaho National Engineering Laboratory	4.9×10^{1}	3.9×10^{1}	3.2×10^{-1}	3.2×10^{1}
Hanford Reservation	4.2×10^{2}	8.2×10^{2}	4.3×10^{0}	3.6×10^{2}
Los Alamos National Laboratory[e]	2.8×10^{0}	1.2×10^{0}	2.8×10^{0}	1.2×10^{0}
Oak Ridge National Laboratory	2.9×10^{2}	2.0×10^{2}	1.8×10^{0}	1.7×10^{2}
Argonne National Laboratory-East	1.5×10^{1}	1.2×10^{1}	8.2×10^{-2}	7.5×10^{0}
TOTAL	7.8×10^{2}	1.1×10^{3}	9.3×10^{0}	5.7×10^{2}

[a] No RH TRU waste is shipped to the WIPP during the Test Phase for any alternative.

[b] Occupational population-quantifies doses received by transportation crews.

[c] Nonoccupational population.

[d] Population group exposures are calculated by multiplying the exposure/shipment identified in Table D.3.9 by the total number of shipments to WIPP by truck or rail, as determined from the projections in Table D.3.2. Rail occupational exposures resulting from normal transportation include the impact of inspection activities.

[e] Waste shipments from this facility are limited to the truck mode. Rail exposures are thus the same as truck exposures.

Doses to a truck crew member include those received while the shipment is moving and stopped. The RADTRAN model was used to determine the exposure to an individual crew member while the shipment is moving. An exposure distance of 13 ft (4 m) was specified. Doses received while stopped are from inspections every 100 miles, refueling, and food stops. A truck driver, rather than a service attendant, is assumed to refuel the truck. Estimated exposure distances and durations for these activities while stopped are given in Table D.3.14. Depending upon the number of shipments from a facility and the travel time to the WIPP, a truck driver may transport all or only a fraction of the shipments. Hypothetical lifetime maximum crew member exposures are projected to be up to 130 rem for CH TRU waste shipments and up to 180 rem for RH TRU waste shipments. However, any monitored crew member who receives an accumulated dose that approaches 5 rem (the regulatory limit for occupational exposures) in any given year would be reassigned to other duties involving no further exposure.

Exposures to rail crew members while shipments are moving were also calculated using the RADTRAN model, with an exposure distance of approximately 490 ft (150 m). Exposure while stopped for inspections and servicing was estimated assuming a crew member radiation dose rate equal to the Transport Index value received over a duration of 1 percent of the total stop time (.033 hours per kilometer, typical of regular freight shipments).

The maximum individual dose to a railyard handler/serviceman was estimated assuming an average exposure distance of 33 ft (10 m) for a duration of 2 hours and that this person is exposed to approximately 13 percent of CH TRU shipments and 17 percent of RH TRU shipments (allowing for a 10-year career in the same position and three shifts/crew).

Maximum individual occupational exposures resulting from inspecting departing trucks were estimated assuming an exposure distance of approximately 3 ft (1 m) for 30 minutes. As above, it was also assumed that this individual would remain in the same job for 10 years, and that there would be three shifts/crews performing the same tasks. Individual dose commitments were projected to range from 0.0041 to 0.76 rem for CH TRU shipments and 0.063 to 3.3 rem for RH TRU shipments. The lifetime occupational exposure for truck inspections at the WIPP was estimated by summing the individual facility departure values, and resulted in a dose of 2.4 rem for CH TRU shipments and 4.8 rem for RH TRU shipments. The transportation worker performing rail departure inspections would receive the same maximum exposure as the worker inspecting departing truck shipments, since there are only one-half the number of shipments but about twice the inspection effort per shipment.

Estimated doses to an individual performing State safety vehicle inspections were calculated assuming the person would be involved in 20 percent of the inspections with an average exposure distance of approximately 3 ft (1 m). Inspections may occur at the origin facility, upon arrival at the WIPP, or in the corridor States at ports of entry for trucks or classification yards (transfer of railcar to another rail carrier) for rail shipments. To allow for queues, a truck inspection time of 1 hour was used. For individual railcar shipments, an inspection time of 45 minutes was assumed. For truck transportation, maximum lifetime inspection doses of 7.3 and 12 rem were calculated for CH TRU and RH TRU waste shipments. For rail transportation, maximum lifetime exposures of 5.9 rem (CH TRU) and 8.9 rem (RH TRU) were estimated.

The maximum radiation dose to an individual member of the public (off-link) due to waste shipments which travel by his or her residence or workplace was calculated using the RADTRAN model. It was assumed that the individual is exposed to every waste shipment at a distance of approximately 100 ft (30 m). For truck shipments, an additional exposure category (on-link) was evaluated to assess the radiation dose to a person in an adjacent traffic lane for an extended length of time due to traffic congestion. Assuming the individual is present for one 30-minute period in the adjacent traffic lane during the lifetime of the WIPP at an exposure distance of about 3 ft (1 m), individual doses could range from 0.2 to 8 mrem depending on the shipment's origin facility and type of waste (CH TRU or RH TRU).

The maximum individual dose to a member of the public working at a truckstop was calculated to be 480 mrem for CH TRU waste shipments and 980 mrem for RH TRU waste shipments. This assumes a stop duration of 2 hours, with an exposure distance of 65 ft (20 m). This also assumes that the individual is exposed to approximately 13 percent of all CH TRU shipments and 17 percent of all RH TRU shipments arriving at the WIPP (assuming all shipments stop at the same location, that the individual works for 10 years at the truckstop, and there are 3 shifts/crew.). Exposures to individuals employed at truckstops along routes leading from the individual waste origin facilities will be lower, ranging from .83 to 660 mrem, depending on the specific origin facility and type of waste shipped (CH TRU or RH TRU).

The maximum exposure to a member of the public residing near a train terminal was estimated assuming an exposure distance of 660 ft and that the individual is exposed to every railcar shipment for a duration of 20 hours per stop (Wooden, 1986 used for guidance). Lifetime doses of 0.3 rem for CH TRU shipments and 0.42 rem for RH TRU shipments were estimated.

D.3.3 **RADIOLOGICAL RISKS OF TRANSPORTATION ACCIDENTS**

D.3.3.1 **Method for Calculating Radiological Risks of Transportation Accidents**

D.3.3.1.1 Severity Categories. CH TRU and RH TRU shipments to the WIPP will be made in NRC-certified Type B containers (TRUPACT-II and RH cask). The certification standards ensure that these containers will withstand virtually any accident condition without releasing their radioactive contents to the environment. Recently, a 1987 NRC study (Fischer et al., 1987) determined that only 0.6 percent of truck and rail accidents involving Type B containers or casks could cause a radiation hazard to the public. The earlier 1977 NRC study (NRC, 1977) conservatively estimated that approximately 9 percent of all truck accidents and 20 percent of rail accidents involving Type B containers or casks would result in radioactive material releases. Thus, a TRU waste transportation accident that exceeds regulatory criteria and causes the release of a portion of the contents of the shipping container has an extremely small chance of occurring. However, in order to assure bounding estimates of environmental impact, the more conservative accident severity probability statistics from the older 1977 NRC study (NRC, 1977) are considered by RADTRAN to determine the overall, probabilistic transportation radiological risk.

The amount of radioactive material released in an accident depends on the severity of the accident, the characteristics of the waste, and the capabilities of the shipping container. Most accidents are unlikely to cause any release, but very severe accidents (much more severe than conditions represented by NRC certification standards for Type B containers) may cause some of the radioactive materials to be released. Thus, the distribution of accidents according to severity must be determined, in addition to the overall accident rate. In this subsection, the accident severity classification scheme that was used in this assessment is discussed. The distribution of accidents according to severity is presented for truck and rail shipping modes.

Accident severity categories define the seriousness of an accident in terms of mechanical and thermal loads. Many methods can be used to classify accidents in terms of mechanical and thermal parameters. The relevant mechanical parameters may include impact speed, impact force, impact location and orientation, impact surface hardness, and impact puncture characteristics. The thermal characteristics may include flame temperature, fire duration, fire source size and orientation with respect to the container, and heat transfer properties (such as flame emissivity and convection coefficients).

The NRC defined eight accident severity categories for each transportation mode in a study performed to assess the adequacy of regulations for radioactive material transport (NRC, 1977). The first two accident categories were defined to be less serious than the hypothetical accident conditions specified in 10 CFR Part 71 for testing Type B packaging (i.e., shipping containers or casks). These tests simulate very severe transportation accidents, with the packaging sequentially subjected to drop, puncture, thermal, and water immersion tests. Thus, accidents in severity categories 1 and 2 are very unlikely to cause any release to the environment because the shipping containers or casks are designed to withstand them without releasing any of their contents.

The NRC (1977) classification scheme for truck accidents, illustrated in Figure D.3.1, uses crush force and fire duration to determine the seriousness of an accident. The crush force may result from either an internal (e.g., container crushed upon impact by other containers in the load) or static load (e.g., container crushed beneath vehicle). The classification approach used for train accidents is shown in Figure D.3.2. While fire duration is retained as the thermal parameter, the NRC decided to use puncture and impact speed as the mechanical measure of accident severity. This was done because crushing from the impact of other containers in the cargo was considered less relevant for rail shipments.

The assessment used in this SEIS retains the severity classification scheme used by the NRC (1977). In order to place the accident severities into perspective, two accidents representative of categories 1 and 2 are described:

Figure D.3.1
Truck accident severity category classification scheme

Figure D.3.2
Railroad accident severity category classification scheme

In the accident known as the I-80 bridge accident, a tractor-trailer rig was struck by a pickup truck while on an overpass bridge on I-80 near San Francisco, California. The tractor-trailer rig veered into the bridge railing and fell to a soil surface 64 feet below. Fischer et al. (1987) determined that a comparable accident involving a Type B certified container would be within the accident conditions specified for the design of the containers and thus would not be expected to cause any significant release.

A truck accident involving a fire occurred in the Caldecott Tunnel near Oakland, California. The accident resulted from a collision involving a gasoline truck, a bus, and a car. The gasoline truck carried approximately 8,800 gallons of gasoline, which acted as the fire source; a resulting peak flame temperature of 1900°F was estimated. Although it took about 2 hours and 42 minutes to completely extinguish the fire, most of the gasoline burned in less than 40 minutes. Fischer et al. (1987) concluded in that the response of Type B containers to an accident of this type would be within the design capabilities.

For higher accident severities, there is an incremental increase in mechanical and thermal loads. At the highest severity category, impact forces can be 100 times greater than those in category 2, and fire durations can exceed 1.5 to 2 hours. For example, a fire that engulfs a truck shipment in a diameter of 40 feet would require approximately 17,000 gallons of hydrocarbon fuel to burn for 2 hours. This would require the very unlikely event of involving three tanker trucks in the incident because a typical tanker carries approximately 5,000 gallons of hydrocarbons (Wolff, 1984). At a minimum, at least two full 10,000-gallon tanker trucks would need to be involved. For a rail incident, the average fire pool size is 2,000 square feet (50 ft in diameter) (Wolff, 1984); over 27,000 gallons of hydrocarbon fuel would be required to maintain a fire of this magnitude for 2 hours. The large majority of truck (99.90 percent) and rail (99.83 percent) accidents that involve fires, however, last less than 30 minutes (Wolff, 1984). The probability of such accidents diminishes as their severity increases, as already noted.

Table D.3.15 presents the fractional occurrences of truck accidents in each of the eight severity categories. The assessment conducted for this SEIS assumes an overall accident rate of 1.1×10^{-6} accidents per kilometer (NRC, 1977). The fraction of accidents in each population zone relevant to TRU waste shipments to the WIPP is also presented in Table D.3.15.

Table D.3.16 presents the fractional occurrence of train accidents in each of the eight accident severity categories. The overall accident rate is 9.3×10^{-7} railcar accidents per railroad-kilometer, assuming an average train length of 70 cars and an average of 10 cars involved in each accident (NRC, 1977). The more severe accidents are assumed to occur in lower-population-density zones, where travel speeds are higher.

D.3.3.1.2 Release Fractions. The DOE plans to ship TRU waste to the WIPP in Type B shipping containers or casks whose designs are approved and certified by the NRC (see Appendix L). Type B containers or casks are designed and tested to NRC requirements to demonstrate that they are sufficiently strong to withstand very severe accidents, with safety largely independent of the transport vehicle and procedural and other controls on the shipment. Testing as specified by the NRC in 10 CFR 71.73

TABLE D.3.15 Fractional occurrences[a] for truck accidents by accident severity category and population density zone

Accident severity category	Fractional occurrences	Fractional occurrences according to population density zones: Low	Medium	High
I	.55	.1	.1	.8
II	.36	.1	.1	.8
III	.07	.3	.4	.3
IV	.016	.3	.4	.3
V	.0028	.5	.3	.2
VI	.0011	.7	.2	.1
VII	8.5×10^{-5}	.8	.1	.1
VIII	1.5×10^{-5}	.9	.05	.05

[a] Overall accident rate = 1.1×10^{-6} accidents/kilometer.

TABLE D.3.16 Fractional occurrences[a] for train accidents by accident severity category and population density zone

Accident severity category	Fractional occurrences	Fractional occurrences according to population density zones		
		Low	Medium	High
I	.50	.1	.1	.8
II	.30	.1	.1	.8
III	.18	.3	.4	.3
IV	.018	.3	.4	.3
V	.0018	.5	.3	.2
VI	1.3×10^{-4}	.7	.2	.1
VII	6.0×10^{-5}	.8	.1	.1
VIII	1.0×10^{-5}	.9	.05	.05

[a] Overall accident rate = 9.3×10^{-7} railcar accidents/kilometer.

encompasses a range of very severe accident conditions that are applied sequentially to determine cumulative effects; it includes impact (free drop), puncture, thermal, and water-immersion tests.

The 1977 NRC study (NRC, 1977) conservatively estimated that approximately 9 percent of all truck accidents and 20 percent of rail accidents involving Type B containers or casks could result in radioactive material releases. More recently, however, Fischer et al. (1987) determined that only 0.6 percent of truck and rail accidents could cause a radiation hazard to the public. To estimate how much radioactive material could be released to the environment for the very small number of accidents that exceed the containment design capabilities of the Type B containers or casks, a release fraction analysis was performed.

<u>Release Fraction Definition</u>. The release fraction analysis determined how much radioactive material could be released to the environment in a respirable, airborne form after a very severe accident that affects the containment capabilities of the shipping containers or casks. The calculation focused on respirable particle sizes with a mean aerodynamic diameter of less than 10 microns because inhalation is the primary exposure pathway for TRU elements. Particles that are larger will be expelled from the body and consequently are not as significant in estimating health effects. This calculational approach is consistent with existing NRC risk assessments (WASH-1400, NUREG-0170, NUREG/CR-4829).

<u>Method of Calculating Release Fractions</u>. In order to calculate release fractions for very severe accidents, it is necessary to:

- Characterize the radioactive material being transported
- Identify and quantify the response of the shipping containers or casks (loss of containment) to accident conditions
- Identify and quantify the release mechanisms resulting in the escape of radioactive material from the containers or casks to the environment.

This analysis used representative values for parameters where published data and test results are applicable and reasonable, and conservative estimates where uncertainties exist. "Conservative" is used in this discussion to mean using such parameter values that the consequences of potential accidents will be overestimated.

<u>Characterization of the TRU Waste</u>. The radionuclide compositions, quantities, and volumes used in the analysis are based on the waste inventory data and projections presented in Appendix B. As noted in Subsection 2.3.1, the DOE has established criteria and procedures which govern the physical, radiological, and chemical composition of the waste. Physical restrictions require that the waste not be in a free-liquid form and that particulate waste materials be limited to specific levels in accordance with DOE (1989). Transuranic radionuclides are generally present as oxides with concentrations exceeding 100 nanocuries per gram.

<u>Response of Shipping Containers and Casks</u>. If a shipping container or cask is involved in an accident, the extent of damage will depend on the design of the container and the severity of the accident. Accident severity is categorized in terms of mechanical (e.g., impact) and thermal loads. Many methods can be used to classify accidents in terms of mechanical and thermal parameters. The relevant mechanical parameters may include impact speed, impact force,

impact location and orientation, impact surface hardness, and impact puncture characteristics. The thermal parameters may include flame temperature, fire duration, fire source size and orientation with respect to the containers, and heat transfer properties (e.g., flame emissivity and convection coefficients).

The analysis conducted for the SEIS used the accident severity model developed by the NRC (1977) as discussed in the preceding subsection. This model conservatively predicts the frequency of accidents whose severity exceeds Type B package test requirements (accident severity category three through eight).

Because NRC regulations do not require Type B containers to be tested to failure, and because there are no historical data on the response of containers to very severe accidents, certain assumptions were required to estimate the extent of damage sustained by the TRUPACT-II container and the RH cask from accidents in severity categories three through eight. Guidance was obtained from the analysis and test data presented in NRC (1977), Fischer et al. (1987), and Jefferson (1978). The data indicate that a catastrophic failure (e.g., gaping hole, container severed in half) of a Type B container or cask would not be expected for accidents more severe than those in severity category two. Because of margins in the materials of construction (e.g., minimum versus actual rupture stress) and structural design (e.g., absorption of energy by plastic deformation), more likely failures would include the formation of cracks in the side of the container or cask, the failure of the closure seals, or the failure of any valves or penetrations.

To define the response of Type B containers or casks to transportation accidents, the following conservative assumptions were made:

- For shipments of several Type B containers on one transport vehicle, it was assumed that all containers would sustain the same damage. No credit was taken for the mitigating effects of one container shielding the others from impact forces or thermal loadings.

- Two package response states were defined for the shipping container or cask:

  1) No leak path and no release of radioactive material

  2) A leak path is present, allowing the release of <u>all</u> respirable airborne radioactive material present inside the containers.

The second state was postulated even though catastrophic failures are very unlikely. This state is consistent with NRC's position (Fischer et al., 1987) and does not take credit for any processes that will tend to reduce radioactive material releases (e.g., particle settlement, vapor plate-out on interior surfaces, filtration effects along leak path) from the containers.

The response states are influenced by both the mechanical and thermal conditions of the accident. The response to the impact conditions will be largely independent of the thermal conditions, with impact effects immediate and thermal effects delayed. Consequently, the analysts elected to use two components for the response state (one for the impact event and one for the thermal event) for each accident severity category. Both components have two accident response states as defined above.

Once the potential response states for the shipping containers or casks have been defined, it is necessary to assign the appropriate response state components to each accident severity category. As previously noted, there are few data that can be used to determine failure thresholds for transport containers involved in accidents with conditions more severe than NRC certification test requirements. NRC (1977) Model II release fractions (Table 5-8 of reference) were used as a primary guide. From impact test data, the NRC (1977) projected Type B shipping containers for plutonium to have a failure threshold at accident severity category six. With current development programs, more recent container designs (1985) were projected to have an increased failure threshold, corresponding to accident severity category seven. The NRC (1977) also projected Type B casks to have a failure threshold at accident severity category three, with more significant releases occurring at accident severity category five. These projections included effects from both impact and thermal events.

For response to an impact event, a failure threshold corresponding to severity category five was assigned; it corresponds to the more significant release state projected by the NRC (1977) for Type B casks. For response to a thermal event, a failure threshold corresponding to severity category three (an accident with conditions slightly exceeding the NRC's test requirements) was conservatively assigned.

Release Mechanisms. Any release of radioactive material due to a transportation accident would normally progress in two stages: release inside the shipping containers or casks, followed by release to the environment. Releases from the container to the environment were addressed in the preceding discussion of accident response states. The discussion that follows evaluates how much radioactive material would be released into the cavities of the shipping containers or casks.

There are multiple release mechanisms and pathways that may lead to the release of respirable radioactive material into container cavities. Impact release mechanisms include waste container (e.g., a 55-gallon drum or standard waste box) failure, fragmentation of solid waste, particulate suspension, and aerodynamic entrainment of particles. Thermal release mechanisms include heat-induced failures of the waste containers; aerosolization of particles by combustion, gas generation, or the heating of contaminated surfaces; and potential volatilization of radionuclides. Impact and thermal release mechanisms were evaluated by using applicable test data and analyses available in the published literature, as supplemented by conservative assumptions where only limited data exist. It was assumed that all failed waste containers, without regard to waste form or type, release an average amount of material for each accident severity category.

In assessing releases from impact events for each severity category, the following procedure was used:

- Identification of the fraction of failed waste containers inside the shipping container or cask

- Determination of the fraction of radioactive material released from the failed waste containers

- Calculation of the fraction of radioactive material released from the failed waste containers that is aerosolized in a respirable form by the mechanical stress of impact

- Calculation of the fraction of radioactive material released from the failed waste containers that becomes aerodynamically entrained in a respirable form after the loss of containment by the shipping containers and any subsequent depressurization (e.g., TRUPACT-II design pressure of 50 psig).

Studies by Huerta (1983) and Shirley (1983) were used to determine the fraction of failed waste containers. The fractions of radioactive material released from the failed waste containers were conservatively estimated using reports by Huerta (1983) and the NRC (1977) for guidance. The fraction of radioactive material converted to a respirable aerosol from impact stresses was calculated by using a resuspension factor approach. This is an accepted analytical method for predicting airborne concentrations of material above contaminated surfaces. The mechanical action of vigorous sweeping was used to represent the respirable airborne contamination fraction, using data taken from an NRC report (NRC, 1980), for the resuspension factor.

It was judged that this approach would be at least representative, if not conservative, in estimating the release of respirable contaminants by impact stresses.

The aerodynamic entrainment of respirable particulates was determined by using data from wind tunnel tests for uranium dioxide power (Mishima and Schwendiman, 1973a). This release mechanism will occur only to the extent that the shipping container is pressurized by the release of gases from the waste containers. The analysis conservatively assumed that maximum pressurization of the container cavity will always occur for every shipment. Based upon the nature of potential container damage previously described, and the void volume space within the container cavity, a depressurization duration of approximately 30 minutes at an average velocity of about 2.5 mph was calculated. For these conditions, the average entrainment value given by Mishima and Schwendiman (1973a) for four surfaces (asphalt, sand, vegetation, and stainless steel) was conservatively assigned.

The algorithm used to calculate the release fraction of respirable radioactive material from impact stresses is summarized in Table D.3.17. Values for specific algorithm parameters are presented in Table D.3.18.

TABLE D.3.17 Estimate of potential accident release fractions for CH and RH TRU waste shipments due to impact events

Impact release fraction (IRF) = (FFC x FMRC) (FMAI + FMEI) (FMRPI)

Where:

FFC = Fraction of failed waste containers

FMRC = Fraction of material released from failed containers into package cavity

FMAI = Fraction of material aerosolized from impact

FMEI = Fraction of material entrained to environment during

FMRPI = Fraction of material released from package cavity during impact event

					TRUPACT-II[a]		RH Cask [a,b]	
Severity category	FMRC	FMAI	FMEI	FMRPI	FFC	IRF	FFC	IRF
1	0×10^{0}	0×10^{0}	0.0×10^{0}	0×10^{0}	0×10^{0}	0×10^{0}	0×10^{0}	0×10^{0}
2	0×10^{0}	0×10^{0}	0.0×10^{0}	0×10^{0}	0×10^{0}	0×10^{0}	0×10^{0}	0×10^{0}
3	1×10^{-1}	8×10^{-5}	0.0×10^{0}	0×10^{0}	3×10^{-1}	0×10^{0}	3×10^{-1}	0×10^{0}
4	3×10^{-1}	8×10^{-5}	0.0×10^{0}	0×10^{0}	5×10^{-1}	0×10^{0}	7×10^{-1}	0×10^{0}
5	5×10^{-1}	8×10^{-5}	1.5×10^{-4}	1×10^{0}	7×10^{-1}	8×10^{-5}	1×10^{0}	1×10^{-4}
6	7×10^{-1}	8×10^{-5}	1.5×10^{-4}	1×10^{0}	1×10^{0}	2×10^{-4}	1×10^{0}	1×10^{-4}
7	1×10^{0}	8×10^{-5}	1.5×10^{-4}	1×10^{0}	1×10^{0}	2×10^{-4}	1×10^{0}	2×10^{-4}
8	1×10^{0}	8×10^{-5}	1.5×10^{-4}	1×10^{0}	1×10^{0}	2×10^{-4}	1×10^{0}	2×10^{-4}

[a] Respirable release fractions.

[b] Release fractions are the same for truck and rail transportation modes.

TABLE D.3.18 Impact release algorithm parameters for CH and RH TRU waste shipments

Parameters	Value	Basis/reference
FFC	.2728 lnF -2.814	Huerta (1983); Shirley (1983). Where F is NRC (1977) accident severity breach force (Newtons)
FMRC	Table D.3.17	Huerta (1983) and NRC (1977) used as guidance
FMAI	Table D.3.17	NRC (1980) resuspension factor of 2.00×10^{-2} m^{-1} used (mechanical stress of vigorous sweeping)
FMEI	1.50×10^{-4}	Mishima and Schwendiman (1973a) average entrainment value for 4 surfaces used with airflow of 2.5 mph for 30 minutes
FMRPI	Accident severity 1-4: 0.0 Accident severity 5-8: 1.0	Type B package design and NRC (1977) used as guidance

Fischer et al. (1987) estimated that 1.7 percent of truck accidents and 6.8 percent of rail accidents will involve fires. For fire events, the following method was used for each accident severity category:

- Identification of the fraction of radioactive material subject to thermal release mechanisms

- Calculation of the fraction of radioactive material released by combustion in a respirable form

- Calculation of the fraction of radioactive material released in a respirable form by the release of gases and the heating of contaminated surfaces

- Determination of the fraction of radioactive material released in a respirable form from any volatilization of radionuclides.

In the absence of detailed knowledge about the responses of shipping containers and waste containers to fires more severe than those specified in regulatory test requirements for Type B packagings, it was conservatively assumed that all radioactive material was available for release for all accidents exceeding severity category two, as limited by the specific release mechanisms.

For combustion related releases, it was assumed that combustible materials could be ignited in all accident severity categories exceeding category two. To maximize the amount of combustible waste burned for a given amount of oxygen, incomplete combustion, producing carbon monoxide (CO), was assumed. The amount of oxygen present to support combustion was calculated by assuming an 85 percent void volume for a loaded shipping container and observing that there would be no external sources of air or oxygen (no major breach of container). From a review of the inorganic compound tables in the Handbook of Chemistry and Physics, it was concluded that any decomposition of metal hydroxides (e.g., $Ca(OH)_2$, $Al(OH)_3$) present in cemented sludges would not act as an internal source of additional oxygen. Finally, the results of experiments conducted by Mishima and Schwendiman (1973b) were used to assess the fraction of radioactive material released in a respirable form from the burning of combustible material.

For accident severity categories four through eight, the fire event may last longer than 1.5 hours. For these more severe conditions, it was assumed that more radioactive material could be converted to an aerosol form because of the release of gases from the waste at elevated temperatures. Potential gas generation was assumed to be comparable for all five accident severity categories and was calculated by assuming a graphite/steam reaction as the off-gassing source. For an upper bound gas generation estimate, it was further assumed that all waste containers within the shipping container were loaded with solidified process waste (water/steam source) and that there was adequate graphite (e.g., molds) present to react with all of the steam.

With these assumptions, gas generation was calculated to be in excess of 600 TRUPACT-II void volumes and 700 RH cask void volumes, at atmospheric pressure. The fraction of respirable radioactive material present in the gases released from the waste containers and subsequently to the environment was calculated by using a resuspension factor approach. A resuspension factor value corresponding to a vigorous and continued surface stress of people walking on a surface contaminated with plutonium dioxide (at a rate of 36 steps per minute) was used in the analysis.

Vaporization was reviewed as another thermal release mechanism. As previously noted, TRU radionuclides are generally present in an oxide form. They are highly stable at elevated temperatures. Alexander et al. (1986) report that volatile releases of transuranic radionuclides are not of any significance until temperatures of 3140°F are reached. The volitization of uranium oxide (e.g., UO_2) becomes measurable at approximately 2960°F. Flame temperatures for the open burning of hydrocarbon fuels (e.g., JP-4, gasoline, diesel) range from 1400°F to 2400°F, with a median temperature of approximately 1800°F. Consequently, a volatile release of TRU or uranium oxide material is not credible for a transportation accident. This is consistent with the release analysis presented by Fischer et al. (1987), in which the releases of TRU material are quantified in terms of particulates only. In conjunction with waste characterization data, it can be concluded that potential accidents involving CH TRU waste shipments cannot result in radioactive material releases in a vapor form. However, RH TRU waste contains activation/fission products that may volatilize at elevated temperatures. These radionuclides are identified as being present in RH TRU waste. Testing conducted by Lorenz (1980) indicates that cesium, antimony, and ruthenium may volatilize at elevated temperatures. Assuming that volatilization mechanisms for RH TRU waste would be similar to the referenced test conditions at 1290°F, it was concluded that the releases of cesium, antimony, and ruthenium vapors would be comparable to the values estimated for respirable particulate releases.

The algorithm for estimating the respirable release fraction of radioactive material from thermal accident events is illustrated in Table D.3.19. Values for specific algorithm parameters are summarized in Table D.3.20.

Total Respirable Release Fractions. The calculated impact release fractions (Table D.3.17) and thermal release fractions (Table D.3.19) were added to determine the total respirable release fractions due to very severe transportation accidents and are summarized in Table D.3.21 and D.3.22. A maximum release fraction of 0.0002 was estimated for accidents involving both CH and RH TRU waste shipments. This is consistent with or bounding of previous transportation risk studies such as the NRC modal study (Fischer et al., 1987), which estimated particulate releases of 0.000002 and vapor (C_S)releases of 0.0002 due to spent fuel shipments, and the WIPP FEIS (DOE, 1980), which incorporated a release fraction of 0.00018 for CH TRU waste shipments.

D.3.3.1.3 Dispersal Conditions. The dispersion of airborne radioactive material during an accident is controlled by meteorological conditions at the time of the accident. The airborne radioactive material moves downwind from the scene of the accident and its dispersal and transport are affected by the degree of atmospheric turbulence. For this analysis, the materials were assumed to move downwind and disperse. As the radioactive cloud disperses, the people in its path will be exposed to external radiation, internal radiation from inhalation, or internal radiation from ingestion. For inhalation and TABLE D.3.19 Estimate of potential accident release fractions for CH and RH TRU waste shipments due to thermal events

Thermal release fraction (TRF) = FAT [(FMC x FMAC) + FMAT] FMRPT

Where: FAT = Fraction of accidents involving a thermal event

FMC = Fraction of material consumed by combustion

FMAC = Fraction of material aerosolized by combustion

FMAT = Fraction of material aerosolized by thermal event

FMRPT= Fraction of material released from package cavity during thermal event

					Truck[a]		Rail[a]	
Severity Category	FMAC	FMAC	FMAT	FMRPT	FAT	TRF	FAT	TRF
TRUPACT-II								
1	0×10^{0}	0×10^{0}	0×10^{0}	0×10^{0}	1.7×10^{-2}	0×10^{0}	6.8×10^{-2}	0×10^{0}
2	0×10^{0}	0×10^{0}	0×10^{0}	0×10^{0}	1.7×10^{-2}	0×10^{0}	6.8×10^{-2}	0×10^{0}
3	9×10^{-4}	5×10^{-4}	2×10^{-8}	1×10^{0}	1.7×10^{-2}	8×10^{-9}	6.8×10^{-2}	2×10^{-8}
4	9×10^{-4}	5×10^{-4}	1×10^{-5}	1×10^{0}	1.7×10^{-2}	2×10^{-7}	6.8×10^{-2}	7×10^{-7}
5	9×10^{-4}	5×10^{-4}	1×10^{-5}	1×10^{0}	1.7×10^{-2}	2×10^{-7}	6.8×10^{-2}	7×10^{-7}
6	9×10^{-4}	5×10^{-4}	1×10^{-5}	1×10^{0}	1.7×10^{-2}	2×10^{-7}	6.8×10^{-2}	7×10^{-7}
7	9×10^{-4}	5×10^{-4}	1×10^{-5}	1×10^{0}	1.7×10^{-2}	2×10^{-7}	6.8×10^{-2}	7×10^{-7}
8	9×10^{-4}	5×10^{-4}	1×10^{-5}	1×10^{0}	1.7×10^{-2}	2×10^{-7}	6.8×10^{-2}	7×10^{-7}
RH Cask								
1	0×10^{0}	0×10^{0}	0×10^{0}	0×10^{0}	1.7×10^{-2}	0×10^{0}	6.8×10^{-2}	0×10^{0}
2	0×10^{0}	0×10^{0}	0×10^{0}	0×10^{0}	1.7×10^{-2}	0×10^{0}	6.8×10^{-2}	0×10^{0}
3	7×10^{-4}	5×10^{-4}	2×10^{-8}	1×10^{0}	1.7×10^{-2}	6×10^{-9}	6.8×10^{-2}	2×10^{-8}
4	7×10^{-4}	5×10^{-4}	9×10^{-5}	1×10^{0}	1.7×10^{-2}	2×10^{-7}	6.8×10^{-2}	7×10^{-7}
5	7×10^{-4}	5×10^{-4}	9×10^{-5}	1×10^{0}	1.7×10^{-2}	2×10^{-7}	6.8×10^{-2}	7×10^{-7}
6	7×10^{-4}	5×10^{-4}	9×10^{-5}	1×10^{0}	1.7×10^{-2}	2×10^{-7}	6.8×10^{-2}	7×10^{-7}
7	7×10^{-4}	5×10^{-4}	9×10^{-5}	1×10^{0}	1.7×10^{-2}	2×10^{-7}	6.8×10^{-2}	7×10^{-7}
8	7×10^{-4}	5×10^{-4}	9×10^{-5}	1×10^{0}	1.7×10^{-2}	2×10^{-7}	6.8×10^{-2}	7×10^{-7}

[a] Respirable release fractions.

TABLE D.3.20 Thermal release algorithm parameters for CH and RH TRU waste shipments

Parameter	Value	Basis/reference
FAT	1.7×10^{-2} (Truck) 6.8×10^{-2} (Rail)	Fischer et al. (1987)
FMC	Accident severity 1-2: 0×10^{0}	
	Accident severity 3-4: 9×10^{-4} (TRUPACT-II) 7×10^{-4} (RH Cask)	Type B package design Limited internal oxygen source: 3.95 lb O_2 (TRUPACT-II) 0.73 lb O_2 (RH Cask)
FMAC	Accident severity 1-2: 0×10^{0}	Type B package design Mishima and Schwendiman (1973b)
	Accident severity 3-8: 5×10^{-4}	
FMAT	Accident severity 1-2: 0×10^{0}	Type B package design
	Accident severity 3: 2×10^{-8}	Only combustion assumed to occur, with attendant off-gas (combustion) products
	Accident severity 4-8: 1×10^{-5} (TRUPACT-II) 9×10^{-6} (RH Cask)	Off-gasing assuming steam/graphite reaction and resuspension factor of 5.00×10^{-6} m^{-1} corresponding to a surface stress from walking (NRC, 1980)
FMRPT	Accident severity 1-2: 0×10^{0}	Type B package design NRC (1977) used as guidance
	Accident severity 3-8: 1×10^{0}	

TABLE D.3.21 CH TRU waste transportation release fractions

Total respirable release fraction (TRRF)	=	Impact release fraction (IRF) + Thermal release fraction (TRF)

Accident severity category	Impact release fraction[a]	Thermal release fraction[b]	Total respirable release fraction
Truck			
1	0×10^{0}	0×10^{0}	0×10^{0}
2	0×10^{0}	0×10^{0}	0×10^{0}
3	0×10^{0}	8×10^{-9}	8×10^{-9}
4	0×10^{0}	2×10^{-7}	2×10^{-7}
5	8×10^{-5}	2×10^{-7}	8×10^{-5}
6	2×10^{-4}	2×10^{-7}	2×10^{-4}
7	2×10^{-4}	2×10^{-7}	2×10^{-4}
8	2×10^{-4}	2×10^{-7}	2×10^{-4}
Rail			
1	0×10^{0}	0×10^{0}	0×10^{0}
2	0×10^{0}	0×10^{0}	0×10^{0}
3	0×10^{0}	2×10^{-8}	2×10^{-8}
4	0×10^{0}	7×10^{-7}	7×10^{-7}
5	8×10^{-5}	7×10^{-7}	8×10^{-5}
6	2×10^{-4}	7×10^{-7}	2×10^{-4}
7	2×10^{-4}	7×10^{-7}	2×10^{-4}
8	2×10^{-4}	7×10^{-7}	2×10^{-4}

[a] From Table D.3.17.

[b] From Table D.3.19.

TABLE D.3.22 RH TRU waste transportation release fractions

Total respirable release fraction (TRRF) = Impact release fraction (IRF) + Thermal release fraction (TRF)

Accident severity category	Impact release fraction[a]	Thermal release fraction[b]	Total respirable release fraction
Truck			
1	0×10^{0}	0×10^{0}	0×10^{0}
2	0×10^{0}	0×10^{0}	0×10^{0}
3	0×10^{0}	6×10^{-9}	6×10^{-9}
4	0×10^{0}	2×10^{-7}	2×10^{-7}
5	1×10^{-4}	2×10^{-7}	1×10^{-4}
6	1×10^{-4}	2×10^{-7}	1×10^{-4}
7	2×10^{-4}	2×10^{-7}	2×10^{-4}
8	2×10^{-4}	2×10^{-7}	2×10^{-4}
Rail			
1	0×10^{0}	0×10^{0}	0×10^{0}
2	0×10^{0}	0×10^{0}	0×10^{0}
3	0×10^{0}	2×10^{-8}	2×10^{-8}
4	0×10^{0}	7×10^{-7}	7×10^{-7}
5	1×10^{-4}	7×10^{-7}	1×10^{-4}
6	1×10^{-4}	7×10^{-7}	1×10^{-4}
7	2×10^{-4}	7×10^{-7}	2×10^{-4}
8	2×10^{-4}	7×10^{-7}	2×10^{-4}

[a] From Table D.3.17.

[b] From Table D.3.19.

ingestion, the degree of exposure depends on the amount of material retained in the lungs or other organs of the exposed persons.

Airborne transport and diffusion can disperse radioactive materials over large areas. The degree of dispersion is influenced by many factors, such as season (which influences atmospheric turbulence), time of day, degree of cloud cover, land surface features and characteristics, and other meteorological parameters. Dispersed material can expose people in many ways, as shown in Figure D.3.3. The principal effect of gamma-emitting materials is a direct external or internal dose. Material that emits alpha or beta radiation if it is converted to an aerosol and inhaled by people produces the largest consequence. Figure D.3.3 illustrates that radioactive materials can also be incorporated in the food chain. Radiation doses received by the population through the food chain pathway are usually more significant if a continuous release exists.

One of the pathways of note is resuspension. This occurs when deposited particulate material becomes airborne through the action of pedestrians, vehicles, plowing, the wind, etc. The resuspended material then becomes available for inhalation and can deliver an additional dose that accumulates with time.

D.3.3.1.4 Pathways and Exposed Populations. RADTRAN or similar analytical tools can be used to evaluate the radiological impacts of transporting radioactive materials under accident conditions. As input to RADTRAN, the exposure pathways must be identified and the size of exposed populations must be estimated. Transportation accidents may be divided into those accidents in which the shipping containers maintain their integrity and there is no release of radioactive materials, and those accidents in which the integrity of the shipping containers is compromised. The exposure pathways and the exposed population subgroups are discussed below.

In an accident that does not compromise the containment of the shipping containers, the exposure pathway is limited to direct exposure by penetrating radiation from the intact package. The dose delivered to any member of an exposed population is evaluated in the same manner as the exposure from normal (incident-free) transportation, with adjustments made for the duration of exposure and the distance between the shipment and the exposed individuals. The exposed populations include the truck or rail crew, the occupants of the other vehicle(s) involved in the accident, bystanders and pedestrians, the occupants of nearby buildings, and the members of emergency response crews.

In an accident that results in a failure of the shipping containers and possible release of radioactive material, exposures may result from both nondispersible and dispersible materials.

The exposure pathway from accidents involving shipping containers with nondispersible materials is direct exposure resulting from the loss of shielding of the contents of the containers. Certain radioactive materials are not dispersible because of their chemical or physical form, such as irradiated steel hardware; these materials may nevertheless result in exposure by penetrating radiation. The doses received by exposed individuals are evaluated in the same manner as other direct exposures, with adjustments made

Figure D.3.3
Possible pathways to man from radionuclide release

for increased dose rates resulting from shielding loss as well as exposure time and distance adjustments. The exposed populations are the same as identified above.

Four exposure pathways may result from accidents that cause a release of dispersible radioactive materials:

- Cloudshine: The exposure from cloudshine is the direct *external* dose from the passing cloud of dispersed material. Dispersion depends on the meteorological conditions at the accident scene, as well as the fraction of failed shipping containers and the fraction of released material that becomes airborne.

- Groundshine: The exposure from groundshine is the direct *external* dose from material that has deposited on the ground after being dispersed from the accident site. The degree of deposition depends on the material being deposited (i.e., the rate at which the dispersed material settles out) and the amount of dispersed material available to settle out (i.e., how much material from the original release has dispersed far enough to deposit on the area of interest).

- Inhalation: The exposure from inhalation is the *internal* exposure that results from breathing aerosolized material. Exposure from inhalation depends on the fraction of failed shipping containers, the fraction of material that becomes airborne, the aerosol fraction of respirable size, the radiation dose delivered per curie of radioactivity inhaled, the dilution factor for radioactive material in the surrounding air, and the breathing rate of the exposed individual.

- Resuspension: The exposure from resuspension is the *internal* exposure that results from the inhalation of material that was dispersed, deposited at a distance from the accident scene and then resuspended as an aerosol and inhaled. Exposure from resuspension requires combining the mechanisms of dispersion, deposition and inhalation described above, as well as estimating the fraction of deposited material that is resuspended. (Resuspension may result from changing weather conditions, such as changes in wind speed or direction, or from disturbing deposited material by other means, such as traffic through a deposition area.) Note that exposure by ingestion is not included in evaluating the radiological impacts of accidents because it is assumed that emergency response and governmental authorities would intervene to impound foodstuffs, provide an alternative water supply, and clean up contaminated land.

The population subgroups that are exposed by an accident that results in dispersion of radioactive material include the individuals who are directly exposed at the scene of the accident and the individuals who are present in the areas over which dispersion occurs.

D.3.3.2 **Results of the Accident Analysis**

The radiological exposures associated with truck or rail accidents involving CH TRU waste are expressed as the exposure per shipment and as a cumulative exposure over the shipping campaign for the alternative being considered. The exposure is the sum of the products of the probability of a given severity accident times the consequences of such an accident for each of the severity categories. The radiological exposures from an accident involving CH TRU waste are expressed in equivalent whole body dose and are tabulated in units of person-rem, and assume three TRUPACT-II containers per truck shipment and six TRUPACT-II containers per rail shipment. Table D.3.23 presents the exposure per shipment for each facility that ships CH TRU waste and the total per shipment exposure for all facilities for truck and rail modes. Table D.3.24 presents the cumulative exposure for all facilities that ship CH TRU waste to the WIPP. This table shows the estimated radiological exposures for transportation accidents in the Proposed Action, which consists of the Test Phase (10 percent of CH TRU waste shipped and all shipments by truck) and the Disposal Phase, in which truck or rail could be used.

No radiological exposures from transportation accidents were calculated for the No Action Alternative because no shipments to the WIPP would be made.

For the Alternative Action, the radiological exposures from truck accidents are the sum of the exposures from the Test Phase and Disposal Phase (Table D.3.24). These exposures would be incurred in a continuous 20-year period after an approximate 5-year Test Phase during which no waste would be shipped to the WIPP but during which approximately seven truck shipments of CH TRU waste would be made from the Rocky Flats Plant to the Idaho National Engineering Laboratory to support bin tests. The accident contribution for these shipments was calculated by subtracting the per-shipment radiological exposure from accidents (Table D.3.23) for a shipment from the Idaho National Engineering Laboratory to the WIPP from that for a shipment from the Rocky Flats Plant to the WIPP. This difference, which represents the Idaho-to-Rocky Flats transportation segment, was multiplied by the number of shipments to arrive at the transportation exposures from the bin tests. Thus, an accident contribution of approximately 5.90×10^{-4} person-rem is expected from the bin test shipments. The radiological exposures from rail accidents for the Proposed Action and the Alternative Action are shown in Table D.3.25.

The radiological exposures from an accident involving a truck or a railcar carrying RH TRU waste are expressed in equivalent whole body dose and are tabulated in units of person-rem, assuming one RH TRU cask per truck shipment and two RH casks per rail shipment. Table D.3.26 presents the per shipment exposure for each facility that ships RH TRU waste by truck or rail and the total exposures for all facilities. Table D.3.27 presents the cumulative exposure for all facilities that ship RH TRU waste to the WIPP. These lifetime radiological exposures from transportation accidents involving RH TRU waste are shown in Table D.3.27 for a 20-year shipping period. No RH TRU waste shipments would occur during the Test Phase of the Proposed Action or the Alternative Action, and therefore no accident exposures result. The radiological exposures of RH TRU shipments are identical for the Proposed Action and the Alternative Action.

TABLE D.3.23 Per shipment accident radiological exposures of CH TRU waste shipments (person-rem)[a,b,c]

Facility	Nonoccupational accident contribution Truck	Rail
Idaho National Engineering Laboratory	7.9×10^{-4}	5.7×10^{-4}
Rocky Flats Plant	2.0×10^{-4}	1.9×10^{-4}
Hanford Reservation	9.9×10^{-4}	8.9×10^{-4}
Savannah River Site	4.2×10^{-2}	4.0×10^{-2}
Los Alamos National Laboratory	1.3×10^{-3}	[d]
Oak Ridge National Laboratory	4.4×10^{-3}	4.22×10^{-3}
Nevada Test Site	8.9×10^{-6}	[d]
Argonne National Laboratory-East	4.9×10^{-4}	3.5×10^{-4}
Lawrence Livermore National Laboratory	1.9×10^{-4}	2.94×10^{-4}
Mound Laboratory	2.8×10^{-5}	5.4×10^{-7}

[a] Population group exposures per waste shipment are expressed in equivalent whole body dose and are tabulated in units of person-rem.

[b] Values for rail are expressed per railcar shipment.

[c] Population group exposures per waste shipment are presented as a function of the Transport Index (TI), which is defined as the dose rate in mrem/hr at 1 m from the waste package.

[d] No railheads present.

TABLE D.3.24 Lifetime radiological exposures for accidents during transportation of CH TRU waste (person-rem): Proposed Action and Alternative Action[a,c]

		Proposed Action		Alternative Action	
		Disposal Phase (20-yr)		Disposal Phase (20-yr)	
Facility	Test Phase[b]	Truck	Max. rail	Truck	Max. rail
Idaho National Engineering Laboratory	3.2×10^{-1}	2.9×10^{0}	1.0×10^{0}	3.2×10^{0}	1.2×10^{0}
Rocky Flats Plant	1.5×10^{-1}	1.4×10^{0}	6.5×10^{-1}	1.5×10^{0}	7.2×10^{-1}
Hanford Reservation	3.1×10^{-1}	2.8×10^{0}	1.2×10^{0}	3.1×10^{0}	1.4×10^{0}
Savannah River Site	1.1×10^{1}	1.0×10^{2}	4.8×10^{1}	1.1×10^{2}	5.3×10^{1}
Los Alamos National Laboratory[d]	2.7×10^{-1}	2.4×10^{0}	2.4×10^{0}	2.7×10^{0}	2.7×10^{0}
Oak Ridge National Laboratory	1.0×10^{-1}	9.0×10^{-1}	4.3×10^{-1}	1.0×10^{0}	4.8×10^{-1}
Nevada Test Site[d]	7.1×10^{-5}	6.4×10^{-4}	6.4×10^{-4}	7.1×10^{-4}	7.1×10^{-4}
Argonne National Laboratory-East	6.9×10^{-4}	6.2×10^{-3}	2.2×10^{-3}	6.9×10^{-3}	2.4×10^{-3}
Lawrence Livermore National Laboratory	1.8×10^{-2}	1.6×10^{-1}	1.3×10^{-1}	1.8×10^{-1}	1.4×10^{-1}
Mound Laboratory	4.2×10^{-4}	3.8×10^{-3}	3.6×10^{-5}	4.2×10^{-3}	4.0×10^{-5}
Total	1.2×10^{1}	1.1×10^{2}	5.4×10^{1}	1.2×10^{2}	6.0×10^{1}

[a] Population group exposures are calculated by multiplying the exposure/shipment identified in Table D.3.23 by the total number of shipments to the WIPP by truck or rail, as determined from the projection in Table D.3.2.

[b] Test Phase assumes 10% of shipment completed by truck.

[c] Nonoccupational population.

[d] Waste shipments from this facility are limited to truck mode, thus rail exposures are the same as truck exposures.

TABLE D.3.25 Summary of lifetime radiological exposure changes between Proposed Action and Alternative Action: CH TRU accident nonoccupational risk (person-rem)

Facility	Proposed Action Truck	Proposed Action Rail	Alternative Action Truck	Alternative Action Rail
Idaho National Engineering Laboratory	3.2×10^{0}	1.3×10^{0}	3.2×10^{0}	1.2×10^{0}
Rocky Flats Plant	1.5×10^{0}	8.0×10^{-1}	1.5×10^{0}	7.2×10^{-1}
Hanford Reservation	3.1×10^{0}	1.5×10^{0}	3.1×10^{0}	1.4×10^{0}
Savannah River Site	1.1×10^{2}	5.9×10^{1}	1.1×10^{2}	5.3×10^{1}
Los Alamos National Laboratory	2.7×10^{0}	2.7×10^{0}	2.7×10^{0}	2.7×10^{0}
Oak Ridge National Laboratory	1.0×10^{0}	5.3×10^{-1}	1.0×10^{0}	4.8×10^{-1}
Nevada Test Site	7.1×10^{-4}	7.1×10^{-4}	7.1×10^{-4}	7.1×10^{-4}
Argonne National Laboratory-East	6.9×10^{-3}	2.9×10^{-3}	6.9×10^{-3}	2.4×10^{-3}
Lawrence Livermore National Laboratory	1.8×10^{-1}	1.5×10^{-1}	1.8×10^{-1}	1.4×10^{-1}
Mound Laboratory	4.2×10^{-3}	4.6×10^{-4}	4.2×10^{-3}	4.0×10^{-5}
Total	1.2×10^{2}	6.6×10^{1}	1.2×10^{2}	6.0×10^{1}

TABLE D.3.26 Per shipment accident radiological exposures of RH TRU shipments (person-rem)[a,b,c]

	Nonoccupational accident contribution	
Facility	Truck	Rail
Idaho National Engineering Laboratory	1.6×10^{-3}	1.3×10^{-3}
Hanford Reservation	4.34×10^{-5}	4.44×10^{-5}
Los Alamos National Laboratory	3.09×10^{-6}	[d]
Oak Ridge National Laboratory	4.84×10^{-6}	5.21×10^{-6}
Argonne National Laboratory-East	6.4×10^{-6}	5.2×10^{-6}

[a] Exposures to the population per waste shipment are expressed in equivalent whole body dose and are tabulated in units of person-rem.

[b] Values for rail are expressed per railcar shipment.

[c] Exposures to the population per waste shipment are presented as a function of the Transport Index (TI) which is defined as the dose rate in mrem/hr at 1 meter from the waste package. Calculations are based on three TRUPACT-II waste packages per truck and six per railcar shipment.

[d] No railheads present.

TABLE D.3.27 Lifetime radiological exposures for accidents during transportation of RH TRU waste (person-rem): Proposed Action and Alternative Action[a,b]

Facility	100% Truck	Maximum rail
Idaho National Engineering Laboratory	7.8×10^{-1}	3.2×10^{-1}
Hanford Reservation	1.1×10^{-1}	5.4×10^{-2}
Los Alamos National Laboratory[c]	3.1×10^{-4}	3.1×10^{-4}
Oak Ridge National Laboratory	2.2×10^{-2}	1.2×10^{-2}
Argonne National Laboratory-East	1.9×10^{-3}	7.8×10^{-4}
Total	9.1×10^{-1}	3.9×10^{-1}

[a] Population group exposures are calculated by multiplying the exposure/shipment identified in Table D.3.26 by the total number of shipments to WIPP by truck or rail, as determined from the projection in Table D.3.22. Rail occupational exposures resulting from normal transportation include the impact of inspection activities.

[b] Nonoccupational populations.

[c] Waste shipments from the facility are limited to truck mode. Rail exposures are thus the same as the truck exposures.

D.3.4 **RADIOLOGICAL CONSEQUENCES OF BOUNDING CASE TRANSPORTATION ACCIDENT**

D.3.4.1 **Assumptions: Bounding Case Accident**

As discussed in Section 5.0, "bounding case" transportation accident scenarios were developed for this SEIS. These scenarios were used to calculate the impact of very severe accidents in higher population areas along the WIPP-preferred transportation routes. Postulated accidents involved both CH and RH truck and rail shipments using TRUPACT-II containers or RH casks. Based on comments received on the draft SEIS, a revised bounding case accident was calculated based on higher curie content CH waste primarily from Los Alamos National Laboratory, the Savannah River Site, and the Idaho National Engineering Laboratory. In the draft SEIS, calculations assuming average CH waste from the Rocky Flats Plant waste were used because these shipments comprise the majority of the total CH waste shipments. Less likelihood of the current bounding case accidents is expected because the number of shipments of maximally loaded containers (WAC or TRUPACT Payload Compliance Plan limits) are smaller than the number of shipments with average waste loadings. Waste compositions from Los Alamos National Laboratory, Savannah River Site, and the Idaho National Engineering Laboratory were analyzed for CH TRU shipments, and from Hanford and the Idaho National Engineering Laboratory for RH TRU shipments. These waste compositions were scaled up to the maximum total curie content of radionuclides allowed by either the WIPP Waste Acceptance Criteria or the TRUPACT Payload Compliance Plan.

During each accident, all TRUPACT-II containers or RH casks were assumed to be equally breached and subsequently engulfed in fire for two hours (it is estimated that at least 17,000 gallons of fuel would be required to provide sufficient fuel to sustain a two-hour fire). External air/oxygen sources were assumed to be limited (internal combustion is limited) because a major breach of the Type B TRUPACT-II containers or RH casks is not credible. Radioactive contamination and hazardous chemicals were assumed to be evenly distributed throughout the waste volume and 0.02 percent of the hazardous and radioactive particulate materials were postulated to be released in a respirable form (less than 10 micron particle size). Each accident was assumed to occur during a period having very stable atmospheric meteorological conditions, so as to limit dispersion or breakup of the plume and maximize radiation doses and hazardous chemical concentrations.

The accident risk analysis method discussed in Subsection D.3.3 relies on the probabilistic approach in RADTRAN to determine cumulative risks of a series of increasingly less probable but more severe accident scenarios. To determine the accident consequences of the "bounding case" accident scenarios, a probability of 100 percent was specified. The specific conditions assumed for these bounding case accidents are summarized in Table D.3.28.

The probability of breaching all Type B containers or casks during truck or rail accidents and engulfing them in a two-hour fire (requiring the fuel equivalent of two fully loaded fuel transports) in an urban area during adverse meteorological conditions

TABLE D.3.28 Bounding case accident scenario assumptions

The waste shipment is assumed to be three fully-loaded TRUPACT-IIs or 1 RH cask on a combination tractor-trailer truck or six fully-loaded TRUPACT-IIs or two RH casks on a railcar. The origin facilities of the waste shipments are those with the greatest likelihood of having a trailer load of waste with a curie content set at the maximum thermal or fissile gram limits specified by the WIPP Waste Acceptance Criteria or WIPP Payload Compliance Plan.

All waste is packaged in Type A drums.

A major breach of any of the Type B TRUPACT-II containers or RH casks that compose a TRU shipment is not credible, limiting external air/oxygen sources.

Loss of packaging containment will result in .0002 fraction of the radioactive waste material in the TRUPACT-II containers or RH casks being released to the environment in a respirable form. These respirable materials are airborne particulates and aerosols, which are all less than 10 microns aerodynamic diameter in size.

Radioactive contamination is evenly distributed throughout the waste volume.

The highest accident severity category, category eight, is assumed, with a fire duration of two hours.

All TRUPACT-II containers or RH casks on the trailer or railcar are equally breached.

The accident occurs in the urban or suburban portion of a nonspecific large (greater than one million population) metropolitan area with a mean population density of 3,861 persons (urban) or 719 persons (suburban) per square kilometer in the subarea immediately surrounding the accident site.

An aerosol cloud of respirable radionuclides is dispersed downwind.

is very small. The probability would be a small fraction of the fraction, $0.05 \times 1.5 \times 10^{-5}$ for a truck shipment or a small fraction of $0.05 \times 1.0 \times 10^{-5}$ for a rail shipment (Tables D.3.15 and D.3.16). Additional conservatism in the analysis included the use of a range of population densities higher than currently exist along most WIPP transportation corridors, including Atlanta, Georgia; Denver, Colorado; and Albuquerque, New Mexico.

These conditions were input to the RADTRAN computer code to determine radiological consequences of these bounding cases. These radiological consequences measure the potential to cause immediate and delayed health effects in the affected population, including early fatalities, early morbidities, latent cancer fatalities, and genetic effects from the inhalation, resuspension, groundshine, and cloudshine of the aerosol cloud of the released radionuclides. As a check on estimated consequences, each bounding case scenario was also analyzed with the AIRDOS model. A comparison or RADTRAN and AIRDOS parameters for CH and RH bounding cases is shown in Tables D.3.29 and D.3.30.

D.3.4.2 **Results: Bounding Case Accident**

The RADTRAN and AIRDOS codes were used to predict the consequences of the bounding case accident scenarios. As previously discussed, health impacts may result from external exposure (e.g., cloudshine, groundshine) and internal exposure (e.g., inhalation, resuspension, and ingestion) to the dispersed radioactive material. Since it was assumed that the accidents occurred in an urban or suburban area, ingestion impacts associated with contamination of agricultural products were not applicable.

The analysis assumed that stable to extremely stable atmospheric conditions predominated. This assumption conservatively predicted high airborne radioactive contaminant concentrations and limited the dispersion of the contaminants to outlying areas. In an urban area, surface irregularities and thermal anomalies will tend to preclude the probability of a prevailing stable atmospheric condition.

The revised results of the bounding case accident analyses are presented in Tables D.3.31 through D.3.34 for CH and RH truck and rail scenarios. Contributions to the total committed effective dose equivalent (CEDE) for the exposed population from various pathways (initial inhalation, inhalation from resuspension processes, groundshine, cloudshine) are shown as calculated by both RADTRAN and AIRDOS. The dose expected for the maximally exposed individual as directly calculated by AIRDOS is also shown for each scenario. Population doses were converted to estimates of health effects (latent cancer fatalities) using a conversion factor of 1 person-rem = 2.8×10^{-4} LCFs.

For all the scenarios analyzed, neither RADTRAN nor AIRDOS estimated any early fatalities or morbidities. The estimated population doses were dominated by inhalation contributions (initial or from resuspension processes). Two values for the resuspended inhalation dose contribution were calculated using RADTRAN. These values were calculated using resuspension particle half-lives of 365 and 60 days and are designated

TABLE D.3.29 CH bounding case accident inputs

Input factor	RADTRAN III	AIRDOS
Curies per TRUPACT-II	Same for each model	Maximum allowed per thermal or fissile grams limits set by WAC or Payload Compliance Plan: LANL 1080 PE-Ci[a] (7170 total Ci) SRS 1100 PE-Ci (3750 total Ci) INEL 1200 PE-Ci (6540 total Ci)
Release fraction	.0002 released of all Ci as airborne, respirable fraction for both models	
Release height	Ground release	Ground release (3.5 meters)
Weather	Same, Stability Class F for both models	
Wind speed	1 meter per second	2 meters per second
Population density	Same for both models (Urban: 3861 people per square kilometer Suburban: 719 people per square kilometer)	
Directly calculated Pathway doses	Inhalation Resuspension Groundshine Cloudshine Ingestion	Inhalation ----------- Groundshine Cloudshine ----------
Calculation of "Maximum Individual" Directly	No	Yes

[a] PE-Ci is plutonium equivalent curies calculated using weighting factors in Appendix F.

in the tables as Resusp. I and Resusp. II, respectively. The resuspension half-life is the required time for half of the initially deposited material to be removed from the accessible environment (i.e., at this point, half of the initially deposited material is still available for resuspension). Because inhalation of resuspended particles is a major contributor to the estimated population dose, variation of the resuspension half-life can significantly affect the total calculated dose as shown in the tables. A resuspension half-life of 365 days is extremely conservative given washing (rain) and weathering (wind) processes which would serve to remove contaminants from the accessible environment. The assumed population density also affects the total calculated dose and estimated health effects as shown by comparing results of Los Alamos National Laboratory bounding case accidents occurring in either urban or suburban population zones (Table D.3.31).

For CH truck shipments, depending on shipment origin facility and using a resuspension half-life of 365 days, the total population doses as calculated by RADTRAN and AIRDOS ranged from 6,550 person-rem (1.8 LCFs) to 180,000 person-rem (50 LCFs). Using a 60-day resuspension half-life, the population doses ranged from 6,550 person-rem (1.8 LCFs) to 55,800 person-rem (15.6 LCFs). The estimated maximum individual doses ranged from 160 mrem to 180 mrem depending on shipment origin site.

Results for CH rail shipments were twice those calculated for truck shipments for those facilities with rail access (Savannah River Site and the Idaho National Engineering Laboratory) because a rail shipment involves twice the number of TRUPACT-II containers as a truck shipment.

For RH truck shipments, depending on shipment origin facility and assuming a resuspension half-life of 365 days, the total population doses as calculated by RADTRAN or AIRDOS ranged from 899 person-rem (.25 LCFs) to 40,100 person-rem (11.2 LCFs). For a 60-day resuspension half-life, population doses ranged from 899 person-rem (.25 LCFs) to 12,400 person-rem (3.5 LCFs). The estimated maximum individual doses ranged from 4 mrem to 40 mrem depending on shipment origin facility.

As for CH shipments, results for RH rail shipments were twice those estimated for RH truck shipments because a rail shipment involves two RH casks, whereas a truck shipment involves one RH cask.

TABLE D.3.1 Year 2013 projected retrievably stored and newly generated TRU waste volumes

Facility	1987 IDB 12/31/86 stored (ft^3)	1987 IDB amount generated through 2013 (ft^3)	Total base (ft^3)	Volume scale-up (ft^3)	Total maximum volume case (ft^3)
Contact-Handled					
Idaho National Engineering Laboratory (INEL)	1.07×10^6	9.92×10^3	1.08×10^6	1.16×10^5	1.20×10^6
Rocky Flats Plant (RFP)	0	2.04×10^6	2.04×10^6	2.19×10^5	2.26×10^6
Hanford Reservation (HANF)	2.93×10^5	5.38×10^5	8.31×10^5	8.93×10^4	9.20×10^5
Savannah River Site (SRS)	9.15×10^4	6.16×10^5	7.07×10^5	7.60×10^4	7.83×10^5
Los Alamos National Laboratory (LANL)	2.51×10^5	3.02×10^5	5.53×10^5	5.95×10^4	6.13×10^5
Oak Ridge National Laboratory (ORNL)	1.92×10^4	4.20×10^4	6.12×10^4	6.77×10^3	6.77×10^4
Nevada Test Site (NTS)	2.13×10^4	0	2.13×10^4	2.29×10^3	2.36×10^4
Argonne National Laboratory-East (ANLE)	0	3.80×10^3	3.80×10^3	4.10×10^2	4.22×10^3
Lawrence Livermore National Laboratory (LLNL)	0	2.59×10^5	2.59×10^5	2.79×10^4	2.87×10^5
Mound Laboratory (Mound)	0	4.01×10^4	4.01×10^4	4.31×10^3	4.45×10^4
TOTAL	1.75×10^6	3.85×10^6	5.60×10^6	6.02×10^5	6.20×10^6
Remote-Handled					
Idaho National Engineering Laboratory	9.85×10^0	4.82×10^3	5.80×10^3	9.48×10^3	$1.53. \times 10^4$
Hanford Reservation	8.48×10^2	2.86×10^4	2.94×10^4	4.80×10^4	7.75×10^4
Oak Ridge National Laboratory	4.55×10^4	9.54×10^3	5.50×10^4	8.97×10^4	1.45×10^5
Argonne National Laboratory-East	0	3.50×10^3	3.50×10^3	5.76×10^3	9.29×10^3
Los Alamos National Laboratory	1.02×10^3	1.91×10^2	1.21×10^3	1.97×10^3	3.18×10^3
TOTAL	4.83×10^4	4.67×10^4	9.29×10^4	1.57×10^5	2.50×10^5

TABLE D.3.3 Average radioactivity in a shipment of CH TRU waste [a]

Radionuclide	Waste facility[b] ANLE	HANF	INEL	LANL	LLNL	Mound	NTS	ORNL	RFP	SRS
Thorium-232	0.00×10^{0}	0.00×10^{0}	5.17×10^{-5}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	4.26×10^{-4}	0.00×10^{0}	0.00×10^{0}
Uranium-233	0.00×10^{0}	0.00×10^{0}	1.53×10^{-1}	2.95×10^{-2}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	3.85×10^{1}	0.00×10^{0}	0.00×10^{0}
Uranium-235	0.00×10^{0}	0.00×10^{0}	5.79×10^{-6}	8.37×10^{-5}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	1.15×10^{-3}	0.00×10^{0}	0.00×10^{0}
Uranium-238	0.00×10^{0}	0.00×10^{0}	9.72×10^{-6}	3.61×10^{-4}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	4.59×10^{-3}	0.00×10^{0}	0.00×10^{0}
Neptunium-237	9.65×10^{-4}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	4.09×10^{-3}
Plutonium-238	5.39×10^{0}	3.08×10^{0}	1.08×10^{1}	1.67×10^{2}	3.42×10^{-1}	1.36×10^{0}	3.82×10^{-2}	5.75×10^{1}	5.37×10^{-1}	1.83×10^{3}
Plutonium-239	3.41×10^{0}	3.30×10^{1}	5.89×10^{0}	8.86×10^{1}	8.23×10^{0}	1.18×10^{-2}	6.46×10^{-1}	1.24×10^{2}	1.82×10^{1}	2.20×10^{0}
Plutonium-240	1.56×10^{0}	1.18×10^{1}	1.44×10^{0}	2.04×10^{1}	2.36×10^{0}	3.10×10^{-3}	1.53×10^{-1}	0.00×10^{0}	4.15×10^{0}	8.81×10^{-1}
Plutonium-241	3.10×10^{1}	5.98×10^{2}	4.55×10^{1}	6.88×10^{2}	7.84×10^{1}	1.19×10^{-3}	5.76×10^{0}	0.00×10^{0}	1.29×10^{2}	6.61×10^{1}
Plutonium-242	0.00×10^{0}	2.66×10^{-3}	0.00×10^{0}	4.00×10^{-3}	1.29×10^{-4}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	3.70×10^{-4}	7.19×10^{-4}
Americium-241	1.41×10^{1}	0.00×10^{0}	3.89×10^{1}	2.90×10^{2}	6.81×10^{0}	0.00×10^{0}	0.00×10^{0}	1.04×10^{1}	8.62×10^{-1}	1.81×10^{-1}
Curium-244	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	6.90×10^{1}	0.00×10^{0}	0.00×10^{0}
Californium-252	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	1.10×10^{1}	0.00×10^{0}	0.00×10^{0}
TOTAL	5.55×10^{1}	6.46×10^{2}	1.03×10^{2}	1.25×10^{3}	9.62×10^{1}	1.38×10^{0}	6.59×10^{0}	3.10×10^{2}	1.53×10^{2}	1.89×10^{3}

[a] Radioactivity in curies per shipment for the volumes of waste assumed for the SEIS analyses (ie., volumes scaled up to correspond to the design capacity of the WIPP--see last column, Table B.2.4). The volume per shipment is 8.4 m^3 (three TRUPACT-II containers per shipment, with 2.8 m^3 per TRUPACT-II shipping container).

[b] Key: ANLE, Argonne National Laboratory--East; HANF, Hanford Reservation; INEL, Idaho National Engineering Laboratory; LANL, Los Alamos National Laboratory; LLNL, Lawrence Livermore National Laboratory; Mound, Mound Laboratory; NTS, Nevada Test Site; ORNL, Oak Ridge National Laboratory; RFP, Rocky Flats Plant; SRS, Savannah River Site.

TABLE D.3.10 Lifetime radiological exposures of incident-free transportation of CH TRU waste (person-rem)[d]

	Proposed Action						Alternative Action			
	Test Phase (5-yr)[a]		Disposal Phase (20-yr)				Disposal Phase (20-yr)			
	100% Truck		100% Truck		Max. rail		100% Truck		Max. rail	
Shipment origin site	Occ[b]	Nonocc[c]	Occ	Nonocc	Occ	Nonocc	Occ	Nonocc	Occ	Nonocc
Idaho National Engineering Laboratory	2.0×10^{1}	8.1×10^{0}	1.8×10^{2}	7.3×10^{1}	5.3×10^{-1}	5.5×10^{1}	2.0×10^{2}	8.1×10^{1}	5.9×10^{-1}	6.1×10^{1}
Rocky Flats Plant	3.0×10^{1}	7.6×10^{0}	2.7×10^{2}	6.8×10^{1}	9.0×10^{-1}	6.8×10^{1}	3.0×10^{2}	7.6×10^{1}	1.0×10^{0}	7.6×10^{1}
Hanford Reservation	1.2×10^{1}	7.1×10^{0}	1.1×10^{2}	6.4×10^{1}	3.6×10^{-1}	5.6×10^{1}	1.2×10^{2}	7.1×10^{1}	4.0×10^{-1}	6.2×10^{1}
Savannah River Site	3.7×10^{1}	1.8×10^{1}	3.3×10^{2}	1.7×10^{2}	1.0×10^{0}	1.4×10^{2}	3.7×10^{2}	1.8×10^{2}	1.1×10^{0}	1.6×10^{2}
Los Alamos National Laboratory[e]	5.8×10^{0}	1.6×10^{0}	5.2×10^{1}	1.5×10^{1}	5.2×10^{1}	1.5×10^{1}	5.8×10^{1}	1.6×10^{1}	5.8×10^{1}	1.6×10^{1}
Oak Ridge National Laboratory	3.0×10^{0}	4.6×10^{0}	2.7×10^{1}	4.1×10^{1}	2.2×10^{-1}	2.0×10^{1}	3.0×10^{1}	4.6×10^{1}	2.4×10^{-1}	2.3×10^{1}
Nevada Test Site[e]	4.0×10^{-1}	1.6×10^{-1}	3.6×10^{0}	1.4×10^{0}	3.6×10^{0}	1.4×10^{0}	4.0×10^{0}	1.6×10^{0}	4.0×10^{0}	1.6×10^{0}
Argonne National Laboratory-East	1.8×10^{-1}	2.0×10^{-1}	1.6×10^{0}	1.8×10^{0}	1.1×10^{-2}	1.2×10^{0}	1.8×10^{0}	2.0×10^{0}	1.3×10^{-2}	1.3×10^{0}
Lawrence Livermore National Laboratory	1.6×10^{0}	8.7×10^{-1}	1.5×10^{1}	7.8×10^{0}	5.2×10^{-2}	7.0×10^{0}	1.6×10^{1}	8.7×10^{0}	5.8×10^{-2}	7.8×10^{0}
Mound Laboratory	2.8×10^{-1}	1.4×10^{-1}	2.6×10^{0}	1.2×10^{0}	7.4×10^{-3}	9.0×10^{-1}	2.8×10^{0}	1.4×10^{0}	8.2×10^{-3}	1.0×10^{-0}
Total	1.1×10^{2}	4.8×10^{1}	9.9×10^{2}	4.4×10^{2}	5.9×10^{1}	3.7×10^{2}	1.1×10^{3}	4.8×10^{2}	6.5×10^{1}	4.1×10^{2}

[a] Test Phase assumes 10% of shipments; all by truck.

[b] Occupational population---quantifies doses received by transportation crews.

[c] Nonoccupational population.

[d] Population group exposures are calculated by multiplying the exposure/shipment identified in Table D.3.8 by the total number of shipments to the WIPP by truck or rail, as determined from the projections in Table D.3.2. Rail occupational exposures resulting from normal transportation include the impact of inspection activities.

[e] Waste shipments are limited to truck mode. Rail exposures are thus the same as truck exposures.

TABLE D.3.14 Estimated maximum exposure to individuals within various population categories from incident-free transportation during the Test Phase and Disposal Phase for the Proposed Action and during the Disposal Phase for the Alternative Action (rem)

100 % Truck shipment case

Occupational | Nonoccupational

Crew member[a]

Departure In-transit[b] | State Stops[c] | Total[d] | inspections[e] | inspections[f] | On-link[g] | Off-link[h] | Stops[i]

Contact-Handled													
INEL	3.5×10^{1}		6.8×10^{0}		4.2×10^{1}		2.7×10^{-1}	8.1×10^{-1}	5.0×10^{-4}	1.5×10^{-4}	5.5×10^{-2}		
RFP	5.4×10^{1}		1.0×10^{1}		6.4×10^{1}		7.6×10^{-1}	2.3×10^{0}		7.5×10^{-4}	4.2×10^{-4}	1.5×10^{-1}	
Hanford	2.4×10^{1}		4.8×10^{0}		2.9×10^{1}		1.5×10^{-1}	4.3×10^{-1}	3.5×10^{-4}	8.1×10^{-5}	2.9×10^{-2}		
SS	1.1×10^{2}		1.9×10^{1}		1.3×10^{2}		4.8×10^{-1}	1.4×10^{0}		1.4×10^{-3}	2.6×10^{-4}	9.6×10^{-2}	
LANL	2.9×10^{1}		7.6×10^{0}		3.7×10^{1}		5.7×10^{-1}	1.7×10^{0}		2.1×10^{-3}	3.1×10^{-4}	1.1×10^{-1}	
ORNL	1.5×10^{1}		1.0×10^{1}		2.5×10^{1}		1.7×10^{-1}	5.0×10^{-1}	5.5×10^{-3}	9.1×10^{-5}	3.4×10^{-2}		
NTS	1.8×10^{0}		3.6×10^{-1}	2.2×10^{0}		6.4×10^{-3}	1.9×10^{-2}	6.0×10^{-4}	3.5×10^{-6}	1.3×10^{-3}			
ANLE	9.1×10^{-1}	4.1×10^{-1}	1.3×10^{0}		7.0×10^{-3}	2.1×10^{-2}	3.8×10^{-3}	3.8×10^{-6}	1.4×10^{-3}				
LLNL	8.2×10^{0}		1.7×10^{0}		9.9×10^{0}		2.6×10^{-2}	7.8×10^{-2}	2.0×10^{-4}	1.5×10^{-5}	5.2×10^{-3}		
MOUND	1.4×10^{0}		2.7×10^{-1}	1.7×10^{0}		4.1×10^{-3}	1.2×10^{-2}	2.1×10^{-4}	2.3×10^{-6}	8.3×10^{-4}			
TOTAL (WIPP)	--		--		--		2.4×10^{0}[q]	7.3×10^{0}		--		1.3×10^{-3}	4.8×10^{-1}
Remote-Handled													
INEL	2.4×10^{1}		1.1×10^{1}		3.5×10^{1}		2.0×10^{-1}	4.9×10^{-1}	2.5×10^{-3}	6.3×10^{-5}	4.1×10^{-2}		
Hanford	8.4×10^{1}		9.2×10^{1}		1.8×10^{2}		3.3×10^{0}		7.9×10^{0}		8.0×10^{-3}	1.1×10^{-4}	6.6×10^{-1}
LANL	1.4×10^{0}		8.1×10^{-1}	2.2×10^{0}		7.5×10^{-2}	1.8×10^{-1}	4.5×10^{-3}	2.4×10^{-5}	1.5×10^{-2}			
ORNL	4.5×10^{1}		1.7×10^{1}		6.2×10^{1}		1.2×10^{0}		3.0×10^{0}		1.6×10^{-3}	4.0×10^{-4}	2.5×10^{-1}
ANLE	7.7×10^{0}		2.9×10^{0}		1.1×10^{1}		6.3×10^{-2}	1.5×10^{-1}	1.3×10^{-3}	2.0×10^{-5}	1.3×10^{-2}		
TOTAL (WIPP)	--		--		--		4.8×10^{0}[q]	1.2×10^{1}		--		6.2×10^{-4}	9.8×10^{-1}

TABLE D.3.14 Continued

Maximum rail shipment case

	Occupational						Nonoccupational						
	Crew member[a,j]												
	In-transit[b]	Stops[k]		Yard Total[d]		Departure crew[l]		State inspections[m]		inspections[n]		Off-link[h]	Stops[o]
Contact-Handled													
INEL	2.0×10^{-2}	1.3×10^{0}		1.3×10^{0}		5.4×10^{-2}	2.7×10^{-1}	6.1×10^{-1}	1.5×10^{-4}	2.8×10^{-2}			
RFP	2.9×10^{-2}	1.9×10^{0}		1.9×10^{0}		1.5×10^{-1}	7.6×10^{-1}	1.7×10^{0}		4.2×10^{-4}	7.9×10^{-2}		
Hanford	1.7×10^{-2}	1.1×10^{0}		1.1×10^{0}		2.9×10^{-2}	1.5×10^{-1}	3.2×10^{-1}	7.9×10^{-5}	1.5×10^{-2}			
SRS	3.5×10^{-2}	2.3×10^{0}		2.3×10^{0}		9.5×10^{-2}	4.8×10^{-1}	1.1×10^{0}		2.6×10^{-4}	5.0×10^{-2}		
LANL[p]	2.9×10^{1}		7.6×10^{0}		3.7×10^{1}		--		5.7×10^{-1}	1.7×10^{0}		3.1×10^{-4}	1.1×10^{-1}
ORNL	1.3×10^{-2}	1.5×10^{0}		1.5×10^{0}		3.4×10^{-2}	1.7×10^{-1}	3.8×10^{-1}	9.2×10^{-5}	1.7×10^{-2}			
NTS[p]	1.8×10^{0}		3.6×10^{-1}	2.2×10^{0}		--		6.4×10^{-3}	1.9×10^{-2}	3.5×10^{-6}	1.3×10^{-3}		
ANLE	1.4×10^{-3}	8.6×10^{-2}	8.7×10^{-2}	1.4×10^{-3}	7.0×10^{-3}	1.6×10^{-2}	3.9×10^{-6}	7.3×10^{-4}					
LLNL	6.9×10^{-3}	4.5×10^{-1}	4.6×10^{-1}	5.2×10^{-3}	2.6×10^{-2}	5.9×10^{-2}	1.4×10^{-5}	2.7×10^{-3}					
MOUND	7.9×10^{-4}	5.2×10^{-2}	5.3×10^{-2}	8.2×10^{-4}	4.1×10^{-3}	9.0×10^{-3}	2.3×10^{-6}	4.3×10^{-4}					
TOTAL (WIPP)	--		--		--		3.7×10^{-1}	2.4×10^{0}[q]	5.9×10^{0}		1.3×10^{-3}	3.0×10^{-1}	
Remote-Handled													
INEL	2.6×10^{-2}	2.0×10^{0}		2.0×10^{0}		4.1×10^{-2}	2.0×10^{-1}	3.7×10^{-1}	6.6×10^{-5}	1.7×10^{-2}			
Hanford	1.1×10^{-1}	2.0×10^{1}		2.0×10^{1}		6.6×10^{-1}	3.3×10^{0}		5.9×10^{0}		1.1×10^{-3}	2.8×10^{-1}	
LANL[p]	1.4×10^{0}		8.1×10^{-1}	2.2×10^{0}		--		7.5×10^{-2}	1.8×10^{-1}	2.4×10^{-5}	1.5×10^{-2}		
ORNL	3.2×10^{-2}	2.5×10^{0}		2.5×10^{0}		2.5×10^{-1}	1.2×10^{0}		2.3×10^{0}		3.9×10^{-4}	1.0×10^{-1}	
ANLE	8.0×10^{-3}	6.1×10^{-1}	6.2×10^{-1}	1.3×10^{-2}	6.3×10^{-2}	1.1×10^{-1}	2.1×10^{-5}	5.2×10^{-3}					
TOTAL (WIPP)	--		--		--		9.6×10^{-1}	4.8×10^{0}[q]	8.9×10^{0}		1.6×10^{-3}	4.2×10^{-1}	

TABLE D.3.14 Concluded

Notes:

[a] The fraction of shipments a crew member is estimated to participate in is calculated based on an availability of 5,400 hours per year (225 days at 24 hours per day) and an average travel speed of 35 mph for truck and 20 mph for rail.

[b] Based on RADTRAN-II model, with an exposure distance of 13 ft for truck shipments and 492 ft for rail shipments.

[c] Based on line source exposure model (l/r) for 100 mile inspections, food stops and refueling stops:

	Exposure Time	Exposure Distance	Comments
Inspections	15 min	3.2 ft	
Food stops			
Dining	1 hr	66 ft	
Surveillance	1 hr	33 ft	
Refueling			
Near activities	20 min	16 ft	Refueling assumed to
Far activities	20 min	33 ft	occur every 850 miles

[d] Total crew member occupational dose will be monitored by a dosimetry program and doses to individuals will be maintained below DOE guidelines.

[e] Calculated using a line source exposure model, with an average exposure distance of 10 ft and an exposure time of 30 minutes, and assuming three shifts per day and that the individual works in same position for 10 years.

[f] Based on line source exposure model with one inspector exposed to 20 percent of all shipments for 1 hour per inspection at an average distance of 3.2 ft (1 m).

[g] Assumes member of public is delayed in traffic adjacent to shipment for one 30-minute period, at a distance of 3.2 ft (1 m). This calculation gives the upper bound for the actual radiation dose due to the usage of conservative assumptions, as discussed in Subsection D.3.2.1 and Appendix B.

[h] Calculated using RADTRAN-II model which assumes that individual is exposed to every waste shipment traveling at 15 mph at a distance of approximately 100 ft.

[i] Estimated exposure using a line source exposure model to a member of the public working at a truckstop (exposure distance of 65 ft and exposure duration of 2 hours) and assuming all trucks stop at that location, three shifts per day, and that individual works at location for 10 years.

[j] Maximum rail crew member exposure calculation based upon the maximum anticipated distance between railcar classification terminals from each shipment site to the WIPP. The distances used in this analysis are: INEL/1,200 mi, RFP/770 mi, HANF/1,910 mi, SRS/875 mi, ORNL/850 mi, ANLE/1,180 mi, LLNL/1,680 mi, Mound/1,220 mi.

[k] Individual crew member doses during stops for inspections and servicing (e.g., air hose connections) were calculated, assuming an exposure duration of 1 percent of the stop time at an exposure level equaling the TI value. A freight stop time of 0.033 hours per kilometer was used for conservatism.

[l] Calculated using line source model (1/r), with an average exposure distance of 33 ft (10 m) and an exposure duration of 2 hours for each shipment and assuming that there are three rotating yard crews, with an individual working 10 years in the same job.

[m] Assumed to be the same as for truck shipments since fewer rail shipments will be required but more items to inspect/survey per shipment.

[n] State inspector exposure parameters for rail are assumed to be the same as the truck mode, but with a reduced exposure time of 45 minutes, since no queue time is expected.

[o] Assumes individual is exposed to every waste shipment stopped at a train terminal, with an average exposure distance of 660 ft (200 m) for a duration of 20 hours. Dose rate calculated as a point source beyond 300 ft (approximately 5 times a railcar length) equaling 6.9×10^{-4} (TI).

[p] Waste shipments are limited to the truck mode.

[q] Arrival inspections.

TABLE D.3.30 RH bounding case accident inputs

Input factor	RADTRAN III	AIRDOS
Curies per RH cask	Same for each model:	Maximum allowed per thermal or fissile grams limits set by WAC: HANF 813 PE-Ci[a] (909 total Ci) INEL 836 PE-Ci (903 total Ci)
Release fraction	.0002 released of all Ci as airborne, respirable fraction for both models.	
Release height	Ground release	Ground release (3.5 m)
Weather	Same, Stability Class F for both models	
Wind speed	1 meter per second	2 meters per second
Population fraction	Same for both models (Urban: 3861 people per square kilometer)	
Directly calculated Pathway doses	Inhalation Resuspension Groundshine Cloudshine	Inhalation ---------- Groundshine Cloudshine
Calculation for "Maximum Individual" Directly	No	Yes

[a] PE-Ci is plutonium equivalent curies calculated by using weighting factors in Appendix F.

TABLE D.3.31 CH bounding case accident results:

Truck accident (CEDE person-rem)

Model	Site	Pop. zone	Resusp. I[a]	Resusp. II[b]	Inhal.	Groundshine	Cloudshine	Ingestion	Total w/Res. I[c]	Total w/Res. II[c]	LCF w/Res. I[d]	LCF w/Res. II[d]	Max. indiv. dose (rem)
RADTRAN III[e]	LANL	Urban	1.30×10^{5}	2.07×10^{4}	2.86×10^{4}	2.54×10^{0}	2.53×10^{-4}	0	1.59×10^{5}	4.39×10^{4}	44.5	12.3	---
AIRDOS[f]	LANL	Urban	------	------	3.52×10^{4}	2.40×10^{0}	2.40×10^{-4}	0	3.52×10^{4}	3.52×10^{4}	9.9	9.9	0.16
RADTRAN III	LANL	Suburban	4.01×10^{4}	6.39×10^{3}	8.82×10^{3}	7.84×10^{-1}	7.79×10^{-5}	0	4.89×10^{4}	1.52×10^{4}	13.7	4.3	---
AIRDOS	LANL	Suburban	------	------	6.55×10^{3}	4.46×10^{-1}	4.46×10^{-5}	0	6.55×10^{3}	6.55×10^{3}	1.8	1.8	0.16
RADTRAN III	SRS	Urban	1.30×10^{5}	2.08×10^{4}	2.87×10^{4}	3.42×10^{-1}	2.33×10^{-6}	0	1.58×10^{5}	4.95×10^{4}	44.2	13.9	---
AIRDOS	SRS	Urban	------	------	3.51×10^{4}	1.50×10^{-1}	2.50×10^{-6}	0	3.51×10^{4}	3.51×10^{4}	9.8	9.8	0.16
RADTRAN III	INEL	Urban	1.47×10^{5}	2.34×10^{4}	3.24×10^{4}	3.87×10^{0}	3.76×10^{-4}	0	1.80×10^{5}	5.58×10^{4}	50.4	15.6	---
AIRDOS	INEL	Urban	-----	------	3.97×10^{4}	3.50×10^{0}	3.50×10^{-3}	0	3.97×10^{4}	3.97×10^{4}	11.1	11.1	0.18

[a] RADTRAN III using a resuspension half life of 365 days.

[b] A more realistic resuspension half life might be 60 days, because material is either cleaned up or washed away.

[c] Total CEDE using each respective resuspension dose.

[d] Conversion: 1 person-rem = 2.8×10^{-4} LCFs (shown for each total CEDE using the two resuspension doses).

[e] RADTRAN III does not directly calculate maximum dose to the individual.

[f] AIRDOS does not calculate resuspension doses.

TABLE D.3.32 CH bounding case accident results:

Rail accident (CEDE person-rem)

Model	Site	Pop. zone	Resusp. I[a]	Resusp. II[b]	Inhal.	Groundshine	Cloudshine	Ingestion	Total w/Res. I[c]	Total w/Res. II[c]	LCF w/Res. I[d]	LCF w/Res. II[d]	Max. indiv. dose (rem)
RADTRAN III[e]	SRS	Urban	2.60×10^{5}	4.16×10^{4}	5.74×10^{4}	6.84×10^{-1}	4.66×10^{-6}	0	3.16×10^{5}	9.90×10^{4}	88.5	27.7	---
AIRDOS[f]	SRS	Urban	------	------	7.02×10^{4}	3.00×10^{-1}	5.00×10^{-6}	0	7.02×10^{4}	7.02×10^{4}	19.7	19.7	0.32
RADTRAN III	INEL	Urban	2.94×10^{5}	4.68×10^{4}	6.48×10^{4}	7.74×10^{0}	7.52×10^{-4}	0	3.60×10^{5}	1.12×10^{5}	100.8	31.4	---
AIRDOS	INEL	Urban	-----	------	7.94×10^{4}	7.00×10^{0}	7.00×10^{-3}	0	7.94×10^{4}	7.94×10^{4}	22.2	22.2	0.36
RADTRAN III[e]	LANL	Urban											
AIRDOS[f]	LANL	Urban											
RADTRAN III	LANL	Suburban											
AIRDOS	LANL	Suburban											

[a] RADTRAN III using a resuspension half life of 365 days.

[b] A more realistic resuspension half life might be 60 days, because material is either cleaned up or washed away.

[c] Total CEDE using each respective resuspension dose.

[d] Conversion: 1 person-rem = 2.8×10^{-4} LCFs (shown for each total CEDE using the two resuspension doses).

[e] RADTRAN III does not directly calculate maximum dose to the individual.

[f] AIRDOS does not calculate resuspension doses.

TABLE D.3.33 RH bounding case accident results:

Truck accident (CEDE person-rem)

Model	Site	Pop. zone	Resusp. I[a]	Resusp. II[b]	Inhal.	Groundshine	Cloudshine	Ingestion	Total w/Res. I[c]	Total w/Res. II[c]	LCF w/Res. I[d]	LCF w/Res. II[d]	Max. indiv. dose (rem)
RADTRAN III[e]	HANF	Urban	3.04×10^{3}	4.83×10^{2}	6.66×10^{2}	1.60×10^{1}	1.22×10^{-3}	0	3.72×10^{3}	1.16×10^{3}	1.0	0.3	---
AIRDOS[f]	HANF	Urban	------	------	8.81×10^{2}	1.84×10^{1}	3.25×10^{-1}	0	8.99×10^{2}	8.99×10^{2}	0.25	0.25	0.004
RADTRAN III	INEL	Urban	3.29×10^{4}	1.50×10^{3}	7.20×10^{3}	1.45×10^{0}	1.65×10^{-4}	0	4.01×10^{4}	1.24×10^{4}	11.2	3.5	---
AIRDOS	INEL	Urban	------	------	9.00×10^{3}	1.22×10^{0}	1.90×10^{-2}	0	9.00×10^{3}	9.00×10^{3}	2.5	2.5	0.04

[a] RADTRAN III using a resuspension half life of 365 days.

[b] A more realistic resuspension half life might be 60 days, because material is either cleaned up or washed away.

[c] Total CEDE using each respective resuspension dose.

[d] Conversion: 1 person-rem = 2.8×10^{-4} LCFs (shown for each total CEDE using the two resuspension doses).

[e] RADTRAN III does not directly calculate maximum dose to the individual.

[f] AIRDOS does not calculate resuspension doses.

TABLE D.3.34 RH bounding case accident results:

Rail accident (CEDE person-rem)

Model	Site	Pop. zone	Resusp. I[a]	Resusp. II[b]	Inhal.	Groundshine	Cloudshine	Ingestion	Total w/Res. I[c]	Total w/Res. II[c]	LCF w/Res. I[d]	LCF w/Res. II[d]	Max. indiv. dose (rem)
RADTRAN III[e]	HANF	Urban	6.08×10^{3}	9.66×10^{2}	1.33×10^{3}	3.20×10^{1}	2.44×10^{-3}	0	7.44×10^{3}	2.32×10^{3}	2.1	0.6	---
AIRDOS[f]	HANF	Urban	------	------	1.76×10^{3}	3.68×10^{1}	6.50×10^{-1}	0	1.80×10^{3}	1.80×10^{3}	0.5	0.5	0.008
RADTRAN III	INEL	Urban	6.58×10^{4}	3.00×10^{3}	1.44×10^{3}	2.90×10^{0}	3.30×10^{-4}	0	8.02×10^{4}	2.48×10^{4}	22.5	6.9	---
AIRDOS	INEL	Urban	------	------	1.80×10^{4}	2.44×10^{0}	3.80×10^{-2}	0	1.80×10^{4}	1.80×10^{4}	5.0	5.0	0.08

[a] RADTRAN III using a resuspension half life of 365 days.

[b] A more realistic resuspension half life might be 60 days, because material is either cleaned up or washed away.

[c] Total CEDE using each respective resuspension dose.

[d] Conversion: 1 person-rem = 2.8×10^{-4} LCFs (shown for each total CEDE using the two resuspension doses).

[e] RADTRAN III does not directly calculate maximum dose to the individual.

[f] AIRDOS does not calculate resuspension doses.

D.4 NONRADIOLOGICAL AND NONCHEMICAL CONSEQUENCES OF TRANSPORTATION

D.4.1 INTRODUCTION

The nonradiological and nonchemical consequences of transporting radioactive waste to the WIPP are discussed in this subsection. These impacts are the same as those resulting from transporting non-nuclear materials and involve accidents and resulting injuries and fatalities from transuranic waste transport and vehicle exhaust emission. The nonradiological and nonchemical impacts do not consider the characteristics of the cargo.

There are two types of nonradiological and nonchemical risks associated with projected TRU waste shipments. These are risks resulting from normal transportation and risks resulting from transportation accidents. The normal risks include the health risks in urban areas caused by the generation of nonradiological air pollutants by the carrier vehicles during waste shipments. Transportation accident risks include injuries and fatalities resulting from shipments that are totally unrelated to radiological and hazardous chemical risks resulting from projected accidents.

D.4.2 METHOD

Two methods were used to estimate the range of nonradiological and nonchemical risks. Using the first method, the risks of adverse urban area pollutant health effects and accident-related injuries and fatalities were calculated on a per shipment basis and a cumulative basis from unit risk factors described by Sandia National Laboratories (see Cashwell et al., 1986). These data were based on heavy truck and Class A rail statistics from the Research and Special Programs Administration of the U.S. Department of Transportation. Using the second method, risks of accident-related injuries and fatalities were calculated by estimating total WIPP lifetime shipment-miles for the truck and maximum rail alternatives and applying injury and fatality rates based on 1987-88 accident statistics from the Federal Railroad Administration (FRA) and from highway traffic statistics along the preferred WIPP highway routes. Tables D.4.1 through D.4.11 summarize risks estimated by the first method. Tables D.4.12 through D.4.14 summarize risks calculated by the second method.

D.4.2.1 Per-Shipment Risk Approach

Estimates of per shipment risk include the probability of adverse urban area pollutant health effects and accident-related injuries and fatalities of a single TRU waste shipment (round trip) to the WIPP. Cumulative risk estimates were determined by multiplying per shipment risks by average annual shipments for both the Proposed Action and Alternative Action. The estimated total number of shipments, both truck and rail, are summarized in Table D.4.1.

TABLE D.4.1 Estimated number of CH TRU and RH TRU waste shipments from generator and storage facilities to the WIPP

CH TRU	Total shipments[a]	
Facility	100% Truck	Maximum rail
Idaho National Engineering Laboratory	4,046	2.023
Rocky Flats Plant	7,608	3,804
Hanford Reservation	3,103	1,552
Savannah River Site	2,640	1,320
Los Alamos National Laboratory	2,065	2,065[c]
Oak Ridge National Laboratory	228	114
Nevada Test Site	80	80[c]
Argonne National Laboratory-East	14	7
Lawrence Livermore National Laboratory	969	485
Mound Laboratory	150	75
Total	20,903	11,525

RH TRU	Total shipments[b]	
Facility	100% Truck	Maximum rail
Idaho National Engineering Laboratory	487	244
Hanford Reservation	2,470	1,235
Los Alamos National Laboratory	101	101[c]
Oak Ridge National Laboratory	4,605	2,303
Argonne National Laboratory-East	300	150
Total	7,963	4,033

[a] Shipments based on 3 TRUPACT-IIs per truck shipment and 6 TRUPACT-IIs per railcar shipment. Shipments calculated from a drum volume of 0.2 m^3 x 14 drums/TRUPACT-IIs.
[b] Shipments based on 1 RH cask per truck shipment and 2 RH casks per railcar shipment. Shipments calculated from a canister volume of 0.89 m^3 x 1 canister/RH cask.
[c] LANL and NTS do not have access to rail, thus truck shipments are included in the maximum rail case.

The average distance and population fraction from Table D.4.2 are used with Table D.4.3 (Air Pollutant Unit Consequence Factors) and Table D.4.4 (Nonradiological and Nonchemical Unit Risk Factors) to calculate the per shipment nonradiological and nonchemical risk of CH TRU and RH TRU waste shipments from each facility for truck and rail alternatives. The air pollutant unit consequence factors represent the estimated additional urban area health effects from particulates and truck or locomotive emissions of sulfur dioxide during a shipment.

Calculated per shipment nonradiological and nonchemical risks for CH TRU and RH TRU shipments to WIPP are summarized in Table D.4.5. These risks include the impact of the return trip by either truck or rail from the WIPP to the generator or storage facility. Each travel mode alternative assumes the uniform maximum use of that mode by all facilities. Therefore, the mode alternatives are labeled as 100 percent truck, and maximum rail for those facilities that have access to rail. Los Alamos National Laboratory and the Nevada Test Site do not have access to rail, and thus, truck mode risks for these two facilities are listed with the maximum rail risks for the purpose of estimating the cumulative risk.

Total cumulative nonradiological and nonchemical CH TRU transportation risks are summarized in Tables D.4.6 and D.4.7 for the Test Phase and 20-year Disposal Phase of the Proposed Action. Tables D.4.8 and D.4.9 summarize the corresponding results for the Alternative Action. Tables D.4.10 and D.4.11 summarize the total cumulative nonradiological and nonchemical RH TRU transportation risks for both the Proposed Action and the Alternative Action.

D.4.2.2 **Lifetime Risk Approach**

During the preparation of the draft SEIS, State transportation departments were contacted and requested to provide estimates of actual annual (1987-1988) heavy truck accident injury and fatality rates per truck vehicle-mile along the WIPP preferred routes. Data received from the States are summarized by specific highway segments in Table D.4.12. Similar route specific accident data for potential rail routes were not available. Table D.4.13 summarizes forecasted percentages of TRU shipments by specific highway segments. These percentages are conservatively estimated by assuming no growth in total truck volumes over the life of the WIPP shipping campaign.

Averages of truck accident, injury and fatality rates by each State and for all affected States are summarized in Table D.4.13 and compared to statistics from the NRC (1977), Chem-Nuclear (1989), and Cashwell et al. (1986).

Table D.4.14 summarizes lifetime shipment-miles for combined CH and RH TRU shipments for the 100 percent truck and maximum rail alternatives for the Proposed Action and the Alternative Action. By using the 1987-1988 WIPP Route Highway System weighted average rates for the 100 percent truck alternative and injury and fatality rates from the Federal Railroad Administration (FRA, 1987) for the maximum rail alternative, total WIPP lifetime accident-related risks were calculated. For comparison purposes, risks of injuries and fatalities calculated using the data from Cashwell et al. (1986) are shown in Table D.4.14.

TABLE D.4.2 Average distances to the WIPP and percent of travel in various population zones[a]

	Average distance	Population zone		
	Miles	R	S	U
Truck				
Idaho National Engineering Laboratory	1521	85.0	13.8	1.2
Rocky Flats Plant	874	82.3	15.7	2.0
Hanford Reservation	1913	85.7	13.4	0.9
Savannah River Site	1585	74.3	25.1	0.6
Los Alamos National Laboratory	343	90.1	9.9	0.0
Oak Ridge National Laboratory	1350	78.6	20.7	0.7
Nevada Test Site	1286	86.8	11.2	2.0
Argonne National Laboratory-East	1387	78.1	21.8	0.1
Lawrence Livermore National Laboratory	1458	86.2	10.1	3.7
Mound Laboratory	1472	75.4	24.1	0.5
Rail				
Idaho National Engineering Laboratory	1761	89.5	9.8	0.7
Rocky Flats Plant	1098	86.7	11.6	1.7
Hanford Reservation	2296	87.8	11.5	0.7
Savannah River Site	1915	76.0	22.4	1.6
Oak Ridge National Laboratory	1630	79.8	18.9	1.3
Argonne National Laboratory-East	1469	81.6	17.0	1.4
Lawrence Livermore National Laboratory	1873	85.0	14.3	0.8
Mound Laboratory	1677	76.8	21.3	1.9

[a] Mean population densities are utilized and correspond to:

R = Rural (6 persons/km^2)
S = Suburban (719 persons/km^2)
U = Urban (3861 persons/km^2).

Source: Madsen et al., 1983.

TABLE D.4.3 Air pollutant unit consequence factors[a]

	Health effects per mile	
	Truck	Rail
Source	(LCF/Mi)	(LCF/Mi)
Pollutants (particulates & sulfur dioxide)	1.6×10^{-7} (urban travel only)	2.1×10^{-7} (urban travel only)

LCF = Latent cancer fatalities.

[a] Rao et al. (R.K. Rao, E. L. Wilmot, and R. E. Luna), 1982. Nonradiological Impacts of Transporting Radioactive Material. SAND81-1703, TTC-0236, Sandia National Laboratories Albuquerque, NM.

TABLE D.4.4 Nonradiological and nonchemical unit risk factors[a]

Mode	Zone	LCF/Mi[a]	Injuries/Mi[b]	Fatalities/Mi[b]
	Rural	0	1.33×10^{-6}	1.09×10^{-7}
Truck	Suburban	0	6.32×10^{-7}	2.69×10^{-8}
	Urban	1.6×10^{-7}	6.16×10^{-7}	1.54×10^{-8}
	Rural	0	4.78×10^{-7}	4.54×10^{-8}
Rail	Suburban	0	4.78×10^{-7}	4.54×10^{-8}
	Urban	2.1×10^{-7}	4.78×10^{-7}	4.54×10^{-8}

LCF - Latent cancer fatalities.

Sources:

[a] Rao et al. (R.K. Rao, E. L. Wilmot, and R. E. Luna), 1982. Nonradiological Impacts of Transporting Radioactive Material. SAND81-1703, TTC-0236, Sandia National Laboratories, Albuquerque, NM.

[b] Cashwell, Jon W., et. al., 1986, Transportation Impacts of the Commercial Radioactive Waste Management Program, Appendix 4, Tables 4-4A and 4-4B. SAND85-2715, TTC-0663, Sandia National Laboratories, Albuquerque, NM. Nonradiological unit risk factors determined from U.S. Dept. of Transportation, Research and Special Programs Administration, Transportation Systems Center, 1986, National Transportation Statistics, Annual Report, 1986, Report No. DOT-TSC-RSPA-86-3, "Truck Profile, Heavy Truck Category" and "Rail Profile, Class I Railroads Category," for 1983 and 1984 calendar year.

TABLE D.4.5 Per shipment nonradiological risk of waste shipments

		Truck			Rail		
		Normal transportation	Accident case		Normal transportation	Accident case	
Facility	Zone	LCF[a,b]	Fatalities	Injuries	LCF	Fatalities	Injuries
INEL	Rural	0.00×10^{0}	2.82×10^{-4}	3.44×10^{-3}	0.00×10^{0}	1.43×10^{-4}	1.51×10^{-3}
	Suburban	0.00×10^{0}	1.13×10^{-5}	2.65×10^{-4}	0.00×10^{0}	1.57×10^{-5}	1.65×10^{-4}
	Urban	5.84×10^{-6}	5.62×10^{-7}	2.25×10^{-5}	5.18×10^{-6}	1.12×10^{-6}	1.18×10^{-5}
RFP	Rural	0.00×10^{0}	1.57×10^{-4}	1.91×10^{-3}	0.00×10^{0}	8.64×10^{-5}	9.10×10^{-4}
	Suburban	0.00×10^{0}	7.38×10^{-6}	1.73×10^{-4}	0.00×10^{0}	1.16×10^{-5}	1.22×10^{-4}
	Urban	5.59×10^{-6}	5.38×10^{-7}	2.15×10^{-5}	7.84×10^{-6}	1.69×10^{-6}	1.78×10^{-5}
HANF	Rural	0.00×10^{0}	3.57×10^{-4}	4.36×10^{-3}	0.00×10^{0}	1.83×10^{-4}	1.93×10^{-3}
	Suburban	0.00×10^{0}	1.38×10^{-5}	3.24×10^{-4}	0.00×10^{0}	2.40×10^{-5}	2.52×10^{-4}
	Urban	5.51×10^{-6}	5.30×10^{-7}	2.12×10^{-5}	6.75×10^{-6}	1.46×10^{-6}	1.54×10^{-5}
SRS	Rural	0.00×10^{0}	2.57×10^{-4}	3.13×10^{-3}	0.00×10^{0}	1.32×10^{-4}	1.39×10^{-3}
	Suburban	0.00×10^{0}	2.14×10^{-5}	5.03×10^{-4}	0.00×10^{0}	3.89×10^{-5}	4.10×10^{-4}
	Urban	3.04×10^{-6}	2.93×10^{-7}	1.17×10^{-5}	1.29×10^{-5}	2.78×10^{-6}	2.93×10^{-5}
LANL	Rural	0.00×10^{0}	6.74×10^{-5}	8.22×10^{-4}			
	Suburban	0.00×10^{0}	1.83×10^{-6}	4.29×10^{-5}	[d]	[d]	[d]
	Urban	[c]	0.00×10^{0}	0.00×10^{0}			
ORNL	Rural	0.00×10^{0}	2.31×10^{-4}	2.82×10^{-3}	0.00×10^{0}	1.18×10^{-4}	1.24×10^{-3}
	Suburban	0.00×10^{0}	1.50×10^{-5}	3.53×10^{-4}	0.00×10^{0}	2.80×10^{-5}	2.95×10^{-4}
	Urban	3.02×10^{-6}	2.91×10^{-7}	1.16×10^{-5}	8.90×10^{-6}	1.92×10^{-6}	2.03×10^{-5}
NTS	Rural	0.00×10^{0}	2.43×10^{-4}	2.97×10^{-3}			
	Suburban	0.00×10^{0}	7.75×10^{-6}	1.82×10^{-4}	[d]	[d]	[d]
	Urban	8.23×10^{-6}	7.92×10^{-7}	3.17×10^{-5}			
ANLE	Rural	0.00×10^{0}	2.36×10^{-4}	2.88×10^{-3}	0.00×10^{0}	1.09×10^{-4}	1.15×10^{-3}
	Suburban	0.00×10^{0}	1.63×10^{-5}	3.82×10^{-4}	0.00×10^{0}	2.27×10^{-5}	2.39×10^{-4}
	Urban	4.44×10^{-7}	4.27×10^{-8}	1.71×10^{-6}	8.64×10^{-6}	1.87×10^{-6}	1.97×10^{-5}
LLNL	Rural	0.00×10^{0}	2.74×10^{-4}	3.34×10^{-3}	0.00×10^{0}	1.45×10^{-4}	1.52×10^{-3}
	Suburban	0.00×10^{0}	7.92×10^{-6}	1.86×10^{-4}	0.00×10^{0}	2.43×10^{-5}	2.56×10^{-4}
	Urban	1.73×10^{-5}	1.66×10^{-6}	6.65×10^{-5}	6.29×10^{-6}	1.36×10^{-6}	1.43×10^{-5}
Mound	Rural	0.00×10^{0}	2.42×10^{-4}	2.95×10^{-3}	0.00×10^{0}	1.17×10^{-4}	1.23×10^{-3}
	Suburban	0.00×10^{0}	1.91×10^{-5}	4.48×10^{-4}	0.00×10^{0}	3.24×10^{-5}	3.41×10^{-4}
	Urban	2.36×10^{-6}	2.27×10^{-7}	9.07×10^{-6}	1.34×10^{-5}	2.89×10^{-6}	3.05×10^{-5}

[a] Numbers are expressed in scientific notation $2.82 \times 10^{-4} = 0.0282$.

[b] Latent cancer fatalities resulting from incremental vehicle pollution in urban population zones.

[c] The preferred route from LANL to WIPP passes through no urban population zones.

[d] LANL and NTS have no rail access.

TABLE D.4.8 Total transportation risk for Alternative Action, CH truck mode

Facility	Zone	Number of shipments	Normal transportation LCFs	Accident case: Fatalities	Accident case: Injuries
INEL	Rural	4046	0.00×10^{0}	1.1×10^{0}	1.4×10^{1}
	Suburban		0.00×10^{0}	4.6×10^{-2}	1.1×10^{0}
	Urban		2.4×10^{-2}	2.3×10^{-3}	9.1×10^{-2}
RFP	Rural	7608	0.00×10^{0}	1.2×10^{0}	1.5×10^{1}
	Suburban		0.00×10^{0}	5.6×10^{-2}	1.3×10^{0}
	Urban		4.3×10^{-2}	4.1×10^{-3}	1.6×10^{-1}
HANF	Rural	3103	0.00×10^{0}	1.1×10^{0}	1.4×10^{1}
	Suburban		0.00×10^{0}	4.3×10^{-2}	1.0×10^{0}
	Urban		1.7×10^{-2}	1.6×10^{-3}	6.6×10^{-2}
SRS	Rural	2640	0.00×10^{0}	6.8×10^{-1}	8.3×10^{0}
	Suburban		0.00×10^{0}	5.6×10^{-2}	1.3×10^{0}
	Urban		8.0×10^{-3}	7.7×10^{-4}	3.1×10^{-2}
LANL	Rural	2065	[a]	1.4×10^{-1}	1.7×10^{0}
	Suburban		[a]	3.8×10^{-3}	8.9×10^{-2}
	Urban		[a]	[a]	[a]
ORNL	Rural	228	0.00×10^{0}	5.3×10^{-2}	6.4×10^{-1}
	Suburban		0.00×10^{0}	3.4×10^{-3}	8.0×10^{-2}
	Urban		6.9×10^{-4}	6.6×10^{-5}	2.6×10^{-3}
NTS	Rural	80	0.00×10^{0}	1.9×10^{-2}	2.4×10^{-1}
	Suburban		0.00×10^{0}	6.2×10^{-4}	1.5×10^{-2}
	Urban		6.6×10^{-4}	6.3×10^{-5}	2.5×10^{-3}
ANL/E	Rural	14	0.00×10^{0}	3.3×10^{-3}	4.0×10^{-2}
	Suburban		0.00×10^{0}	2.3×10^{-4}	5.3×10^{-3}
	Urban		6.2×10^{-6}	6.0×10^{-7}	2.4×10^{-5}
LLNL	Rural	969	0.00×10^{0}	2.7×10^{-1}	3.2×10^{0}
	Suburban		0.00×10^{0}	7.7×10^{-3}	1.8×10^{-1}
	Urban		1.7×10^{-2}	1.6×10^{-3}	6.4×10^{-2}
Mound	Rural	150	0.00×10^{0}	3.6×10^{-2}	4.4×10^{-1}
	Suburban		0.00×10^{0}	2.9×10^{-3}	6.7×10^{-2}
	Urban		3.5×10^{-4}	3.4×10^{-5}	1.4×10^{-3}
	Total	20,903	1.1×10^{-1}	4.9×10^{0}	6.3×10^{1}

[a] The preferred route from LANL to WIPP passes through no urban population zones.

TABLE D.4.9 Total transportation risk for Alternative Action, CH rail mode

Facility	Zone	Number of shipments	Normal transportation LCFs	Accident case	
				Fatalities	Injuries
INEL	Rural	2023	0	2.9×10^{-1}	3.1×10^{0}
	Suburban		0	3.2×10^{-2}	3.3×10^{-1}
	Urban		1.0×10^{-2}	2.3×10^{-3}	2.4×10^{-2}
RFP	Rural	3804	0	3.3×10^{-1}	3.5×10^{0}
	Suburban		0	4.4×10^{-2}	4.6×10^{-1}
	Urban		3.0×10^{-2}	6.4×10^{-3}	6.8×10^{-2}
HANF	Rural	1552	0	2.8×10^{-1}	3.0×10^{0}
	Suburban		0	3.7×10^{-2}	3.9×10^{-1}
	Urban		1.0×10^{-2}	2.3×10^{-3}	2.4×10^{-2}
SRS	Rural	1320	0	1.7×10^{-1}	1.8×10^{0}
	Suburban		0	5.1×10^{-2}	5.4×10^{-1}
	Urban		1.7×10^{-2}	3.7×10^{-3}	3.9×10^{-2}
LANL[b]	Rural	2065	a	1.4×10^{-1}	1.7×10^{0}
	Suburban		a	3.8×10^{-3}	8.9×10^{-2}
	Urban		a	[a]	[a]
ORNL	Rural	114	0	1.3×10^{-2}	1.4×10^{-1}
	Suburban		0	3.2×10^{-3}	3.4×10^{-2}
	Urban		1.0×10^{-3}	2.2×10^{-4}	2.3×10^{-3}
NTS[b]	Rural	80	0	1.9×10^{-2}	2.4×10^{-1}
	Suburban		0	6.2×10^{-4}	1.5×10^{-2}
	Urban		6.6×10^{-4}	6.3×10^{-5}	2.5×10^{-3}
ANL/E	Rural	7	0	7.6×10^{-4}	8.1×10^{-3}
	Suburban		0	1.6×10^{-4}	1.7×10^{-3}
	Urban		6.0×10^{-5}	1.3×10^{-5}	1.4×10^{-4}
LLNL	Rural	485	0	7.0×10^{-2}	7.4×10^{-1}
	Suburban		0	1.2×10^{-2}	1.2×10^{-1}
	Urban		3.1×10^{-3}	6.6×10^{-4}	6.9×10^{-3}
MOUND	Rural	75	0	8.8×10^{-3}	9.2×10^{-2}
	Suburban		0	2.4×10^{-3}	2.6×10^{-2}
	Urban		1.0×10^{-3}	2.2×10^{-4}	2.3×10^{-3}
TOTAL		11525	7.3×10^{-2}	1.5×10^{0}	1.6×10^{1}

[a] The preferred route from LANL to WIPP passes through no urban population zones.

[b] For the maximum rail case, shipments from LANL and NTS are made by truck.

TABLE D.4.10 Total transportation risk for Proposed Action and Alternative Action, RH truck mode

Facility	Zone	Number of shipments	Normal transportation LCFs	Accident case Fatalities	Accident case Injuries
INEL	Rural	487	0.00×10^{0}	1.4×10^{-1}	1.7×10^{0}
	Suburban		0.00×10^{0}	5.5×10^{-3}	1.3×10^{-1}
	Urban		2.8×10^{-3}	2.7×10^{-4}	1.1×10^{-2}
HANF	Rural	2470	0.00×10^{0}	8.8×10^{-1}	1.1×10^{1}
	Suburban		0.00×10^{0}	3.4×10^{-2}	8.0×10^{-1}
	Urban		1.4×10^{-2}	1.3×10^{-3}	5.2×10^{-2}
LANL	Rural	101	[a]	6.8×10^{-3}	8.3×10^{-2}
	Suburban		[a]	1.8×10^{-4}	4.3×10^{-3}
	Urban		[a]	[a]	[a]
ORNL	Rural	4605	0.00×10^{0}	1.1×10^{0}	1.3×10^{1}
	Suburban		0.00×10^{0}	6.9×10^{-2}	1.6×10^{0}
	Urban		1.4×10^{-2}	1.3×10^{-3}	5.3×10^{-2}
ANL/E	Rural	300	0.00×10^{0}	7.1×10^{-2}	8.6×10^{-1}
	Suburban		0.00×10^{0}	4.9×10^{-3}	1.1×10^{-1}
	Urban		1.3×10^{-4}	1.3×10^{-5}	5.1×10^{-4}
	Total	7963	6.2×10^{-2}	2.3×10^{0}	2.9×10^{1}

[a] The preferred route from LANL to WIPP passes through no urban population zones.

TABLE D.4.11 Total transportation risk for Proposed Action and Alternative Action, RH rail mode

Facility	Zone	Number of shipments	Normal transportation LCFs	Accident case Fatalities	Accident case Injuries
INEL	Rural	244	0.00×10^{0}	3.5×10^{-2}	3.7×10^{-1}
	Suburban		0.00×10^{0}	3.8×10^{-3}	4.0×10^{-2}
	Urban		1.3×10^{-3}	2.7×10^{-4}	2.9×10^{-3}
HANF	Rural	1235	0.00×10^{0}	2.3×10^{-1}	2.4×10^{0}
	Suburban		0.00×10^{0}	3.0×10^{-2}	3.1×10^{-1}
	Urban		8.3×10^{-3}	1.8×10^{-3}	1.9×10^{-2}
LANL	Rural	101	[a]	6.8×10^{-3}	8.3×10^{-2}
	Suburban		[a]	1.8×10^{-4}	4.3×10^{-3}
	Urban		[a]	[a]	[a]
ORNL	Rural	2303	0.00×10^{0}	2.7×10^{-1}	2.9×10^{0}
	Suburban		0.00×10^{0}	6.4×10^{-2}	6.8×10^{-1}
	Urban		2.0×10^{-2}	4.4×10^{-3}	4.7×10^{-2}
ANL/E	Rural	150	0.00×10^{0}	1.6×10^{-2}	1.7×10^{-1}
	Suburban		0.00×10^{0}	3.4×10^{-3}	3.6×10^{-2}
	Urban		1.3×10^{-3}	2.8×10^{-4}	3.0×10^{-3}
	Total	4033	3.1×10^{-2}	6.6×10^{-1}	7.1×10^{0}

[a] No rail access at LANL. Consequences shown are for truck transport of LANL RH TRU waste. No LANL shipments are planned through urban areas.

TABLE D.4.13 Traffic statistics: Recent year Statewide and systemwide annual weighted averages

Jurisdiction/ statistics source	Route[a] miles	Accident rate/ truck vehicle-mile	Injury rate/ truck vehicle-mile	Fatality rate/ truck vehicle-mile
New Mexico	888.1	7.95×10^{-7}	2.97×10^{-7}	1.11×10^{-8}
Colorado	312.2	1.24×10^{-6}	N/A	N/A
Wyoming	368.3	1.26×10^{-6}	6.10×10^{-7}	3.71×10^{-8}
Utah	167.9	9.16×10^{-7}	N/A	N/A
Idaho	368.1	7.05×10^{-7}	5.03×10^{-7}	4.26×10^{-8}
Oregon	213	2.86×10^{-7}	2.08×10^{-7}	0.00×10^{0}
Washington	50.2	1.09×10^{-6}	6.99×10^{-7}	0.00×10^{0}
Arizona	359.3	7.28×10^{-7}	3.73×10^{-7}	1.77×10^{-8}
California	625.7	6.68×10^{-7}	3.71×10^{-7}	2.53×10^{-8}
Nevada	142.6	2.36×10^{-6}	9.03×10^{-7}	1.56×10^{-8}
Texas	824	6.94×10^{-7}	N/A	2.29×10^{-8}
Oklahoma	539.1	8.02×10^{-7}	3.26×10^{-7}	2.31×10^{-8}
Missouri	285.9	2.49×10^{-6}	N/A	N/A
Illinois	428.3	N/A	N/A	N/A
Indiana	157	N/A	N/A	N/A
Ohio	53.2	2.16×10^{-6}	9.33×10^{-7}	1.10×10^{-8}
Arkansas	285.6	6.11×10^{-7}	2.01×10^{-7}	3.16×10^{-8}
Tennessee	363.8	7.46×10^{-7}	3.88×10^{-7}	1.38×10^{-8}
Louisiana	188.5	1.78×10^{-5}	4.34×10^{-7}	1.26×10^{-8}
Mississippi	156.1	8.03×10^{-8}	2.68×10^{-8}	3.29×10^{-9}
Alabama	218.1	4.81×10^{-7}	1.86×10^{-7}	1.38×10^{-8}
Georgia	212.2	1.02×10^{-1}	4.52×10^{-7}	2.03×10^{-8}
South Carolina	27.4	2.03×10^{-6}	6.02×10^{-7}	0.00×10^{0}
Weighted avg.[b]		1.37×10^{-6}	3.75×10^{-7}	1.98×10^{-8}
Systemwide		6649.3 Miles	5059.3 Miles	5983.3 Miles
NUREG-0170 (1977)		1.70×10^{-6}		
Chem-Nuclear (1989)		1.16×10^{-6}		
Cashwell et al. (1986)				
Rural			1.33×10^{-6}	1.09×10^{-7}
Suburban			6.32×10^{-7}	2.69×10^{-8}
Urban			6.16×10^{-7}	1.54×10^{-8}

[a] Only route miles for which traffic data was collected is listed.

[b] Excludes States and route segments of States where insufficient truck accident, truck injury, and truck fatality data was available.

N/A = not available.

TABLE D.4.14Summary of nonradiological and nonchemical impacts: Traffic accidents, injuries, and fatalities

A. WIPP shipment-miles summary statistics: CH and RH combined

Proposed Action		Alternative Action	
Mode: 100% Truck	Shipment-miles	Mode: 100% Truck	Shipment-miles
Test Phase and Disposal Phase	75,658,244	Test Phase and Disposal Phase	75,658,244
Mode: Maximum rail		Mode: Maximum rail	
Test Phase (all truck)	5,139,642		
Disposal Phase (rail, 8 sites)	34,506,160	Disposal Phase (rail, 8 sites)	44,600,508
Disposal Phase (truck, 2 sites)	1,529,058	Disposal Phase (truck, 2 sites)	1,691,536

B. Comparison of WIPP lifetime risks by traffic statistics source

B.1 Proposed Action - Mode: 100% Truck

Statistics	Accidents		Injuries		Fatalities	
Source	Rate/Mile	Total	Rate/Mile	Total	Rate/Mile	Total
Cashwell et al. (1986)						
(SEIS Tables D.4.6, D.4.10)				92.3		7.18
NUREG 0170 (1977)	1.70×10^{-6}	129.				
Chem-Nuclear (1989)	1.16×10^{-6}	88.0				
WIPP route highway	1.37×10^{-6}	104.	3.75×10^{-7}	28.0	1.98×10^{-8}	1.50
system (1987-1988)						

B.2 Proposed Action - Mode: Maximum rail

Statistics	Accidents		Injuries		Fatalities	
Source	Rate/Mile	Total	Rate/Mile	Total	Rate/Mile	Total
Cashwell et al. (1986)						
(SEIS Tables D.4.7, D.4.11) 2.54				28.4		
NUREG 0170 (1977)						
Test Phase (truck)	1.70×10^{-6}	8.74				
Disposal Phase (rail, 8 sites) 1.50×10^{-6}	51.8					
Disposal Phase (truck,						

2 sites)	1.70×10^{-6}	2.60
TOTAL		63.1

TABLE D.4.14 Continued

B. Comparison of WIPP lifetime risks by traffic statistics source

B.2 Proposed Action - Mode: Maximum rail, continued

Statistics Source	Accidents Rate/Mile	Total	Injuries Rate/Mile	Total	Fatalities Rate/Mile	Total
WIPP route highway system (1987-1988)/ Fed. R.R. Admin. (1987)[a]						
Test Phase (truck)	1.37×10^{-6}	7.04	3.75×10^{-7}	1.93	1.98×10^{-8}	0.102
Disposal Phase (rail, 8 sites)	4.55×10^{-6}	157.00	1.05×10^{-6}	36.2	1.14×10^{-7}	3.93
Disposal Phase (truck, 2 sites)	1.37×10^{-6}	2.09	3.75×10^{-7}	0.57	1.98×10^{-8}	0.030
TOTAL		166.		38.7		4.06

B.3 Alternative Action - Mode: 100% Truck

Statistics Source	Accidents Rate/Mile	Total	Injuries Rate/Mile	Total	Fatalities Rate/Mile	Total
Cashwell et al. (1986)						
(SEIS Tables D.4.8, D.4.10)				92.0		7.20
NUREG 0170 (1977)	1.70×10^{-6}	129.				
Chem-Nuclear (1989)	1.16×10^{-6}	88.0				
WIPP route highway system (1987-1988)	1.37×10^{-6}	104.	3.75×10^{-7}	28.0	1.98×10^{-8}	1.50

B.4 Alternative Action - Mode: Maximum rail

Statistics Source	Accidents Rate/Mile	Total	Injuries Rate/Mile	Total	Fatalities Rate/Mile	Total
Cashwell et al. (1986)						
(SEIS Tables D.4.9, D.4.11)				23.1		2.16
NUREG 0170 (1977)						
Disposal Phase (rail, 8 sites)	1.50×10^{-6}	66.9				
Disposal Phase (truck, 2 sites)	1.70×10^{-6}	2.88				

TOTAL	69.8

TABLE D.4.14 Concluded

B. Comparison of WIPP lifetime risks by traffic statistics source

B.4 Alternative Action - Mode: Maximum rail, continued

Statistics Source	Accidents Rate/Mile	Accidents Total	Injuries Rate/Mile	Injuries Total	Fatalities Rate/Mile	Fatalities Total
WIPP route highway system (1987-1988) Fed. R.R. Admin. (1987)[a]						
Disposal Phase (rail, 8 sites)	4.55×10^{-6}	203.	1.05×10^{-6}	46.8	1.14×10^{-7}	5.08
Disposal Phase (truck, 2 sites)	1.37×10^{-6}	2.32	3.75×10^{-7}	0.634	1.98×10^{-8}	0.0335
TOTAL		205.		47.4		5.11

[a] See Tables 1 (p. 5) and 8 (p. 16) of reference, "Accident/Incident Bulletin No. 156, Calendar Year 1987," U.S. DOT, Federal Railroad Administration Office of Safety, July, 1988.

D.4.3 **RESULTS**

D.4.3.1 **Results from Per-Shipment Risk Approach**

The results in Table D.4.5 show very small per shipment nonradiological and nonchemical risks for all facilities. The volumes of particulates and sulfur dioxide emitted by a single truck or rail shipment in an urban area are so small that one million or more similar pollutant generating shipments would be needed simultaneously to achieve the minimum required pollutant volume of particulates and sulfur dioxide to cause one latent cancer fatality (LCF). The probability of causing one injury from a truck accident from a single shipment ranges from 1.7×10^{-6} to 4.4×10^{-3}. The probability of causing one fatality from a truck accident ranges from 4.3×10^{-8} to 3.6×10^{-4}.

By summarizing estimated fatalities and injuries in Tables D.4.6 and D.4.10 for the Proposed Action, approximately 7 fatalities and 92 injuries were calculated for combined CH and RH shipments using 100 percent trucks. Approximately 3 fatalities and 28 injuries were calculated for combined CH and RH shipments in the Proposed Action for the maximum rail case. (See Tables D.4.7, D.4.9, and D.4.11.)

Similar results for the Alternative Action were calculated from Tables D.4.8 and D.4.10. Approximately 7 fatalities and 92 injuries were estimated for combined CH and RH shipments for the 100 percent truck case. Approximately 2 fatalities and 23 injuries were estimated for combined CH and RH shipments for the maximum rail case. (See Tables D.4.9 and D.4.11.)

D.4.3.2 **Results from Lifetime Risk Approach**

Table D.4.12 summarizes traffic statistics along the WIPP preferred routes. For each segment, a description is provided of endpoints, length, average daily truck volume, population density, annual truck vehicle-miles, estimated annual TRU shipments, TRU shipments as a percentage of total miles, and annual average accident injury and fatality statistics.

The route-specific truck injury and fatality rates are very low; they are usually lower than the corresponding rates from Cashwell et al. (1986), as shown in Table D.4.13. There are no segments with a recent history of relatively high injury or fatality rates which could indicate a high-hazard segment.

Estimated TRU shipment volumes as a percentage of total truck volumes are extremely small for most route segments. The highest TRU shipment volume percentage is 4 percent to 5 percent for US 285 in New Mexico between I-25, Eldorado and US 70, Roswell. Because future truck volumes will likely increase, percentages calculated are conservative upper bounds.

Average State and systemwide truck accident, injury, and fatality rates compare favorably with the corresponding rates from other quoted sources (see Table D.4.13). The calculated WIPP Route Highway System Weighted Average accident rate is 1.37×10^{-6}. This is less than the rate (1.70×10^{-6}) quoted by the NRC (1977) and slightly higher than the rate (1.16×10^{-6}) experienced by Chem-Nuclear Systems, Inc. for Type B nationwide shipments. The WIPP Highway System Weighted Average injury and fatality rates are also less than the corresponding rates quoted by Cashwell et al. (1986). Consequently, statistical analyses indicate that the preferred WIPP highway routes are safer than the U.S. highway system as a whole. The SEIS analysis of nonradiological and nonchemical risks based on Cashwell et al.

data is conservative.

Table D.4.14 and Figure D.4.1 compare lifetime risks for 1) Proposed Action--100 percent truck, 2) Proposed Action--maximum rail, 3) Alternative Action--100 percent truck, and 4) Alternative Action--maximum rail using the two methods discussed above to estimate nonradiological and nonchemical consequences.

Figure D.4.1 shows a range of forecasted estimates based on various statistics and indicates no clear difference between 100 percent truck and maximum rail modes.

D.4.3.3 **Comparison of Transuranic Waste Transport Accident, Injury, and Fatality Projections**

In the draft SEIS, impacts were assessed for waste transport by truck (34,144 shipments) and by maximum rail (18,467 shipments) for the proposed 25-year combined Test Phase and Disposal Phase at the WIPP. Based on revisions to the overall number of projected shipments required to transport waste to the WIPP, the final SEIS estimates a total number of truck shipments (28,866 shipments) and maximum rail shipments (15,558 shipments). For the truck shipment of TRU waste, the total estimated consequences for the projected 25-year Test and Disposal Phases in the draft SEIS was 8.3 fatalities and 106 injuries for the Proposed Action, as opposed to the revised final supplement which calculated 7 fatalities and 92 injuries, respectively.

The total estimated consequences for the maximum rail shipment mode for the Proposed Action in the draft supplement were 3 fatalities and 34 injuries. For this final supplement, the numbers have been revised to a projection of approximately 3 fatalities and 28 injuries.

It is important to restate that the total number of injuries and fatalities projected for truck transport in the draft SEIS were calculated based on Cashwell et al. data (1986). However, only in those projections, the projected injury rate per truck vehicle-mile ranged from 6.16×10^{-7} for urban areas to 1.33×10^{-6} for rural areas. This is in contrast to the actual values that were obtained from 23 States during the preparation of this final SEIS, which indicate an overall weighted average systemwide of 3.75×10^{-7}, which is significantly lower than the number that was projected in the EIS (see Table D.4.13).

Similar analyses of 100 percent truck mode fatality rates show that the Cashwell et al. (1986) numbers used in preparation of the SEIS ranged from 1.54×10^{-8} for urban areas to 1.9×10^{-7} for rural areas, as opposed to an overall preferred route highway system weighted average as presented based on State data of 1.98×10^{-8} fatalities per truck vehicle-mile of travel.

Table D.4.13 also compares the accident rates used in the draft SEIS (1.70×10^{-6} accidents per truck vehicle-mile) to the State data (overall average of 1.37×10^{-6} accidents per truck vehicle-mile) supplied for the final supplement. Probabilistic risks calculated using the higher rate (1.70×10^{-6}) from the NRC (NRC, 1977) are thus conservative given expected lower numbers of accidents based on actual route-specific data.

Table D.4.15 summarizes data on radioactive material shipments. The data was compiled from actual shipping records supplied by private sector radioactive waste transporters and the Department of Energy/Albuquerque Operations. As shown, the industry and the DOE have compiled an excellent safety record for shipping radioactive materials. The use of certified TRUPACT shipping containers and casks for TRU shipments and the extensive system of oversight and management developed for these shipments ensure that transportation risks for the Proposed Action or Alternative Action will be comparable, if not less, than those in similar shipping campaigns, as shown in Table D.4.15.

TABLE D.4.15 Comparison of radioactive material shipments

Source	Total mileage	Number of shipments	Accidents/ incidents	Injuries	Fatalities
SEIS					
Truck	74 million[a]	28,866	NR[b]	92	7
Rail	30 million	15,558	NR	25	3
Chem-Nuclear[c]					
Truck	26 million	NR	2	0	0
Spectra Research/SNL[d]					
Truck	NR	2,000,000[e]	828	NR	NR
Rail	NR		25	NR	NR
DOE/Albuquerque					
Truck	30.8		3	0	0[f]

[a] The total estimated mileage was not presented in the SEIS, the total estimated mileage represents a 25-year shipping campaign.

[b] NR = Not reported.

[c] Reporting period of 1987-1988.

[d] Reporting period of 1971-1988.

[e] The number of shipments were not broken down in truck and rail.

[f] Fatalities, but not attributable to project.

Figure D.4.1 Lifetime nonradiological and nonchemical transportion risks: ranges of projections for CH and RH shipments.

REFERENCES FOR APPENDIX D

AEC (Atomic Energy Commission), December, 1972. Environmental Survey of Transportation of Radioactive Materials, to and from Nuclear Power Plants, WASH-1238, Directorate of Regulatory Standards, Washington, D.C.

Alexander et al. (C. A. Alexander, J. S. Ogden, and L. Chan), 1986. "Actinide Release from Irradiated Fuel at High Temperature," IAEA-SM-281/10, Source Term Evaluations for Accident Conditions, from the Proceedings of a Symposium in Columbus, Ohio, for the International Atomic Energy Agency, Vienna.

Cashwell, J. W., et al., 1986. Transportation Impacts of the Commercial Radioactive Waste Management Program, SAND 85-2715, TTC-0663, Sandia National Laboratories, Albuquerque, New Mexico.

Chem-Nuclear Systems, Inc., 1989. Letter, TR-89-T055, dated May 25, 1989, from Leonard Toner to John Arthur, DOE-AL for WIPP SEIS, "Chem-Nuclear Systems Accident Analysis for Radioactive Material Shipments: 1987 and 1988."

DOE (U.S. Department of Energy), 1989. TRU Waste Acceptance Criteria for the Waste Isolation Pilot Plant, WIPP-DOE-069, Rev. 3, Carlsbad, New Mexico.

DOE (U.S. Department of Energy), 1986. Transportation Assessment and Guidance Report, DOE/JIO-002, Rev. 1, Albuquerque, New Mexico.

DOE (U.S. Department of Energy), 1980. Final Environmental Impact Statement Waste Isolation Pilot Plant, DOE-EIS-0026, Vols. 1 and 2, Carlsbad, New Mexico.

FRA (Federal Railroad Administration), 1988. Accident/Incident Bulletin No. 156: Calendar Year 1987, Office of Safety, U.S. Department of Transportation, Washington, D.C.

Huerta, M., et al., 1983. Analysis, Scale Modeling, and Full-Scale Test of Low-Level Nuclear Waste Drum Response to Accident Environments, SAND80-2517, Sandia National Laboratories, Albuquerque, New Mexico.

Joy, D. S., et al., 1982. Highway, a Transportation Routing Model: Program Description and User's Manual, ORNL/TM 84-19, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Lorenz et al. (R. A. Lorenz, J. L. Collins, A. P. Malinauskas, O. L. Kirkland, and R. L Towns), April 1980. Fission Product Release From Highly Irradiated LWR Fuel, NUREG/CR-0772, ORNL/NUREG/TM-287/R2, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Madsen et al. (M. M. Madsen, E. L. Wilmot, and J. M. Taylor), 1983. RADTRAN II User Guide, SAND82-2681, TTC-0399, Sandia National Laboratories, Albuquerque, New Mexico.

Mishima, J., and L. C. Schwendiman, 1973a. Some Experimental Measurements of Airborne Uranium (Representing Plutonium) in Transportation Accidents, BNWL-1732, Pacific Northwest Laboratories.

Mishima, J., and L. C. Schwendiman, 1973b. Fractional Airborne Release of Uranium (Representing Plutonium) During the Burning of Contaminated Wastes, BNWL-1730, Pacific Northwest Laboratories.

Fischer et al. (L. E. Fischer, C. K. Choy, M. A. Gerhard, C. Y. Kimura, R. W. Martin, R. W. Mensing, M. E. Mount, and M. C. Witte), 1987. Shipping Container Response to Severe Highway and Railway Accident Conditions, NUREG/CR-4829, Vols. 1 and 2, Lawrence Livermore National Laboratory, Livermore, California.

NRC (Nuclear Regulatory Commission), 1980. Accident Generated Particulate Materials and Their Characteristics - A Review of Background Information, NUREG/CR-2651, PNL-4154, Pacific Northwest Laboratory.

NRC (Nuclear Regulatory Commission), 1977. Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes, NUREG-0170, Vol. 1, Office of Standards Development, Washington, D.C.

ORNL (Oak Ridge National Laboratory), 1988. Integrated Data Base for 1988: Spent Fuel and Radioactive Waste Inventories, Projections, and Characteristics, DOE/RW-0006, Rev. 4, for U.S. DOE.

ORNL (Oak Ridge National Laboratory), 1987. Integrated Data Base for 1987: Spent Fuel and Radioactive Waste Inventories, Projections, and Characteristics, DOE/RW-0006, Rev. 3, for U.S. Department of Energy.

Peterson, B. E., 1984. INTERLINE: A Railroad Routing Model, ORNL/TM-8944, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Rand McNally, 1988. Rand McNally Handy Railroad Atlas of the United States, Rand McNally & Company, 1988.

Rand McNally, 1987. Rand McNally Road Atlas, Rand McNally & Company, 1987.

Rao et al. (R.K. Rao, E. L. Wilmot, and R. E. Luna), 1982. Nonradiological Impacts of Transporting Radioactive Material, SAND81-1703, TTC-0236, Sandia National Laboratories, Albuquerque, New Mexico.

Shirley, C. G., 1983. "A Simple Risk Analysis Tool for Estimating Release Fractions From Shipments of 55-Gallon Drums Involved in Transportation Accidents," Seventh International Symposium on Packaging and Transportation of Radioactive Materials (PATRAM 83), New Orleans, Louisiana, pp. 160-167.

Taylor, J. M. and S. L. Daniel, 1982. RADTRAN II: Revised Computer Code to Analyze Transportation of Radioactive Material, SAND80-1943, TTC-0239, Sandia National Laboratories, Albuquerque, New Mexico.

Taylor, J. M. and S. L. Daniel, 1977. RADTRAN: A Computer Code to Analyze Transportation of Radioactive Material, SAND76-0243, Sandia National Laboratories, Albuquerque, New Mexico.

Wilmot, E. L., M. M. Madsen, J. W. Cashwell, and D. S. Joy, 1983. A Preliminary Analysis of the Cost and Risk of Transporting Nuclear Waste to Potential Candidate Commercial Repository Sites, SAND83-0867, TTC-0434, Sandia National Laboratories, Albuquerque, New Mexico.

Wolff, T. A., 1984. The Transportation of Nuclear Materials, SAND84-0062, TTC-0471, Sandia National Laboratory, Albuquerque, New Mexico.

Wooden, D. G., 1986. Railroad Transportation of Spent Fuel, SAND86-7083, TTC-0574, Sandia National Laboratories, Albuquerque, New Mexico.

TABLE D.4.6 Total transportation risk for Proposed Action Alternative, CH truck mode

		Test Phase[a,c]				Disposal Phase[b,c]			
			Normal transportation	Accident case			Normal transportation	Accident case	
Facility	Zone	Number of shipments	LCFs	Fatalities	Injuries	Number of shipments	LCFs	Fatalities	Injuries
INEL	Rural	405	0	1.1×10^{-1}	1.4×10^{0}	3,641	0	1.0×10^{0}	1.3×10^{1}
	Suburban		0	4.6×10^{-3}	1.1×10^{-1}		0	4.1×10^{-2}	9.6×10^{-1}
	Urban		2.4×10^{-3}	2.3×10^{-4}	9.1×10^{-3}		2.1×10^{-2}	2.0×10^{-3}	8.2×10^{-2}
RFP	Rural	761	0	1.2×10^{-1}	1.5×10^{0}	6,847	0	1.1×10^{0}	1.3×10^{1}
	Suburban		0	5.6×10^{-3}	1.3×10^{-1}		0	5.1×10^{-2}	1.2×10^{0}
	Urban		4.3×10^{-3}	4.1×10^{-4}	1.6×10^{-2}		3.8×10^{-2}	3.7×10^{-3}	1.5×10^{-1}
HANF	Rural	310	0	1.1×10^{-1}	1.4×10^{0}	2,793	0	1.0×10^{0}	1.2×10^{1}
	Suburban		0	4.3×10^{-3}	1.0×10^{-1}		0	3.9×10^{-2}	9.0×10^{-1}
	Urban		1.7×10^{-3}	1.6×10^{-4}	6.6×10^{-3}		1.5×10^{-2}	1.5×10^{-3}	5.9×10^{-2}
SRS	Rural	264	0	6.8×10^{-2}	8.3×10^{-1}	2,376	0	6.1×10^{-1}	7.4×10^{0}
	Suburban		0	5.6×10^{-3}	1.3×10^{-1}		0	5.1×10^{-2}	1.2×10^{0}
	Urban		8.0×10^{-4}	7.7×10^{-5}	3.1×10^{-3}		7.2×10^{-3}	7.0×10^{-4}	2.8×10^{-2}
LANL	Rural	207	[d]	1.4×10^{-2}	1.7×10^{-1}	1,858	[d]	1.3×10^{-1}	1.5×10^{0}
	Suburban		[d]	3.8×10^{-4}	8.9×10^{-3}		[d]	3.4×10^{-3}	8.0×10^{-2}
	Urban		[d]	[d]	[d]		[d]	[d]	[d]
ORNL	Rural	23	0	5.3×10^{-3}	6.5×10^{-2}	205	0	4.7×10^{-2}	5.8×10^{-1}
	Suburban		0	3.5×10^{-4}	8.1×10^{-3}		0	3.1×10^{-3}	7.2×10^{-2}
	Urban		6.9×10^{-5}	6.7×10^{-6}	2.7×10^{-4}		6.2×10^{-4}	6.0×10^{-5}	2.4×10^{-3}

TABLE D.4.6 Concluded

		Test Phase[a,c]				Disposal Phase[b,c]			
		Number of	Normal transportation	Accident case		Number of	Normal transportation	Accident case	
Facility	Zone	shipments	LCFs	Fatalities	Injuries	shipments	LCFs	Fatalities	Injuries
NTS	Rural	8	0	1.9×10^{-3}	2.4×10^{-2}	72	0	1.7×10^{-2}	2.1×10^{-1}
	Suburban		0	6.2×10^{-5}	1.5×10^{-3}		0	5.6×10^{-4}	1.3×10^{-2}
	Urban		6.6×10^{-5}	6.3×10^{-6}	2.5×10^{-4}		5.9×10^{-4}	5.7×10^{-5}	2.3×10^{-3}
ANLE	Rural	1	0	2.4×10^{-4}	2.9×10^{-3}	13	0	3.1×10^{-3}	3.7×10^{-2}
	Suburban		0	1.6×10^{-5}	3.8×10^{-4}		0	2.1×10^{-4}	5.0×10^{-3}
	Urban		4.4×10^{-7}	4.3×10^{-8}	1.7×10^{-6}		5.8×10^{-6}	5.6×10^{-7}	2.2×10^{-5}
LLNL	Rural	97	0	2.7×10^{-2}	3.2×10^{-1}	872	0	2.4×10^{-1}	2.9×10^{0}
	Suburban		0	7.7×10^{-4}	1.8×10^{-2}		0	6.9×10^{-3}	1.6×10^{-1}
	Urban		1.7×10^{-3}	1.6×10^{-4}	6.5×10^{-3}		1.5×10^{-2}	1.4×10^{-3}	5.8×10^{-2}
Mound	Rural	15	0	3.6×10^{-3}	4.4×10^{-2}	135	0	3.3×10^{-2}	4.0×10^{-1}
	Suburban		0	2.9×10^{-4}	6.7×10^{-3}		0	2.6×10^{-3}	6.0×10^{-2}
	Urban		3.5×10^{-5}	3.4×10^{-6}	1.4×10^{-4}		3.2×10^{-4}	3.1×10^{-5}	1.2×10^{-3}
Total		2,091	1.1×10^{-2}	4.8×10^{-1}	6.3×10^{0}	18,812	9.9×10^{-2}	4.4×10^{0}	5.7×10^{1}

[a] The Test Phase assumes a 5-year time frame and 10 percent waste emplacement and shipment for the Test Phase.

[b] Operation assumes 20 years of rail shipment.

[c] Numbers are expressed in scientific notation $8.92 \times 10^{-7} = 0.000000892$.

[d] The preferred route from LANL to WIPP passes through no urban population zones.

TABLE D.4.7 Total transportation risk for Proposed Action Alternative, CH rail mode

		Test Phase[a,c]				Disposal Phase[b,c]			
			Normal transportation	Accident case			Normal transportation	Accident case	
Facility	Zone	Number of shipments	LCFs	Fatalities	Injuries	Number of shipments	LCFs	Fatalities	Injuries
INEL	Rural	405	0	1.1×10^{-1}	1.4×10^{0}	1,821	0	2.6×10^{-1}	2.7×10^{0}
	Suburban		0	4.6×10^{-3}	1.1×10^{-1}		0	2.9×10^{-2}	3.0×10^{-1}
	Urban		2.4×10^{-3}	2.3×10^{-4}	9.1×10^{-3}		9.4×10^{-3}	2.0×10^{-3}	2.1×10^{-2}
RFP	Rural	761	0	1.2×10^{-1}	1.5×10^{0}	3,429	0	3.0×10^{-1}	3.1×10^{0}
	Suburban		0	5.6×10^{-3}	1.3×10^{-1}		0	4.0×10^{-2}	4.2×10^{-1}
	Urban		4.3×10^{-3}	4.1×10^{-4}	1.6×10^{-2}		2.7×10^{-2}	5.8×10^{-3}	6.1×10^{-2}
HANF	Rural	310	0	1.1×10^{-1}	1.4×10^{0}	1,396	0	2.6×10^{-1}	2.7×10^{0}
	Suburban		0	4.3×10^{-3}	1.0×10^{-1}		0	3.4×10^{-2}	3.5×10^{-1}
	Urban		1.7×10^{-3}	1.6×10^{-4}	6.6×10^{-3}		9.4×10^{-3}	2.0×10^{-3}	2.1×10^{-2}
SRS	Rural	264	0	6.8×10^{-2}	8.3×10^{-1}	1,188	0	1.6×10^{-1}	1.7×10^{0}
	Suburban		0	5.6×10^{-3}	1.3×10^{-1}		0	4.6×10^{-2}	4.9×10^{-1}
	Urban		8.0×10^{-4}	7.7×10^{-5}	3.1×10^{-3}		1.5×10^{-2}	3.3×10^{-3}	3.5×10^{-2}
LANL[e]	Rural	207	[d]	1.4×10^{-2}	1.7×10^{-1}	1,858	[d]	1.3×10^{-1}	1.5×10^{0}
	Suburban		[d]	3.8×10^{-4}	8.9×10^{-3}		[d]	3.4×10^{-3}	8.0×10^{-2}
	Urban		[d]	[d]	[d]		[d]	[d]	[d]
ORNL	Rural	23	0	5.3×10^{-3}	6.5×10^{-2}	103	0	1.2×10^{-2}	1.3×10^{-1}
	Suburban		0	3.5×10^{-4}	8.1×10^{-3}		0	2.9×10^{-3}	3.0×10^{-2}
	Urban		6.9×10^{-5}	6.7×10^{-6}	2.7×10^{-4}		9.2×10^{-4}	2.0×10^{-4}	2.1×10^{-3}

TABLE D.4.7 Concluded

		Test Phase[a,c]				Disposal Phase[b,c]			
			Normal transportation	Accident case			Normal transportation	Accident case	
Site	Zone	Number of shipments	LCFs	Fatalities	Injuries	Number of Shipments	LCFs	Fatalities	Injuries
NTS[e]	Rural	8	0	1.9×10^{-3}	2.4×10^{-2}	72	0	1.7×10^{-2}	2.1×10^{-1}
	Suburban		0	6.2×10^{-5}	1.5×10^{-3}		0	5.6×10^{-4}	1.3×10^{-2}
	Urban		6.6×10^{-5}	6.3×10^{-6}	2.5×10^{-4}		5.9×10^{-4}	5.7×10^{-5}	2.3×10^{-3}
ANLE	Rural	1	0	2.4×10^{-4}	2.9×10^{-3}	7	0	7.6×10^{-4}	8.1×10^{-3}
	Suburban		0	1.6×10^{-5}	3.8×10^{-4}		0	1.6×10^{-4}	1.7×10^{-3}
	Urban		4.4×10^{-7}	4.3×10^{-8}	1.7×10^{-6}		6.0×10^{-5}	1.3×10^{-5}	1.4×10^{-4}
LLNL	Rural	97	0	2.7×10^{-2}	3.2×10^{-1}	436	0	6.3×10^{-2}	6.6×10^{-1}
	Suburban		0	7.7×10^{-4}	1.8×10^{-2}		0	1.1×10^{-2}	1.1×10^{-1}
	Urban		1.7×10^{-3}	1.6×10^{-4}	6.5×10^{-3}		2.7×10^{-3}	5.9×10^{-4}	6.2×10^{-3}
Mound	Rural	15	0	3.6×10^{-3}	4.4×10^{-2}	68	0	8.0×10^{-3}	8.4×10^{-2}
	Suburban		0	2.9×10^{-4}	6.7×10^{-3}		0	2.2×10^{-3}	2.3×10^{-2}
	Urban		3.5×10^{-5}	3.4×10^{-6}	1.4×10^{-4}		9.1×10^{-4}	2.0×10^{-4}	2.1×10^{-3}
Total		2,091	1.1×10^{-2}	4.8×10^{-1}	6.3×10^{0}	10,378	6.7×10^{-2}	1.4×10^{0}	1.5×10^{1}

[a] The Test Phase assumes a 5-year time frame and 10 percent waste emplacement shipment for the Test Phase.

[b] Disposal Phase assumes 20 years of rail shipment.

[c] Numbers are expressed in scientific notation $8.92 \times 10^{-7} = 0.000000892$.

[d] The preferred route from LANL to WIPP passes through no urban population zones.

[e] For the maximum rail case, shipments from LANL and NTS are made by truck.

TABLE D.4.12 Traffic Statistics: Truck Volume and Accidents by Segment

Route	State	Segment Description	Length Miles (L)	Truck Avg Daily Traffic (ADT)	Land use	Annual Truck Vehicle-Miles of travel (VMT) (L*ADT*365.25)	Annual TRU[a] Shipments	TRU as % of Total Truck-Miles	Annual Accidents No.	Rate/Truck VMT	Annual Injuries No.	Rate/Truck VMT	Annual Fatalities No.	Rate/Truck VMT
		Year: 1988												
I-25	NM	I-40 to San Mateo I/C, Albuquerque[b]	4.2	3902	U	5.98×10^{6}	1818[b]	0.128%[b]	23	3.84×10^{-6}	12	2.00×10^{-6}	0	0
I-25	NM	San Mateo I/C to Bernalillo/Sandoval County Line, Albuq.[b]	4.4	2039	U	3.28×10^{6}	1818[b]	0.244%[b]	1	3.10×10^{-7}	2	6.10 x 10-^{7}	0	0
I-25	NM	Bernalillo/Sandoval County Line to NM 44, Bernalillo[b]	7.3	1791	R	4.78×10^{6}	1818[b]	0.278%[b]	2	4.20×10^{-7}	2	4.20×10^{-7}	0	0
I-25	NM	NM 44, Bernalillo to US84/285, St. Francis Dr., Santa Fe[b]	40.4	1163	R	1.72×10^{7}	1818[b]	0.428%[b]	11	6.40×10^{-7}	9	5.20×10^{-7}	2	1.20×10^{-7}
I-25	NM	US84/285 (N), St. Francis Dr., Santa Fe to US285 (S), Eldorado	7.9	883	S	2.40×10^{6}	196	0.061%	1	4.20×10^{-7}	2	8.30×10^{-7}	0	0
I-25	NM	US285 (S), Eldorado to US84 (S), Romeroville	49.5	695	R	1.26×10^{7}	1622	0.640%	8	6.40×10^{-7}	7	5.60×10^{-7}	0	0
I-25	NM	US84 (S), Romeroville to US56, Springer	72.5	571	R	1.51×10^{7}	1622	0.779%	11	7.30×10^{-7}	5	3.30×10^{-7}	1	7.00×10^{-8}
I-25	NM	US56, Springer to US64(W)	34.6	568	R	7.18×10^{6}	1622	0.783%	3	4.20×10^{-7}	3	4.20×10^{-7}	0	0
I-25	NM	US64 (W) to Colorado Line	13.7	1022	R	5.11×10^{6}	1622	0.435%	2	3.90×10^{-7}	2	3.90×10^{-7}	0	0
I-40	NM	Arizona Line to US666, Gallup	20.8	4129	R	1.51×10^{6}	96	0.006%	21	6.70×10^{-7}	7	2.20×10^{-7}	0	0
I-40	NM	US666, Gallup to NM371, Thoreau	32.5	4632	R	5.50×10^{7}	96	0.006%	23	4.20×10^{-7}	12	2.20×10^{-7}	1	2.00×10^{-8}

I-40	NM	NM371, Thoreau to NM53, Grants	28.6	4025	R	4.20×10^{7}	96	0.006%	18	4.30×10^{-7}	11	2.60×10^{-7}	1	2.00×10^{-8}
I-40	NM	NM53, Grants to W. Central I/C, Albuquerque	67.7	3814	R	9.04×10^{10}	96	0.007%	58	6.20×10^{-7}	46	4.90×10^{-7}	2	2.00×10^{-8}
I-40	NM	W. Central I/C to Rio Grande Blvd I/C, Albuquerque	7.4	4066	S	1.10×10^{7}	96	0.006%	17	1.55×10^{-6}	20	1.82×10^{-6}	0	0
I-40	NM	Rio Grande Blvd I/C to I-25, Albuquerque	2.4	5510	U	4.83×10^{6}	96	0.005%	26	5.38×10^{-6}	9	1.86×10^{-6}	0	0
I-40	NM	I-25 to San Mateo Blvd I/C, Albuquerque	2.4	7590	U	6.65×10^{6}	96	0.003%	38	5.71×10^{-6}	17	2.56×10^{-6}	0	0
I-40	NM	San Mateo I/C to Tramway I/C, Albuquerque	5.7	4753	U	9.90×10^{6}	96	0.006%	27	2.73×10^{-6}	10	1.01×10^{-6}	0	0
I-40	NM	Tramway I/C, Albuquerque to US285, Clines Corners	50.6	4566	R	8.44×10^{7}	96	0.006%	48	5.70×10^{-7}	31	3.70×10^{-7}	1	1.00×10^{-8}
I-40	NM	US285, Clines Corners to US84 (N)	38.3	3433	R	4.80×10^{7}	528[b]	0.042%[b]	27	5.60×10^{-7}	17	3.50×10^{-7}	1	2.00×10^{-8}
I-40	NM	US84 (N) to US84 (S), Santa Rosa	20.4	3521	R	2.62×10^{7}	528[b]	0.041%[b]	13	5.00×10^{-7}	11	4.20×10^{-7}	0	0.00×10^{0}
I-40	NM	US84 (S), Santa Rosa to US54, Tucumcari (W)	52.4	4708	R	9.01×10^{7}	528	0.031%	22	2.40×10^{-7}	15	1.70×10^{-7}	1	1.00×10^{-8}
I-40	NM	US54, Tucumcari (W) to Texas Line	44.2	3587	R	5.79×10^{7}	528	0.040%	10	1.70×10^{-7}	4	7.00×10^{-8}	0	0
US285	NM	Texas Line to US180 (W), El Paso Rd, Carlsbad	31.5	203	R	2.34×10^{6}	238	0.320%	1	4.30×10^{-7}	0	0	0	0

TABLE D.4.12 Continued

Route	State	Segment Description	Length Miles (L)	Truck Avg Daily Traffic (ADT)	Land use	Annual Truck Vehicle-Miles of travel (VMT) (L*ADT*365.25)	Annual TRU[a] Shipments	TRU as % of Total Truck-Miles	Annual Accidents No.	Rate/Truck VMT	Annual Injuries No.	Rate/Truck VMT	Annual Fatalities No.	Rate/Truck VMT
US285	NM	US180 (W), El Paso Rd to US62-180 (E), Greene St., Carlsbad	2.0	613	S	4.48×10^{5}	238	0.106%	2	4.47×10^{-6}	3	6.70×10^{-6}	0	0
US285	NM	US62-180 (E), Greene St. to N Urban Limit, Carlsbad	3.5	379	S	4.84×10^{5}	2442	1.76%	1	2.07×10^{-6}	0	0	0	0
US285	NM	N Urban Limit, Carlsbad to S Urban Limit, Artesia	30.3	402	R	4.44×10^{6}	2442	1.66%	5	1.13×10^{-6}	4	9.00×10^{-7}	0	0
US285	NM	S Urban Limit, Artesia to US82, Artesia	1.8	381	S	2.50×10^{5}	2442	1.75%	0	0	0	0	0	0
US285	NM	US82, Artesia to N Urban Limit, Artesia	1.6	416	S	2.43×10^{5}	2442	1.61%	1	4.12×10^{-6}	0	0	0	0
US285	NM	N Urban Limit, Artesia to S Urban Limit, Roswell	34.5	349	R	4.40×10^{6}	2442	1.92%	0	0	0	0	0	0
US285	NM	S Urban Limit, Roswell to US70 (W), 2nd St., Roswell	4.2	658	S	1.01×10^{6}	2442	1.02%	3	2.97×10^{-6}	4	3.96×10^{-6}	0	0
US285	NM	US70 (W), 2nd St., to N Urban Limit, Roswell	3.6	971	S	1.28×10^{6}	2442	0.688%	5	3.92×10^{-6}	0	0	0	0
US285	NM	N Urban Limit, Roswell to US70 (E)	1.6	590	R	3.45×10^{5}	2442	1.13%	0	0	0	0	0	0
US285	NM	US70 (E) to US54 (E) Vaughn	89.6	157	R	5.41×10^{6}	2442	4.26%	4	7.80 x 10-[7]	0	0	0	0
US285/60	NM	US54 (E) Vaughn to US54 (W)	3.9	381	R	5.43×10^{5}	1914	1.38%	0	0	0	0	0	0

TABLE D.4.12 Continued

Route	State	Segment Description	Length Miles (L)	Truck Avg Daily Traffic (ADT)	Land use	Annual Truck Vehicle-Miles of travel (VMT) (L*ADT*365.25)	Annual TRU[a] Shipments	TRU as % of Total Truck-Miles	Annual Accidents No.	Rate/Truck VMT	Annual Injuries No.	Rate/Truck VMT	Annual Fatalities No.	Rate/Truck VMT

TABLE D.4.12 Continued

Route	State	Segment Description	Length Miles (L)	Truck Avg Daily Traffic (ADT)	Land use	Annual Truck Vehicle-Miles of travel (VMT) (L*ADT*365.25)	Annual TRU[a] Shipments	TRU as % of Total Truck-Miles	Annual Accidents No.	Rate/Truck VMT	Annual Injuries No.	Rate/Truck VMT	Annual Fatalities No.	Rate/Truck VMT
US54	NM	US285 (S), Vaughn to US60 (E)	0.6	244	R	4.90×10^{4}	534	0.595%	0	2.04×10^{-5}	0	0	0	0
US54	NM	US60 (E) to I-40, Santa Rosa	37.2	149	R	2.01×10^{6}	534	0.974%	2	9.90×10^{-7}	0	0	0	0
US60/285	NM	US54 (W) to US60 (W) Encino	14.2	223	R	1.16×10^{6}	1914	2.35%	1	8.60×10^{-7}	0	0	0	0
US285	NM	US60 (W), Encino to I-40, Clines Corners	27	152	R	1.49×10^{6}	1914	3.45%	3	2.01×10^{-6}	0	0	0	0
US285	NM	I-40, Clines Corners to I-25, Eldorado	41.3	99	R	1.49×10^{6}	1818	5.03%	3	2.01×10^{-6}	0	0	0	0
US84/285	NM	I-25/St. Francis Dr. I/C to N Urban Limit, Santa Fe	6.8	2275	S	5.65×10^{6}	196	0.024%	13	2.30×10^{-6}	6	1.06×10^{-6}	1	1.77×10^{-7}
US84/285	NM	N Urban Limit, Santa Fe to NM 502, Pojoaque	12.6	771	R	3.55×10^{6}	196	0.07%	2	5.60×10^{-7}	1	2.80×10^{-7}	0	0
NM502	NM	US84/285, Pojoaque to NM4, White Rock Wye	12.2	421	R	1.88×10^{6}	196	0.127%	3	1.60×10^{-6}	0	0	0	0
NM502	NM	NM4, White Rock Wye to E Urban Limit, Los Alamos	3.3	371	R	4.48×10^{5}	196	0.145%	1	2.23×10^{-6}	0	0	0	0
NM502	NM	E Urban Limit to Diamond Dr. LANL Entrance, Los Alamos	3	344	S	3.78×10^{5}	196	0.156%	1	2.65×10^{-6}	0	0	0	0
US62/180	NM	US285, Canal Rd/Greene St Intersection to E Urban Limit, Carlsbad	1.1	504	S	2.02×10^{5}	2680	1.46%	1	4.94×10^{-6}	1	4.94×10^{-6}	0	0

TABLE D.4.12 Continued

Route	State	Segment Description	Length Miles (L)	Truck Avg Daily Traffic (ADT)	Land use	Annual Truck Vehicle-Miles of travel (VMT) (L*ADT*365.25)	Annual TRU[a] Shipments	TRU as % of Total Truck-Miles	Annual Accidents No.	Rate/Truck VMT	Annual Injuries No.	Rate/Truck VMT	Annual Fatalities No.	Rate/Truck VMT
US62/180	NM	E Urban Limit, Carlsbad to WIPP N Entrance Rd	27.8	636	R	6.46×10^{06}	2680	1.16%	0	0	0	0	0	0

TABLE D.4.12 Continued

Route	State	Segment Description	Length Miles (L)	Truck Avg Daily Traffic (ADT)	Land use	Annual Truck Vehicle-Miles of travel (VMT) (L*ADT*365.25)	Annual TRU[a] Shipments	TRU as % of Total Truck-Miles	Annual Accidents No.	Rate/Truck VMT	Annual Injuries No.	Rate/Truck VMT	Annual Fatalities No.	Rate/Truck VMT
							Year: 1987							
I-25	CO	New Mexico Line to US160 (W), Walsenburg (W)	52.3	1276	R	2.44×10^{7}	1622	0.348%	32[c]	1.31×10^{-6}[c]	--	--	--	--
I-25	CO	US160 (W), Walsenburg (W) to Pueblo S. Urban Limit	40.7	1520	R	2.26×10^{7}	1622	0.292%	23[c]	1.02×10^{-6}[c]	--	--	--	--
I-25	CO	Pueblo S. Urban Limit to Pueblo N. Urban Limit, Pueblo	10.6	1688	U	6.54×10^{6}	1622	0.263%	20[c]	3.06×10^{-6}[c]	--	--	--	--
I-25	CO	Pueblo N. Urban Limit to Colorado Springs S Urban Limit	24.3	2506	R	2.22×10^{7}	1622	0.177%	20[c]	8.99×10^{-7}[c]	--	--	--	--
I-25	CO	Colorado Springs S Urban Limit to US24, Colorado Springs	13.3	2880	S	1.40×10^{7}	1622	0.154%	23[c]	1.64×10^{-6}[c]	--	--	--	--
I-25	CO	US24 Colorado Springs to N Urban Limit, Colorado Springs	16.1	3440	U	2.02×10^{7}	1622	0.129%	35[c]	1.73×10^{-6}[c]	--	--	--	--
I-25	CO	N Urban Limit Colorado Springs to S Urban Limit, Denver	36.6	3797	R	5.07×10^{7}	1622	0.117%	48[c]	9.46×10^{-7}[c]	--	--	--	--
I-25	CO	S Urban Limit to I-225 Denver	6.2	7933	U	1.80×10^{7}	1622	0.056%	27[c]	1.50×10^{-6}[c]	--	--	--	--
I-25	CO	I-225 to SH2, Colo. Blvd. Denver	4	6772	U	9.89×10^{6}	1622	0.066%	14[c]	1.42×10^{-6}[c]	--	--	--	--
I-25	CO	SH2, Colo. Blvd. to US6, Denver	5.2	4383	U	8.32×10^{6}	1622	0.101%	21[c]	2.52×10^{-6}[c]	--	--	--	--

TABLE D.4.12 Continued

Route	State	Segment Description	Length Miles (L)	Truck Avg Daily Traffic (ADT)	Land use	Annual Truck Vehicle-Miles of travel (VMT) (L*ADT*365.25)	Annual TRU[a] Shipments	TRU as % of Total Truck-Miles	Annual Accidents No.	Rate/Truck VMT	Annual Injuries No.	Rate/Truck VMT	Annual Fatalities No.	Rate/Truck VMT

TABLE D.4.12 Continued

Route	State	Segment Description	Length Miles (L)	Truck Avg Daily Traffic (ADT)	Land use	Annual Truck Vehicle-Miles of travel (VMT) (L*ADT*365.25)	Annual TRU[a] Shipments	TRU as % of Total Truck-Miles	Annual Accidents No.	Rate/Truck VMT	Annual Injuries No.	Rate/Truck VMT	Annual Fatalities No.	Rate/Truck VMT
I-25	CO	US6 TO I-70, Denver	4.5	8336	U	1.37×10^{7}	1622	0.053%	32[c]	2.52×10^{-6c}	--	--	--	--
I-25	CO	I-70 to US36, Boulder Turnpike, Denver	3.2	6183	U	7.23×10^{6}	1622	0.072%	14[c]	1.94×10^{-6c}	--	--	--	--
I-25	CO	US36 Boulder Turnpike to SH7	12.1	3676	U	1.62×10^{7}	938	0.070%	23[c]	1.42×10^{-6c}	--	--	--	--
I-25	CO	SH7 to US34, Loveland	28.2	3302	R	3.40×10^{7}	938	0.078%	28[c]	8.23×10^{-7c}	--	--	--	--
I-25	CO	US34, Loveland to N Urban Limit, Fort Collins	15.1	2914	S	1.61×10^{7}	938	0.088%	16[c]	9.96×10^{-7c}	--	--	--	--
I-25	CO	N Urban Limit, Fort Collins to Wyoming Line	26.5	1686	R	1.63×10^{7}	938	0.152%	12[c]	7.35×10^{-7c}	--	--	--	--
US36	CO	I-25 to Sheridan Blvd I/C Westminster	4.8	1400	U	2.45×10^{6}	684	0.134%	3[c]	1.22×10^{-6c}	--	--	--	--
US36	CO	Sheridan Blvd to SH121, Wadsworth Blvd, Broomfield	4.5	1368	U	2.25×10^{6}	684	0.137%	1[c]	4.45×10^{-7c}	--	--	--	--
SH121	CO	US36, Boulder Turnpike to SH128, W 120th Ave. Broomfield	0.2	964	S	7.04×10^{4}	684	0.194%	1[c]	4.09×10^{-6c}	--	--	--	--
SH128	CO	SH121, Wadsworth Blvd to Indiana St. (near Rocky Flats Plant Entrance)	3.8	310	S	4.30×10^{5}	684	0.604%	1[c]	2.32×10^{-6c}	--	--	--	--
I-80	WY	Uinta County	57	2960	R	6.16×10^{7}	938	0.086%	80	1.30×10^{-6}	36	5.84×10^{-7}	1	1.62×10^{-8}
I-80	WY	Sweetwater County	142	2830	R	1.47×10^{8}	938	0.090%	149	1.02×10^{-6}	86	5.86×10^{-7}	9	6.13×10^{-8}
I-80	WY	Carbon County	81.8	2667	R	7.97×10^{7}	938	0.096%	87	1.09×10^{-6}	26	3.26×10^{-7}	0	0

TABLE D.4.12 Continued

Route	State	Segment Description	Length Miles (L)	Truck Avg Daily Traffic (ADT)	Land use	Annual Truck Vehicle-Miles of travel (VMT) (L*ADT*365.25)	Annual TRU[a] Shipments	TRU as % of Total Truck-Miles	Annual Accidents No.	Rate/Truck VMT	Annual Injuries No.	Rate/Truck VMT	Annual Fatalities No.	Rate/Truck VMT

TABLE D.4.12 Continued

Route	State	Segment Description	Length Miles (L)	Truck Avg Daily Traffic (ADT)	Land use	Annual Truck Vehicle-Miles of travel (VMT) (L*ADT*365.25)	Annual TRU[a] Shipments	TRU as % of Total Truck-Miles	Annual Accidents No.	Rate/Truck VMT	Annual Injuries No.	Rate/Truck VMT	Annual Fatalities No.	Rate/Truck VMT
I-80	WY	Albany County	55.7	2427	R	4.94×10^{7}	938	0.106%	61	1.24×10^{-6}	34	6.88×10^{-7}	1	2.02×10^{-8}
I-80	WY	Albany/Laramie County Line to I-25 Cheyenne	23	1868	R	1.57×10^{7}	938	0.137%	49	3.12×10^{-6}	27	1.72×10^{-6}	2	1.27×10^{-7}
I-25	WY	I-80 Cheyenne to Colorado Line	8.8	1511	R	4.86×10^{6}	938	0.170%	9	1.85×10^{-6}	2	4.12×10^{-7}	0	0
I-80	UT	Wyoming Line to I-84 Echo	29.5	2780	R	3.00×10^{7}	938	0.092%	20	6.68×10^{-7}	--	--	--	--
I-84	UT	I-80 Echo to US80 Uintah	33.2	1250	R	1.52×10^{7}	938	0.205%	10	6.60×10^{-7}	--	--	--	--
I-84	UT	US89 Uintah to I-15 Ogden	7.1	1100	S	2.85×10^{6}	938	0.233%	6	2.10×10^{-6}	--	--	--	--
I-15/I-84	UT	I-84 Ogden to N Ogden	9	4000	S	1.31×10^{7}	938	0.064%	15	1.14×10^{-6}	--	--	--	--
I-15/I-84	UT	N Ogden to US91, Brigham City	12.5	2995	S	1.37×10^{7}	938	0.086%	9	6.58×10^{-7}	--	--	--	--
I-15/I-84	UT	US91 Brigham City to I-15 (Travel Way) Elwood	14.4	2170	R	1.14×10^{7}	938	0.118%	7	6.13×10^{-7}	--	--	--	--
I-15 (Travel Way)Plywood	UT	Elwood to Temp. End, 10	1045	R	3.82×10^{6}	412	0.108%	5	1.31×10^{-6}	--	--	--	--	
I-15	UT	Temp. End, Plymouth to Idaho Line	7	900	R	2.30×10^{6}	412	0.125%	1	4.34×10^{-7}	--	--	--	--
I-15/I-84	UT	Elwood to Future I-15 I/C Tremonton	3.4	1125	R	1.27×10^{6}	526	0.128%	2	1.57×10^{-6}	--	--	--	--
I-84	UT	Future I-15 I/C Tremonton to Idaho Line	41.8	1125	R	1.72×10^{7}	526	0.128%	20	1.16×10^{6}	--	--	--	--

TABLE D.4.12 Continued

Route	State	Segment Description	Length Miles (L)	Truck Avg Daily Traffic (ADT)	Land use	Annual Truck Vehicle-Miles of travel (VMT) (L*ADT*365.25)	Annual TRU[a] Shipments	TRU as % of Total Truck-Miles	Annual Accidents No.	Rate/Truck VMT	Annual Injuries No.	Rate/Truck VMT	Annual Fatalities No.	Rate/Truck VMT
		Year: 1988												
I-15	ID	Utah Line to US91 Virginia	36	900	R	1.18×10^{7}	412	0.125%	9	7.60×10^{-7}	6	5.07×10^{-7}	1	8.45×10^{-8}
I-15	ID	US91 Virginia to US30 McCammon	11	910	R	3.66×10^{6}	412	0.124%	6	1.64×10^{-6}	3	8.20×10^{-7}	0	0
I-15	ID	US30 McCammon to 5th Ave. Pocatello	20	1813	R	1.32×10^{7}	412	0.062%	2	1.51×10^{-7}	0	0	0	0
I-15	ID	5th Ave. to I-80 Pocatello	5	2165	S	3.95×10^{6}	412	0.052%	2	5.06×10^{-7}	1	2.53×10^{-7}	0	0
I-15	ID	I-86 Pocatello to US26 Blackfoot (Access to INEL)	20.5	2261	R	1.69×10^{7}	412	0.050%	6	3.54×10^{-7}	0	0	0	0
I-84	ID	Utah Line to I-86 I/C	53.6	1125	R	2.20×10^{7}	526	0.128%	28	1.27×10^{-6}	32	1.45×10^{-6}	3	1.36×10^{-7}
I-84	ID	I-86 I/C to US93 Twin Falls	49	2025	R	3.62×10^{7}	526	0.071%	31	8.55×10^{-7}	17	4.69×10^{-7}	2	5.52×10^{-8}
I-84	ID	US93 Twin Falls to US26 Bliss	32	1863	R	2.18×10^{7}	526	0.077%	7	3.21×10^{-7}	3	1.38×10^{-7}	1	4.59×10^{-8}
I-84	ID	US26 Bliss to US20 Mt. Home	46	1575	R	2.64×10^{7}	526	0.091%	8	3.02×10^{-7}	3	1.13×10^{-7}	0	0
I-84	ID	US20 Mt. Home to Broadway Ave. Boise	41	2542	R	3.81×10^{7}	526	0.056%	25	6.57×10^{-7}	16	4.20×10^{-7}	0	0
I-84	ID	Broadway, Boise to I-184 (w) Boise	5	3400	S	6.21×10^{6}	526	0.042%	2	3.22×10^{-7}	1	1.61×10^{-7}	0	0
I-84	ID	I-184 (W) Boise to Bus I-84 (E) Nampa	11	2800	S	1.12×10^{7}	526	0.051%	7	6.22×10^{-7}	5	4.44×10^{-7}	0	0
I-84	ID	Bus I-84 (E) Nampa to US20/26 (W) N Caldwell	12	2334	S	1.02×10^{7}	526	0.061%	14	1.37×10^{-6}	9	8.80×10^{-7}	1	9.78×10^{-8}

TABLE D.4.12 Continued

Route	State	Segment Description	Length Miles (L)	Truck Avg Daily Traffic (ADT)	Land use	Annual Truck Vehicle-Miles of travel (VMT) (L*ADT*365.25)	Annual TRU[a] Shipments	TRU as % of Total Truck-Miles	Annual Accidents No.	Rate/Truck VMT	Annual Injuries No.	Rate/Truck VMT	Annual Fatalities No.	Rate/Truck VMT

TABLE D.4.12 Continued

Route	State	Segment Description	Length Miles (L)	Truck Avg Daily Traffic (ADT)	Land use	Annual Truck Vehicle-Miles of travel (VMT) (L*ADT*365.25)	Annual TRU[a] Shipments	TRU as % of Total Truck-Miles	Annual Accidents No.	Rate/Truck VMT	Annual Injuries No.	Rate/Truck VMT	Annual Fatalities No.	Rate/Truck VMT
I-84	ID	US20/26 (W) N Caldwell to Oregon Line	26	2138	R	2.03×10^{7}	526	0.067%	12	5.91×10^{-7}	10	4.92×10^{-7}	0	0
I-84/I-82	OR	Idaho Line to Washington Line	213	2158	R	1.68×10^{8}	526	0.066%	48	2.86×10^{-7}	35	2.08×10^{-7}	0	0
I-82	WA	Oregon Line to US395	19.8	2224	R	1.61×10^{7}	526	0.064%	23	1.43×10^{-6}	14	8.70×10^{-7}	0	0
I-82	WA	US395 to I-184 W. Richland	11	1276	S	5.13×10^{6}	526	0.112%	7	1.36×10^{-6}	6	1.17×10^{-6}	0	0
I-184	WA	I-82 W. Richland to SR240 by-pass Hwy. Richland	5	1584	S	2.89×10^{6}	526	0.090%	4	1.38×10^{-6}	1	3.46×10^{-7}	0	0
SR240	WA	I-182 Richland to Horn Rd HANF	14.4	1680	U	8.84×10^{6}	526	0.085%	3	3.40×10^{-7}	2	2.26×10^{-7}	0	0
I-40	AZ	California Line to US93 (W) Kingman	48.9	2679	R	4.78×10^{7}	96	0.010%	17	3.55×10^{-7}	6	1.25×10^{-7}	0	0
I-40	AZ	US93 (W) Kingman to US93 (S) Round Valley	22.7	3667	R	3.04×10^{7}	96	0.007%	19	6.25×10^{-7}	15	4.93×10^{-7}	0	0
I-40	AZ	US93 (S) Round Valley to SR66 E. Seligman	51.4	2404	R	4.51×10^{7}	96	0.011%	41	9.08×10^{-7}	25	5.54×10^{-7}	0	0
I-40	AZ	SR66 E. Seligman to US89 (S) Ash Fork	22.9	3616	R	3.02×10^{7}	96	0.007%	14	4.63×10^{-7}	6	1.98×10^{-7}	0	0
I-40	AZ	US89 (S) Ash Fork to I-17 Flagstaff	49.2	4128	R	7.42×10^{7}	96	0.006%	80	1.08×10^{-6}	42	5.66×10^{-7}	0	0
I-40	AZ	I-17 Flagstaff to US180 (S) Holbrook	91.4	4250	R	1.42×10^{8}	96	0.006%	127	8.95×10^{-7}	63	4.44×10^{-7}	1	7.05×10^{-9}
I-40	AZ	US180 (S) Holbrook to												

TABLE D.4.12 Continued

Route	State	Segment Description	Length Miles (L)	Truck Avg Daily Traffic (ADT)	Land use	Annual Truck Vehicle-Miles of travel (VMT) (L*ADT*365.25)	Annual TRU[a] Shipments	TRU as % of Total Truck-Miles	Annual Accidents No.	Rate/Truck VMT	Annual Injuries No.	Rate/Truck VMT	Annual Fatalities No.	Rate/Truck VMT
		New Mexico Line	72.8	3545	R	9.43×10^{7}	96	0.007%	49	5.20×10^{-7}	20	2.12×10^{-7}	3	3.18×10^{-8}

TABLE D.4.12 Continued

Route	State	Segment Description	Length Miles (L)	Truck Avg Daily Traffic (ADT)	Land use	Annual Truck Vehicle-Miles of travel (VMT) (L*ADT*365.25)	Annual TRU[a] Shipments	TRU as % of Total Truck-Miles	Annual Accidents No.	Rate/Truck VMT	Annual Injuries No.	Rate/Truck VMT	Annual Fatalities No.	Rate/Truck VMT
						Year: 1987-88								
I-580	CA	SR84 Livermore (LLNL) Alameda County to San Joaquin County Line	10.7	10750	S	4.20×10^{7}	88	0.002%	69[d]	8.21×10^{-7}	39[d]	4.64×10^{-7}	2[d]	2.38×10^{-8}
I-580	CA	San Josquin County from Alameda County Line to 1-5 Vernalis	15.3	2900	R	1.62×10^{7}	88	0.008%	27[d]	8.33×10^{-7}	20[d]	6.17×10^{-7}	1[d]	3.08×10^{-8}
I-5	CA	From I-580 Vernalis through Stanislaus County	28.7	5280	R	5.53×10^{7}	88	0.004%	76[d]	6.86×10^{-7}	39[d]	3.52×10^{-7}	2[d]	1.81×10^{-8}
I-5	CA	Merced County from Stanislaus County Line to Fresno County Line	32.5	5800	R	6.88×10^{7}	88	0.004%	110[d]	7.99×10^{-7}	67[d]	4.86×10^{-7}	2[d]	1.45×10^{-8}
I-5	CA	Fresno County from Merced County Line to Kings County Line	66.2	6500	R	1.57×10^{8}	88	0.004%	140[d]	4.45×10^{-7}	109[d]	3.47×10^{-7}	7[d]	2.23×10^{-8}
I-5	CA	Kings County from Fresno County Line to Kern County Line	26.7	6800	R	6.63×10^{7}	88	0.004%	48[d]	3.62×10^{-7}	65[d]	4.90×10^{-7}	0[d]	0
I-5	CA	Kern County from Kings County Line to Los Angeles County Line	87	8521	R	2.71×10^{8}	88	0.003%	289[d]	5.34×10^{-7}	211[d]	3.90×10^{-7}	20[d]	3.69×10^{-8}
I-5	CA	Los Angeles County from Kern County Line to I-210 Foothill Freeway Los Angeles	44.6	18834	U	3.07×10^{8}	88	0.001%	303[d]	4.94×10^{-7}	203[d]	3.31×10^{-7}	12[d]	1.96×10^{-8}

TABLE D.4.12 Continued

Route	State	Segment Description	Length Miles (L)	Truck Avg Daily Traffic (ADT)	Land use	Annual Truck Vehicle-Miles of travel (VMT) (L*ADT*365.25)	Annual TRU[a] Shipments	TRU as % of Total Truck-Miles	Annual Accidents No.	Rate/Truck VMT	Annual Injuries No.	Rate/Truck VMT	Annual Fatalities No.	Rate/Truck VMT
I-210	CA	I-5 Los Angeles to I-10, San Bernadino Freeway, Pomona	48.5	8318	U	1.47×10^{8}	88	0.003%	523[d]	1.77×10^{-6}	250[d]	8.48×10^{-7}	9[d]	3.05×10^{-8}
I-10	CA	I-210 Foothill Freeway Pomona to San Bernadino County Line	5.8	15800	U	3.35×10^{7}	88	0.002%	232[d]	3.46×10^{-6}	113[d]	1.69×10^{-6}	0[d]	0
I-10	CA	San Bernadino County Line to I-15 Devore Freeway Ontario	9.9	16644	U	6.02×10^{7}	88	0.001%	246[d]	2.04×10^{-6}	127[d]	1.06×10^{-6}	4[d]	3.32×10^{-8}
I-15	CA	I-10 Ontario to I-40 Barstow	72	9324	S	2.45×10^{8}	88	0.003%	321[d]	6.54×10^{-7}	201[d]	4.10×10^{-7}	13[d]	2.65×10^{-8}
I-40	CA	I-15 Barstow to US95(N) and to Arizona Line	154.6	4500	R	2.54×10^{8}	88 96	0.005% 0.006%	152[d]	2.99×10^{-7}	103[d]	2.03×10^{-7}	3[d]	5.90×10^{-9}
US95	CA	I-40 to Nevada Line	23.2	348	R	2.95×10^{6}	8	0.006%	6[d]	1.02×10^{-6}	6[d]	1.02×10^{-6}	1[d]	1.69×10^{-7}
							Year: 1987							
US95	NV	California Line to SR164 Searchlight	20.4	410	R	3.05×10^{6}	8	0.005%	59[e]	6.45×10^{-6}	25[e]	2.73×10^{-6}	1[e]	1.09×10^{-7}
US95	NV	SR164 Searchlight to US93 Alunite	35.9	432	R	5.62×10^{6}	8	0.005%	68[e]	4.04×10^{-6}	25[e]	1.48×10^{-6}	0[e]	0
US95	NV	US93 Alunite to SR147 Henderson	9.4	1379	S	4.73×10^{6}	8	0.002%	21[e]	1.48×10^{-6}	11[e]	7.75×10^{-7}	0[e]	0
US93/95	NV	SR147 Henderson to Boulder Hwy E. Las Vegas	8.8	1940	U	6.42×10^{6}	8	0.001%	[f]	[f]	[f]	[f]	[f]	[f]
US93/95	NV	Boulder Hwy E. Las Vegas												

TABLE D.4.12 Continued

Route	State	Segment Description	Length Miles (L)	Truck Avg Daily Traffic (ADT)	Land use	Annual Truck Vehicle-Miles of travel (VMT) (L*ADT*365.25)	Annual TRU[a] Shipments	TRU as % of Total Truck-Miles	Annual Accidents No.	Rate/Truck VMT	Annual Injuries No.	Rate/Truck VMT	Annual Fatalities No.	Rate/Truck VMT
		to I-15 Las Vegas	5.6	3588	U	7.34×10^{6}	8	0.006%	13[e]	5.90×10^{-7}	1[e]	4.54×10^{-8}	0[e]	0
US95	NV	I-15 to Rainbow Blvd Las Vegas	5.1	2230	U	4.15×10^{6}	8	0.001%	40[e]	3.21×10^{-6}	6[e]	4.82×10^{-7}	0[e]	0

TABLE D.4.12 Continued

Route	State	Segment Description	Length Miles (L)	Truck Avg Daily Traffic (ADT)	Land use	Annual Truck Vehicle-Miles of travel (VMT) (L*ADT*365.25)	Annual TRU[a] Shipments	TRU as % of Total Truck-Miles	Annual Accidents No.	Rate/Truck VMT	Annual Injuries No.	Rate/Truck VMT	Annual Fatalities No.	Rate/Truck VMT
US95	NV	Rainbow Blvd to Rancho Road Las Vegas	5.9	728	S	1.57×10^{6}	8	0.003%	6[e]	1.27×10^{-6}	2[e]	4.25×10^{-7}	0[e]	0
US95	NV	Rancho Rd. Las Vegas to Indian Springs	33.2	737	R	8.94×10^{6}	8	0.003%	7[e]	2.61×10^{-7}	6[e]	2.24×10^{-7}	0[e]	0
US95	NV	Indian Springs to Mercury I/C, NTS	18.3	374	R	2.50×10^{6}	8	0.006%	4[e]	5.33×10^{-7}	0[e]	0	0[e]	0
							Year: 1988							
US285	TX	New Mexico Line to I-20, Pecos	53.4	372	R	2.53×10^{7}	238	0.175%	7[c]	5.16×10^{-7}[c]	Unknown	Unknown	0[c]	0[c]
I-20	TX	US285, Pecos to US87, Big Spring	136	4191	R	2.08×10^{8}	238	0.016%	119[c]	5.27×10^{-7}[c]	Unknown	Unknown	3[c]	1.44×10^{-8}[c]
I-20	TX	US87, Big Spring to US84, Roscoe	63	2409	R	5.54×10^{7}	238	0.027%	70[c]	1.26×10^{-6}[c]	Unknown	Unknown	7[c]	1.26×10^{-7}[c]
I-20	TX	US84, Roscoe to US183, Cisco	90	4093	R	1.34×10^{8}	238	0.016%	98[c]	7.28×10^{-7}[c]	Unknown	Unknown	1[c]	7.43×10^{-9}[c]
I-20	TX	US183, Cisco to I-30, Ft. Worth (W)	81	3345	R	9.90×10^{7}	238	0.019%	99[c]	1.00×10^{-6}[c]	Unknown	Unknown	4[c]	4.04×10^{-8}[c]
I-20	TX	I-30, Ft. Worth (W) to US287 (S), Ft. Worth Area	26	3956	U	3.76×10^{7}	238	0.016%	51[c]	1.36×10^{-6}[c]	Unknown	Unknown	0[c]	0[c]
I-20	TX	US287 (S), To US80 (E), Dallas Area	55.6	6755	U	1.37×10^{8}	238	0.010%	48[c]	3.50×10^{-7}[c]	Unknown	Unknown	1[c]	7.29×10^{-9}[c]
I-20	TX	US80 (E), Terrell to Louisiana Line	144	5263	R	2.77×10^{8}	238	0.012%	189[c]	6.83×10^{-7}[c]	Unknown	Unknown	4[c]	1.44×10^{-8}[c]

TABLE D.4.12 Continued

Route	State	Segment Description	Length Miles (L)	Truck Avg Daily Traffic (ADT)	Land use	Annual Truck Vehicle-Miles of travel (VMT) (L*ADT*365.25)	Annual TRU[a] Shipments	TRU as % of Total Truck-Miles	Annual Accidents No.	Rate/Truck VMT	Annual Injuries No.	Rate/Truck VMT	Annual Fatalities No.	Rate/Truck VMT
I-40	TX	New Mexico Line to Bus. Loop: I-40 (W), W. Amarillo	63	3910	R	9.00×10^{7}	528	0.037%	40[c]	4.44×10^{-7}[c]	Unknown	Unknown	1[c]	1.11×10^{-8}[c]

TABLE D.4.12 Continued

Route	State	Segment Description	Length Miles (L)	Truck Avg Daily Traffic (ADT)	Land use	Annual Truck Vehicle-Miles of travel (VMT) (L*ADT*365.25)	Annual TRU[a] Shipments	TRU as % of Total Truck-Miles	Annual Accidents No.	Rate/Truck VMT	Annual Injuries No.	Rate/Truck VMT	Annual Fatalities No.	Rate/Truck VMT
I-40	TX	Bus. Loop I-40 (W) to Bus Loop I-40 (E), Armarillo	23	7867	U	6.61×10^{7}	528	0.018%	40[c]	6.05×10^{-7}[c]	Unknown	Unknown	1[c]	1.51×10^{-8}[c]
							Year: 1987							
I-40	TX	Bus. Loop I-40 (E) Amarillo to Oklahoma Line	89	3626	R	1.18×10^{8}	528	0.040%	69[c]	5.85×10^{-7}[c]	Unknown	Unknown	2[c]	1.67×10^{-8}[c]
I-40	OK	Arkansas Line to E Oklahoma City Urban Area, Pottawatomie/ Oklahoma County Line	157.8	2578	R	1.49×10^{8}	482	0.051%	69	4.63×10^{-7}	31	2.08×10^{-7}	4	2.68×10^{-8}
I-40	OK	Pottawatomie County Line to I-44, Oklahoma City	26.9	3218	U	3.16×10^{7}	482	0.041%	95	3.00×10^{-6}	46	1.46×10^{-6}	1	3.16×10^{-8}
I-40	OK	I-44, Oklahoma City to W Oklahoma City Urban Area Oklahoma/Canadian County Line	5.6	4917	U	1.00×10^{7}	528	0.029%	23	2.30×10^{-6}	1	1.00×10^{-7}	0	0
I-40	OK	Oklahoma/Canadian County Line to Texas Line	139.8	3991	R	1.97×10^{8}	528	0.036%	104	5.28×10^{-7}	47	2.38×10^{-7}	2	1.02×10^{-8}
I-44	OK	I-40, Oklahoma City to Oklahoma/Lincoln County Line	26.5	3779	U	3.68×10^{7}	46	0.003%	76	2.06×10^{-6}	18	4.89×10^{-7}	3	8.15×10^{-8}
I-44	OK	Oklahoma/Lincoln County Line to Tulsa County Line	75	2910	R	7.97×10^{7}	46	0.004%	43	5.40×10^{-7}	16	2.01×10^{-7}	0	0
I-44	OK	Tulsa Urban Area, Tulsa County	19.5	4749	U	3.38×10^{7}	46	0.003%	70	2.07×10^{-6}	17	5.03×10^{-7}	1	2.96×10^{-8}
I-44	OK	Tulsa/Rogers County Line to Missouri Line	88	3143	R	1.10×10^{8}	46	0.004%	71	6.45×10^{-7}	40	3.64×10^{-7}	4	3.64×10^{-8}
I-44	MO	Oklahoma Line to Springfield												

TABLE D.4.12 Continued

Route	State	Segment Description	Length Miles (L)	Truck Avg Daily Traffic (ADT)	Land use	Annual Truck Vehicle-Miles of travel (VMT) (L*ADT*365.25)	Annual TRU[a] Shipments	TRU as % of Total Truck-Miles	Annual Accidents No.	Rate/Truck VMT	Annual Injuries No.	Rate/Truck VMT	Annual Fatalities No.	Rate/Truck VMT
		Urban Area, Greene County	59.9	1207	R	2.64×10^{7}	46	0.010%	--	1.13×10^{-6}[g]	--	--	--	--

TABLE D.4.12 Continued

Route	State	Segment Description	Length Miles (L)	Truck Avg Daily Traffic (ADT)	Land use	Annual Truck Vehicle-Miles of travel (VMT) (L*ADT*365.25)	Annual TRU[a] Shipments	TRU as % of Total Truck-Miles	Annual Accidents No.	Rate/Truck VMT	Annual Injuries No.	Rate/Truck VMT	Annual Fatalities No.	Rate/Truck VMT
I-44	MO	Greene County, Springfield Urban Area	31.1	1492	S	1.69×10^7	46	0.008%	--	1.29×10^{-6}[g]	--	--	--	--
I-44	MO	Greene County Line to St. Louis County Line	167.3	1208	R	7.38×10^7	46	0.010%	--	3.14×10^{-6}[g]	--	--	--	--
I-44/I-270/I-255	MO	St. Louis County, St. Louis Urban Area, to Illinois Line	27.6	1066	U	1.07×10^7	46	0.012%	--	2.82×10^{-6}[g]	--	--	--	--
I-255/I-55	IL	Missouri Line to I-70, E. St. Louis	30.3	1550	U	1.66×10^7	46	0.0084%	N/A	--	--	--	--	--
I-55	IL	I-70, E. St. Louis to Cass Ave., Chicago (ANLE)	258.6	3514	U,S,R	3.32×10^8	32	0.0025%	N/A	--	--	--	--	--
I-70	IL	I-55, E. St. Louis to Indiana Line	140	4686	R	2.40×10^8	14	0.0008%	N/A	--	--	--	--	--
I-70	IN	Illinois Line to I-465 Indianapolis	71	6035	R	1.56×10^8	14	0.0006%	N/A	--	--	--	--	--
I-465	IN	I-70 (W) to I-70 (E), Indianapolis	19	9586	U	6.65×10^7	14	0.0004%	N/A	--	--	--	--	--
I-70	IN	I-465, Indianapolis to Ohio Line	67	7338	R	1.80×10^8	14	0.0005%	N/A	--	--	--	--	--
							Year: 1986							
SR725	OH	First St., (MOUND Plant Vicinity) to I-75, Miamisburg	3	658	U	7.21×10^5	14	0.006%	15[h]	7.56×10^{-6}	10[h]	5.04×10^{-6}	0[h]	0
I-75	OH	SR725, Miamisburg Pike, Miamisburg to I-70, Dayton	16.3	11200	U	6.67×10^7	14	0.0003%	576[h]	3.14×10^{-6}	219[h]	1.19×10^{-6}	1[h]	5.45×10^{-9}

TABLE D.4.12 Continued

Route	State	Segment Description	Length Miles (L)	Truck Avg Daily Traffic (ADT)	Land use	Annual Truck Vehicle-Miles of travel (VMT) (L*ADT*365.25)	Annual TRU[a] Shipments	TRU as % of Total Truck-Miles	Annual Accidents No.	Rate/Truck VMT	Annual Injuries No.	Rate/Truck VMT	Annual Fatalities No.	Rate/Truck VMT

TABLE D.4.12 Continued

Route	State	Segment Description	Length Miles (L)	Truck Avg Daily Traffic (ADT)	Land use	Annual Truck Vehicle-Miles of travel (VMT) (L*ADT*365.25)	Annual TRU[a] Shipments	TRU as % of Total Truck-Miles	Annual Accidents No.	Rate/Truck VMT	Annual Injuries No.	Rate/Truck VMT	Annual Fatalities No.	Rate/Truck VMT
I-70	OH	I-75, Dayton to Preble/ Montgomery County Line	16.2	8000	R	4.73×10^{7}	14	0.0005%	199[h]	1.53×10^{-6}	79[h]	6.07×10^{-7}	3[h]	2.30×10^{-8}
I-70	OH	Preble County from Montgomery County Line to Indiana Line	17.7	7990	R	5.16×10^{7}	14	0.0005%	129[h]	9.08×10^{-7}	42[h]	3.00×10^{-7}	1[h]	7.04×10^{-9}
						Year: 1987								
I-40	AR	Texas Line to SR9, Russellville	81	3850	R	1.14×10^{8}	482	0.034%	49	4.30×10^{-7}	16	1.40×10^{-7}	2	1.76×10^{-8}
I-40	AR	SR9, Russellville to US65, Conway	44	4917	R	7.90×10^{7}	482	0.027%	27	3.42×10^{-7}	9	1.14×10^{-7}	2	2.53×10^{-8}
I-40	AR	US65, Conway to I-430, Little Rock	23.4	6000	S	5.13×10^{7}	482	0.022%	49	9.56×10^{-7}	16	3.12×10^{-7}	0	0
I-40	AR	I-430 to I-440, Little Rock	11.4	5460	U	2.27×10^{7}	482	0.024%	69	3.04×10^{-6}	23	1.01×10^{-6}	1	4.40×10^{-8}
I-40	AR	I440, Little Rock to I-55 (N), W. Memphis	118.4	7200	R	3.11×10^{8}	482	0.018%	124	3.98×10^{-7}	41	1.32×10^{-7}	7	2.25×10^{-8}
I-40	AR	W. Memphis to Tennessee Line	7.4	4918	S	1.33×10^{7}	482	0.027%	37	2.78×10^{-6}	12	9.03×10^{-7}	6	4.51×10^{-7}
I-40	TN	Arkansas Line to I-240 (N), Memphis	2.8	5300	U	5.42×10^{6}	482	0.025%	--	3.24×10^{-6c}	--	1.54×10^{-6c}	--	0[c]
I-40/ I-240 (N)	TN	I-40 (W) to I-40 (E), Memphis	11.6	6983	U	2.96×10^{7}	482	0.019%	--	6.02×10^{-7c}	--	5.17×10^{-7c}	--	3.72×10^{-9c}
I-40	TN	I-240 (N), Memphis to SR15/64, E. Memphis	7.1	5340	S	1.38×10^{7}	482	0.025%	--	1.57×10^{-6c}	--	5.73×10^{-7c}	--	9.39×10^{-9c}

TABLE D.4.12 Continued

Route	State	Segment Description	Length Miles (L)	Truck Avg Daily Traffic (ADT)	Land use	Annual Truck Vehicle-Miles of travel (VMT) (L*ADT*365.25)	Annual TRU[a] Shipments	TRU as % of Total Truck-Miles	Annual Accidents No.	Rate/Truck VMT	Annual Injuries No.	Rate/Truck VMT	Annual Fatalities No.	Rate/Truck VMT

TABLE D.4.12 Continued

Route	State	Segment Description	Length Miles (L)	Truck Avg Daily Traffic (ADT)	Land use	Annual Truck Vehicle-Miles of travel (VMT) (L*ADT*365.25)	Annual TRU[a] Shipments	TRU as % of Total Truck-Miles	Annual Accidents No.	Rate/Truck VMT	Annual Injuries No.	Rate/Truck VMT	Annual Fatalities No.	Rate/Truck VMT
I-40	TN	SR15/64, E. Memphis to US BP 45, Jackson	61.5	5580	R	1.25×10^{8}	482	0.024%	--	7.16×10^{-7c}	--	3.92×10^{-7c}	--	9.18×10^{-9c}
I-40	TN	US BP 45, Jackson to Davidson County Line, W. Nashville	110.5	6100	R	2.46×10^{8}	482	0.022%	--	5.96×10^{-7c}	--	3.23×10^{-7c}	--	1.83×10^{-8c}
I-40	TN	Davidson County, Nashville Urban Area from W to E Nashville	31.2	8400	U	9.57×10^{7}	482	0.016%	--	1.78×10^{-6c}	--	6.63×10^{-7c}	--	1.13×10^{-8c}
I-40	TN	E. Nashville to SR111, Cookeville	66.1	5250	R	1.27×10^{8}	482	0.025%	--	6.17×10^{7c}	--	$3.37 \times 10^{7*}$	--	$2.17 \times 10^{8*}$
I-40	TN	SR111, Cookeville to SR95 ORNL Vicinity	73	5350	R	1.43×10^{8}	482	0.025%	--	5.20×10^{7c}	--	$3.31 \times 10^{7*}$	--	$6.73 \times 10^{9*}$
I-20	LA	Texas Line to SR526, Shreveport	8.6	9150	R	2.87×10^{7}	238	0.007%	14	4.87×10^{-7}	4	1.39×10^{-7}	0	0
I-20	LA	SR526 to I-220 (E), Shreveport	17.7	17653	U	1.14×10^{8}	238	0.004%	240	2.10×10^{-6}	112	9.81×10^{-7}	2	1.75×10^{-8}
I-20	LA	I-220 (E) to SR34, W Monroe	89	8250	R	2.68×10^{8}	238	0.008%	155	5.78×10^{-7}	119	4.44×10^{-7}	5	1.86×10^{-8}
I-20	LA	SR34 to SR594 (E), Monroe Urban Area	8.8	13030	S	4.19×10^{7}	238	0.005%	53	1.26×10^{-6}	38	9.07×10^{-7}	0	0
I-20	LA	SR594 (E), Monroe to Mississippi Line	64.4	6630	R	1.56×10^{8}	238	0.010%	43	2.76×10^{7}	38	2.44×10^{-7}	1	6.41×10^{-9}
I-20	MS	Louisiana Line to Jackson East Urban Limit, Hinds/ Rankin County Line	45.9	5340	R	8.95×10^{7}	238	0.012%	21	2.34×10^{-7}	7	7.82×10^{-8}	1	1.12×10^{-8}

TABLE D.4.12 Continued

Route	State	Segment Description	Length Miles (L)	Truck Avg Daily Traffic (ADT)	Land use	Annual Truck Vehicle-Miles of travel (VMT) (L*ADT*365.25)	Annual TRU[a] Shipments	TRU as % of Total Truck-Miles	Annual Accidents No.	Rate/Truck VMT	Annual Injuries No.	Rate/Truck VMT	Annual Fatalities No.	Rate/Truck VMT
I-20	MS	Hinds/Rankin County Line to Alabama Line	110.2	4568	R	1.84×10^{8}	238 Year: 1988	0.014%	3	1.63×10^{-8}	1	5.44×10^{-9}	0	0

TABLE D.4.12 Continued

Route	State	Segment Description	Length Miles (L)	Truck Avg Daily Traffic (ADT)	Land use	Annual Truck Vehicle-Miles of travel (VMT) (L*ADT*365.25)	Annual TRU[a] Shipments	TRU as % of Total Truck-Miles	Annual Accidents No.	Rate/Truck VMT	Annual Injuries No.	Rate/Truck VMT	Annual Fatalities No.	Rate/Truck VMT
I-20	AL	Mississippi Line to I-459 Birmingham Urban Area	106.3	6025	R	2.34×10^{8}	238	0.011%	118	5.04×10^{-7}	51	2.18×10^{-7}	2	8.55×10^{-9}
I-459	AL	I-20 (W) TO I-20 (E) Birmingham Urban Area	33.5	3000	S	3.67×10^{7}	238	0.022%	17	4.63×10^{-7}	3	8.17×10^{-8}	0	0
I-20	AL	I-459, E. Birmingham to Georgia Line	78.3	7800	R	2.23×10^{8}	238	0.008%	102	4.57×10^{-7}	42	1.88×10^{-7}	6	2.69×10^{-8}
I-20	GA	Alabama Line to Atlanta W. Urban Limit	30.3	4420[i]	R	4.89×10^{7}	238	0.015%[i]	32[i]	6.54×10^{-7}[i]	22[i]	4.50×10^{-7}[i]	0[i]	0[i]
I-20	GA	Atlanta W. Urban Limit to I-285 (W), Atlanta	20.7	6750[i]	S	5.10×10^{7}	238	0.10%[i]	136[i]	2.66×10^{-6}[i]	57[i]	1.12×10^{-6}[i]	4[i]	7.84×10^{-8}[i]
I-285	GA	I-20 (W) to I-20 (E), Atlanta	26.1	9100[i]	U	8.68×10^{7}	238	0.007%[i]	164[i]	1.89×10^{-6}[i]	75[i]	8.64×10^{-7}[i]	4[i]	4.61×10^{-8}[i]
I-20	GA	I-285 (E), Atlanta to SR138, Conyers	14.9	5110[i]	S	2.78×10^{7}	238	0.013%[i]	63[i]	2.26×10^{-6}[i]	12[i]	4.32×10^{-7}[i]	1[i]	3.60×10^{-8}[i]
I-20	GA	SR138, Conyers to Lewiston, Augusta W. Urban Limit	108.5	2890[i]	R	1.14×10^{8}	238	0.022%[i]	50[i]	4.36×10^{-7}[i]	27[i]	2.36×10^{-7}[i]	1[i]	8.73×10^{-9}[i]
I-20	GA	Lewiston to South Carolina Line, Augusta Urban Area	11.7	2400[i]	S	1.02×10^{7}	238	0.027%[i]	9[i]	8.78×10^{-7}[i]	4[i]	3.90×10^{-7}[i]	0[i]	0
I-20	SC	Georgia Line to US25, N. Augusta	5	5800[j]	S	1.06×10^{7}	238	0.011%	--	6.04×10^{-7}[c]	--	1.13×10^{-7}[c]	--	0[c]
US25	SC	I-20 to SR125, N. Augusta	5.6	1150[j]	U	2.35×10^{6}	238	0.057%	--	4.27×10^{-6}[c]	--	1.38×10^{-6}[c]	--	0[c]
SR125	SC	US25, N. Augusta to SRS Entrance	16.8	635[j]	U	3.90×10^{6}	238	0.103%	--	1.71×10^{-6}[c]	--	4.88×10^{-7}[c]	--	0[c]

TABLE D.4.12 Concluded

Notes:

[a] Average annual truck shipments of both CH TRU TRUPACTs and RH TRU NuPac 72B casks during 20-yr Disposal Phase of the Proposed Action, going both to and from WIPP.
[b] Alternate route to preferred route.
[c] Based on the assumption on that truck equals the overall motor vehicle accident and fatality rate.
[d] 2-yr total; accident, injury, and fatality rates are 1-yr averages.
[e] 3-yr total; however, the resultant accident, injury, or fatality rate is an average 1-yr rate.
[f] New freeway segment; 3-yr of accident history not available.
[g] 1.917-yr period; based on the assumption that truck equals the overall accident rate.
[h] 2.75-yr period; accident, injury and fatality rates are an average 1-yr period.
[i] Truck value includes only combination tractor-trailer trucks.
[j] Estimated truck volume, based on typical values for given land use area.

Land Use Key: R = Rural, S = Suburban or Small Urban; U = Urban
N/A = Not Available

APPENDIX E

HYDRAULIC AND GEOTECHNICAL MEASUREMENTS

AT THE WIPP HORIZON

TABLE OF CONTENTS

E.1 **INTRODUCTION**

Appendix E contains excerpts from published documents that primarily support conclusions regarding the hydraulic and geotechnical characteristics of the Salado Formation. This appendix is not intended to provide a complete understanding of the various studies, but is intended to provide enough data and interpretation to provide the reader with an adequate level of information to independently assess the conclusions presented in the text.

In this final SEIS, the introductions to all sections (E.1 through E.7) are published, as well as a modified Section E.3; a new Sections E.8, Delineation of the Disturbed Rock Zone (DRZ); and a new Section E.9, Seal Design and Evaluation. The reader is referred to the draft SEIS for the complete sections E.1, E.2, and E.4 through E.7, which remain unchanged.

E.2 **BRINE INFLOW MEASUREMENTS**

This subsection of Appendix E describes preliminary sampling and evaluations of brine occurrences at the WIPP facility horizon. Included is a discussion and description of sampling methodology, the manner in which the data were used, calculations made, and a location-by-location description of sampling results.

This subsection was excerpted from Appendix D of Deal and Case, 1987, Brine Sampling and Evaluation Program, Phase I Report. This subsection is included to provide evidence of brine inflow rates defined in the text.

E.3 **BRINE INFLOW MODEL**

This subsection of Appendix E presents and describes the WIPP Darcian Brine Flow Model that has been used to analyze brine inflow rates to observed boreholes and moisture release experiments and is provided here to support brine inflow rates defined in the text. Included in this section are the assumptions inherent in the model.

This subsection has been excerpted from Chapters 2 through 6 of Nowak et al., 1988, Brine Inflow to WIPP Disposal Rooms: Data, Modeling, and Assessment. Sections specifically related to nonisothermal flow have been deleted. The nonisothermal aspect of the model was used to simulate inflow due to heat generated by high-level waste. Since high-level waste will not be disposed of at the WIPP, these sections are no longer pertinent. Some reference with respect to nonisothermal conditions is left in portions of the text to provide more generic aspects of the model development.

E.4 WIPP HORIZON GAS FLOW MEASUREMENT RESULTS SUMMARY THROUGH 1986

This subsection of Appendix E contains information and data on the WIPP facility horizon in situ flow tests and measurements conducted through 1986. Flow measurement tests can be grouped into three categories: 1984 tests, N1420 drift tests, and first storage panel tests. The results of these tests are briefly summarized in the following excerpt. More detail on the 1984 and N1420 tests can be found in this SEIS Appendix E and Subsections E.5 and E.6. This subsection is provided to support near-field permeability rates defined in the text.

This subsection is excerpted from Appendices B and C from Stormont et al., 1987, Summary of and Observations About WIPP Facility Horizon Flow Measurements through 1986.

E.5 1984 GAS FLOW MEASUREMENT TEST RESULTS

This subsection of Appendix E contains Phase I test results of in situ gas flow measurement results collected in 1984. A summary of this test and its results can be found in this SEIS Subsection E.4. This subsection is presented to support near-field horizon permeability rates detailed in the text.

This subsection was excerpted from Chapters 4 and 5 of Peterson et al., 1985, WIPP Horizon In-Situ Permeability Measurements.

E-133/134

E.6 **N1420 DRIFT GAS FLOW DATA ANALYSIS AND EVALUATION**

This subsection of Appendix E contains a description of gas flow measurement data collected during N1420 drift testing. A summary of this test and its results is presented in this SEIS Subsection E.4. This subsection is presented to support near-field horizon permeability rates defined in the text.

This subsection is excerpted from Chapters 4, 5, and 6 of Peterson et al., 1987, WIPP Horizon Free Field Fluid Transport Characteristics.

E.7 WASTE-HANDLING SHAFT PULSE TESTING DATA SUMMARY AND CONCLUSIONS

This subsection of Appendix E contains the test result summary and conclusions from testing in the waste-handling shaft. The results of this test were used to measure the far-field hydraulic conductivities within the Salado Formation. The far-field hydraulic conductivities were converted to permeabilities in the range of 10^{-20} to 10^{-21} m^2. See Table 5.3 in this SEIS for a summary of hydraulic conductivities and calculated permeabilities. This subsection is presented to support far-field permeability estimates.

The text, figures, and tables contained in this subsection are excerpted from Saulnier and Avis, 1988, Interpretation of Hydraulic Tests Conducted in the Waste-Handling Shaft at the Waste Isolation Pilot Plant (WIPP) Site. For a complete reference, please see the Appendix E reference list.

E.8 **DELINEATION OF THE DISTURBED ROCK ZONE (DRZ)**

This subsection presents a summary of the observations and measurements that have been conducted in the underground workings. Data collected from these investigations provide the initial results of an ongoing experimental program which is developing a more detailed three-dimensional definition of the DRZ.

This subsection was excerpted from Borns and Stormont, 1989, A Report on Excavation Effect Studies at the WIPP: The Delineation of the Disturbed Rock Zone surrounding excavations in salt.

E.9 SEAL DESIGN AND EVALUATION

This subsection of Appendix E evaluates the design concepts for the tunnel and shaft seals required for the WIPP, as they are presently envisioned. The principal design strategy involves the use of salt as the primary structural seal material, relying on creep closure of the surrounding host rock to compress this salt into a low-permeability plug. Key elements of the supporting experimental program are also outlined.

This subsection consists of Stormont (1988), entitled Preliminary Seal Design Evaluation for the Waste Isolation Pilot Plant.

REFERENCES FOR APPENDIX E

Borns, D.J., and J.C. Stormont, 1988. An Interim Report on Excavation Effect Studies at the Waste Isolation Pilot Plant; The Delineation of the Disturbed Rock Zone, SAND87-1375, Sandia National Laboratories, Albuquerque, New Mexico.

Deal, D. E., and J. B. Case, 1987. IT Corporation, Brine Sampling and Evaluation Program, Phase I Report, DOE-WIPP-87-008, prepared by the Engineering and Technology Department of the Management and Operating Contractor, Waste Isolation Pilot Plant Project, for the U.S. Department of Energy.

Nowak et al. (E. J. Nowak, D. F. McTigue, and R. Beraun), 1988. Brine Inflow to WIPP Disposal Rooms: Data, Modeling and Assessment, SAND88-0112, Sandia National Laboratories, Albuquerque, New Mexico.

Peterson et al. (E. W. Peterson, P. L. Lagus, and K. Lie), 1987. WIPP Horizon Free Field Fluid Transport Characteristics, SAND87-7164, Sandia National Laboratories, Albuquerque, New Mexico.

Peterson et al. (E. W. Peterson, P. L. Lagus, J. Brown, and K. Lie), 1985. WIPP Horizon In Situ Permeability Measurements, Final Report, SAND85-7166, Sandia National Laboratories, Albuquerque, New Mexico.

Saulnier, G. J., and J. D. Avis, 1988. Interpretation of Hydraulic Tests Conducted in the Waste-Handling Shaft at the Waste Isolation Pilot Plant (WIPP) Site, SAND88-7001, prepared for Sandia National Laboratories by INTERA Technologies, Inc.

Stormont, J. C., 1988. Preliminary Seal Design Evaluation for the Waste Isolation Pilot Plant, SAND 87-3083, Sandia National Laboratories, Albuquerque, New Mexico.

Stormont et al. (J. C. Stormont, E. W. Peterson and P. L. Lagus), 1987. Summary of and Observations about WIPP Facility Horizon Flow Measurements Through 1986, SAND87-0176, Sandia National Laboratories, Albuquerque, New Mexico.

APPENDIX F

RADIOLOGICAL RELEASE AND DOSE MODELING FOR PERMANENT DISPOSAL OPERATIONS

TABLE OF CONTENTS

LIST OF TABLES

F.1 **INTRODUCTION**

This appendix provides information concerning the radiological dose assessment modeling used to evaluate the risks associated with WIPP operations. A discussion of the AIRDOS-EPA computer model and its input parameters is provided, the concept of plutonium equivalent curies is explained, and descriptions of accident scenarios are presented. In response to numerous comments on the draft SEIS accident analysis, variations to the accident scenarios have been postulated in F.3 to consider alternate assumptions which result in more severe but less likely consequences. The credible accident scenario having the highest projected consequences is that of a postulated drum fire in the underground waste disposal area.

F.1.1 **OVERVIEW OF AIRDOS-EPA**

AIRDOS-EPA (Moore et al., 1979) estimates the radiation dose to either a maximally exposed individual or to an exposed population from the release of a specified quantity of radionuclides to the atmosphere. The code estimates concentrations of radioactivity in air, deposition buildup on ground surface, and ground surface concentrations based on release information, characteristics of the area surrounding the release site (e.g., agricultural productivity and land use), and specified meteorological conditions. These estimates, combined with intake rates for man, were used to estimate the radiation dose to an exposed adult human from potential exposure pathways for routine and accidental releases.

F.1.2 **METEOROLOGICAL MODELING**

The WIPP site area was modeled as a 50-mile-radius circular grid system with the site located at the center. Site-specific meteorological data, typical of annual average conditions, were specified for the assessment of routine annual releases. The annual frequency of wind direction was first determined for each of the 16 principal compass directions. The frequency of each Pasquill stability category, ranging from category A (very unstable) to category G (extremely stable), was then determined for each of the 16 directions. The average wind speed was entered for each wind direction and Pasquill category. The average depth of the atmospheric mixing layer (lid) for the area was specified to limit the vertical dispersion of the plume after it travels some distance downwind of the source. The lid value used applies to routine and accidental releases. For the assessment of accidental releases from the WIPP, stable meteorological conditions that allow minimal dispersion were assumed: a wind speed of 2 m/s under stability class F (very stable) conditions with wind direction constrained to a single direction for the maximum individual and annual average conditions with wind direction constrained to the direction having the highest consequences for the general population.

F.1.3 **STACK EFFLUENT MODELING**

The waste handling building stack and/or the exhaust shaft are the two possible release points for routine and accidental releases (release points are referred to as "stacks" for modeling purposes). AIRDOS-EPA requires input describing each area or point of release.

Because the air will be discharged from the "stacks" at a relatively high velocity, the release will effectively take place at a height above the physical stack. Models for momentum-dominated plumes (Rupp et al., 1948) were used to estimate effective stack heights for releases associated with routine operations and projected accidents. This method employed an effective "stack velocity" in the vertical direction to determine the effective height of the release since the discharge from the stack will be angled. The effective point of release was also offset to account for the angled discharge. For releases associated with postulated accidents, the effective stack heights were estimated using Rupp's equation and reflected actual stack velocity measured during the postulated accidental release.

F.1.4 **DISPERSION MODELING**

The Gaussian plume model of Pasquill (1961), as modified by Gifford (1961), estimates plume dispersion in the downwind direction. The values recommended by Briggs (1969) for the horizontal and vertical dispersion coefficients were used for dispersion and depletion calculations. The code permits consideration of dry deposition and scavenging for determining deposition of radionuclides on ground surfaces. Dry deposition is the process by which particles are deposited on grass, leaves, and other surfaces by impingement, electrostatic deposition, chemical reactions, or chemical reactions with surface components. The rate of deposition on earth surfaces is proportional to the ground-level concentrations of the radionuclides in the air (Slade, 1968).

Scavenging is primarily due to washout of particles from a plume by rain or snow and is, therefore, a function of the precipitation rate. The scavenging coefficient was averaged over an entire year, including periods during which rain or snow would not fall. Scavenging can thus be described as a continuous removal of a fraction of the plume per second over the entire year.

The value for the total ground deposition rate used in assessing routine releases was the sum of the dry deposition and the scavenging rates. The code removes the deposited fraction and maintains a mass balance along the plume as the concentration of the plume decreases. For the accidental release assessment, scavenging due to precipitation was conservatively ignored.

F.1.5 **TERRESTRIAL MODELING**

As previously stated, the area surrounding the WIPP site was modeled as a 50-mile radius circular grid system with the WIPP facilities located at the center. Within the grid, 20 distances were specified in each of the 16 compass directions. Each distance represented the midpoint of a sector. Eleven distances were specified within a 5-mile radius of the WIPP. The remaining nine distances were specified at about 5-mile incremental distances from the center of the site. Within each sector formed by the grid system, WIPP-specific data were used for population, agricultural area, surface-water area, and numbers of beef and dairy cattle. These data are summarized in Section 2.1 of the draft Final Safety Analysis Report (DOE, 1989).

Other factors used in modeling terrestrial and food crop transport are provided in U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide 1.109 (NRC, 1977). One-half of the anticipated operational life of the facility, 12.5 years, was specified as the period of time allowed for long-term buildup of radioactivity on surface soils.

F.1.6 **DOSE MODELING**

The AIRDOS-EPA computer model estimates radiological intake rates at specified environmental locations. Resultant doses are then calculated through various exposure modes, using the ground-level concentrations in air and ground deposition rates computed from the meteorological input. To estimate the collective population dose, average values in the crosswind direction over each sector were used for the air concentrations and ground deposition rates. The average individual dose was determined by dividing the population dose by the number of individuals in the exposed population. The dose to an individual receiving a maximum dose (maximally exposed) was determined directly by the code.

For accident assessments, it was assumed that the maximally-exposed individual was located on the center line of the discharge plume at the point of highest off-site ground-level concentration for the entire duration of the accident. The population dose for accident assessments was calculated using annual average meteorological conditions (wind speed and stability class frequency distribution) with a constant wind in the direction which maximizes the collective population doses.

Exposure pathways, primarily the air pathway, are discussed in Subsection 5.2.3.2. The model calculates doses to total body, lungs, red bone marrow, lower large intestine wall, stomach wall, kidneys, liver, endosteal cells, thyroid, testes, and ovaries. The doses calculated are 50-year Committed Effective Dose Equivalents (CEDE) resulting from a one-year exposure for routine releases or one-time exposure for accidental releases.

The internal dose conversion factors used in the calculation were those reported in Dunning (1986). The inhalation factors were based on the ICRP Task Group Lung Model, which simulates the behavior of particulate matter in the respiratory tract. The inhalation factors used correspond to a median aerodynamic diameter of 1 micron. The ingestion factors were based on a four-segment catenary model with exponential transfer of radioactivity from one segment to the next. Retention of nuclides in other organs was represented by linear combinations of decaying exponential functions. In the inhalation and ingestion models, cross-irradiation (irradiation of one organ by nuclides contained in another) is included.

The Dunning dose factors are based on the ICRP and NCRP models endorsed by the DOE in its August 5, 1985, Vaughan memorandum (DOE, 1985). Further, Dunning calculated dose factor using the same organ uptake fractions for daughter products as for the parent, as recommended in more recent ICRP guidance. Comparison of the Dunning dose factors with those recommended by the Vaughan memorandum (DOE, 1985) indicates that Dunning's approach is slightly more conservative. External dose rate conversion factors developed by Kocher (1981) are used.

Where the chemical form and solubility of nuclides in the source term was not known, the solubility class which yielded the highest effective dose commitment was used in the model. For the alpha emitters, a quality factor of 20 was used in the calculation as recommended in ICRP Publication 26 (ICRP, 1977).

Input parameters to the AIRDOS-EPA model specific to the WIPP site are documented in Tables F.1 through F.11.

TABLE F.1 Meteorological data: assessment of routine releases

Parameter	Value (units)
Lid height	1,435 (m)
Average temperature	288.8 (^oK)
Average rainfall	24.13 (cm/yr)
Frequency of atmospheric stability classes for each direction	Table F-2
Frequencies of wind directions and true-average wind speeds	Table F-3
Frequencies of wind directions and reciprocal-average wind speeds	Table F-4
Pasquill Category Temperature Gradients[a]	
E	0.0055 ($^oK/m$)
F	0.0280 ($^oK/m$)
G	0.0400 ($^oK/m$)

[a] Categories A-D are not utilized in the AIRDOS-EPA Code; Categories E-G are AIRDOS-EPA Code default values.

TABLE F.3 Frequencies of wind directions and true-average wind speeds

Wind toward[a]	Frequency	Wind speeds for each stability class (meters/sec) A	B	C	D	E	F	G
1	0.091	3.90	2.62	2.62	3.69	3.29	3.58	2.40
2	0.151	4.36	3.91	3.25	3.94	4.79	5.54	3.03
3	0.188	3.94	3.77	3.85	3.86	4.18	4.54	2.94
4	0.085	3.28	4.00	3.87	3.95	3.93	3.32	2.45
5	0.052	4.46	5.32	6.61	5.33	5.39	4.80	3.01
6	0.049	4.67	5.10	6.25	5.65	6.18	5.16	2.93
7	0.043	4.40	2.98	3.05	4.17	4.90	4.04	2.65
8	0.033	4.06	3.38	4.36	4.23	4.29	3.57	2.65
9	0.034	4.25	4.28	3.15	3.87	4.40	3.74	2.70
10	0.031	4.02	2.26	2.25	3.16	3.52	3.97	2.94
11	0.029	3.57	2.26	2.76	3.31	3.41	4.54	2.79
12	0.031	4.28	3.18	0.85	3.08	4.88	5.21	3.36
13	0.050	5.64	3.37	5.11	4.74	5.10	6.01	3.57
14	0.042	4.84	0.85	4.10	3.73	3.40	5.39	3.01
15	0.038	3.75	3.60	4.08	2.73	3.58	2.90	2.63
16	0.053	3.54	2.27	3.15	2.74	2.75	2.11	2.23

[a] Wind directions are numbered counterclockwise starting at 1 for due north.

TABLE F.4 Frequencies of wind directions and reciprocal-average wind speeds

Wind toward[a]	Frequency	Wind speeds for each stability class (meters/sec) A	B	C	D	E	F	G
1	0.091	3.11	2.00	2.00	2.71	2.58	2.78	2.40
2	0.151	3.46	2.74	2.99	2.76	3.35	4.45	3.03
3	0.188	3.04	2.46	3.21	3.09	3.04	3.55	2.94
4	0.085	2.51	3.20	3.37	2.84	2.93	2.50	2.45
5	0.052	3.31	4.09	5.99	4.08	3.68	3.64	3.01
6	0.049	3.11	3.59	5.55	3.81	4.16	3.69	2.93
7	0.043	3.12	1.80	2.80	2.85	3.46	2.57	2.65
8	0.033	2.84	2.28	2.84	2.75	3.21	2.34	2.65
9	0.034	3.00	2.12	2.89	1.99	2.70	2.08	1.91
10	0.031	2.75	1.47	2.25	1.71	2.04	2.51	2.02
11	0.029	2.52	1.40	3.10	1.99	2.30	2.76	2.01
12	0.031	2.68	1.96	0.85	1.47	1.94	2.55	2.11
13	0.050	3.57	1.76	2.64	2.39	2.71	4.35	2.15
14	0.042	3.14	0.85	2.02	1.99	1.71	4.23	2.11
15	0.038	2.50	2.42	2.05	1.62	2.19	1.76	1.83
16	0.053	2.70	1.21	2.89	2.04	1.83	1.46	1.63

[a] Wind directions are numbered counterclockwise starting at 1 for due north.

TABLE F.5 Stack information

Parameter	Waste handling building	Storage exhaust filter building
Number of stacks	1	2
Physical stack height	14.9 (m)	8.2 (m)
Stack diameter	2.4 (m)[a]	4.4 (m)
Velocity of stack gas	9.5 (m/s)	6.7 (m/s)

[a] Equivalent diameter.

TABLE F.6 Terrestrial modeling assumptions

Parameters	Value (units)	Basis
Buildup time for surface deposition	4,562.5 (day)	
Fraction of locally grown produce Conservatism	1.0	
Fraction of radioactivity retained on leafy vegetables after washing	0.5	NRC, 1977
Time delay for ingestion:		
Pasture grass by animals	0 (hr)	NRC, 1977
Stored feed by animals	2160 (hr)	
Leafy vegetables by man	24 (hr)	
Produce by man	24 (hr)	
Removal rate constant for physical loss by weathering	2.1×10^{-3} (hr^{-1})	NRC, 1977
Period of exposure during growing season:		NRC, 1977
Pasture grass	720 (hr)	
Crops and leafy vegetables	1440 (hr)	
Agricultural productivity per unit area:		Baes and Orton, 1979
Grass-cow-milk pathway	0.28 (kg/m^2)	
Produce and leafy vegetable	1.9 (kg/m^2)	
Effective surface density of soil	240 (kg/m^2) 1979	Moore et al.,
Fraction of yearly and daily feed from pasture Conservatism	1.0	
Consumption rate of contaminated feed or forage by animals (fresh weight)	15.6 (kg/day)	Baes and Orton, 1979
Transport time from animal Feed-milk-man	2.0 (day)	NRC, 1977
Average time from slaughter of meat to consumption	20.0 (day)	NRC, 1977

TABLE F.6 Concluded

Parameters	Value (units)	Basis
Fraction of meat-producing herd slaughtered each day	2.74×10^{-3}	Conservatism
Muscle mass of meat-producing animal	200 (kg)	Site-specific evaluation
Milk production of cow	11 (l/day)	Site-specific evaluation
Fallout interception fraction:		
Pasture	0.57	Miller, 1979
Vegetables	0.20	NRC, 1977
Fraction of food grown in local gardens:		Conservatism
Produce	0.76	
Leafy vegetables	1.00	

TABLE F.7 Bioaccumulation factors[a]

Element	Uptake fraction Milk (days/l)	Uptake fraction Meat (days/kg)	Concentration factor Pasture	Concentration factor Crops
Cobalt	2.0×10^{-5}	2.0×10^{-2}	2.0×10^{-2}	3.1×10^{-3}
Strontium	1.5×10^{-3}	3.0×10^{-4}	2.5×10^{0}	1.1×10^{-1}
Ruthenium	6.0×10^{-7}	2.0×10^{-3}	7.5×10^{-2}	8.7×10^{-3}
Antimony	1.0×10^{-4}	1.0×10^{-3}	2.0×10^{-1}	1.3×10^{-2}
Cesium	7.0×10^{-3}	2.0×10^{-2}	8.0×10^{-2}	1.3×10^{-2}
Cerium	2.0×10^{-5}	7.5×10^{-4}	1.0×10^{-2}	1.7×10^{-3}
Plutonium	1.0×10^{-7}	5.0×10^{-7}	4.5×10^{-4}	2.0×10^{-5}

[a] From Baes et al., 1984.

TABLE F.8 Dose receptor assumptions

Parameter	Value (units)	Basis
Breathing rate of man	1.26×10^6 (cm^3/hr)	Conservatism
Depth of water for immersion dose	244 (cm) Conservatism	
Fraction of time spent swimming	0.01	Conservatism
Rate of human ingestion		NRC, 1977
Average individual:		
Produce	190 (kg/yr)	
Milk	110 (l/yr)	
Meat	95 (kg/hr)	
Leafy vegetables	18 (kg/yr)	
Maximum individual:		
Produce	520 (kg/yr)	
Milk	310 (l/yr)	
Meat	110 (kg/yr)	
Leafy vegetables	64 (kg/yr)	

Table F.9 Dose rate conversion factors[a]

Radio-nuclide	Decay constant (day^{-1})	Photon dose rate conversion factors: Immersion in air (rem-cm^3/μCi-hr)	Immersion in water (rem-cm^3/μCi-hr)	Surface (rem-cm^2/μCi-hr)
Co-60	4.96×10^{-4}	2.465×10^{3}	5.360×10^{0}	4.305×10^{-1}
Sr-90	8.98×10^{-5}	0	0	0
Ru-106	1.88×10^{-3}	0	0	0
Sb-125	2.50×10^{-1}	4.204×10^{2}	9.159×10^{-1}	8.948×10^{-2}
Cs-137	8.72×10^{-5}	0	0	0
Ce-144	2.44×10^{-3}	1.785×10	4.124×10^{-2}	4.558×10^{-3}
Pu-239	7.78×10^{-8}	5.655×10^{-1}	1.431×10^{-3}	1.27×10^{-3}

[a] From Kocher, 1981.

F.2 PLUTONIUM-EQUIVALENT CURIE

The PE-Ci is intended to eliminate the dependency of radiological analyses on the specific radionuclide composition of a TRU waste stream. A unique radionuclide composition and/or waste disposal distribution is associated with each TRU waste generator and storage facility. By normalizing all radionuclides to a common radiotoxic hazard index, radiological analyses can be conducted for the WIPP which are independent of these variations. Plutonium-239, as a common component of defense TRU wastes, was selected as the radionuclide to which the radiotoxic hazard of other TRU radionuclides could be indexed. Since TRU radionuclides primarily represent inhalation hazards, a valid relationship can be established which normalize the inhalation hazard of a TRU radionuclide to that of Pu-239.

To obtain this correlation, the 50-year CEDE or dose conversion factor (DCF) for a unit intake of each radionuclide is used. These DCFs have been determined by the method described in International Commission on Radiological Protection (ICRP) Publications 26 and 30 (ICRP-26, 1977; ICRP-30, 1979).

For a known quantity of radioactivity and radionuclide distribution, the Pu-239 equivalent activity is determined using radionuclide-specific weighting factor. The Pu-239 equivalent activity (AM) can be characterized by:

$$AM = \sum_{i=1}^{K} \frac{A_i}{WF_i}$$

where:

K = the number of TRU radionuclides
A_i = the activity of radionuclide i
WF_i = the PE-Ci weighting factor of radionuclide i.

WF_i is further defined as the ratio:

$$WF_i = \frac{E_o}{E_i}$$

where:

E_o(rem/Ci) = the 50-year effective whole-body dose commitment due to the inhalation of Pu-239 particulates with a 1.0 μm AMAD (activity median aerodynamic diameter) and a W pulmonary clearance class.

E_i(rem/Ci) = the 50-year effective whole-body dose commitment due to the inhalation of radionuclide i particulates with a 1.0-μm AMAD and the pulmonary clearance class resulting in the highest 50-year effective committed dose equivalent.

The values of E_o and E_i can be obtained from Dunning (1986, Appendix I). Weighting factors calculated in this manner are presented in Table F.12 for selected radionuclides of interest.

TABLE F.12 PE-Ci weighting factors for selected radionuclides

Radionuclide	Pulmonary clearance class[a]	Weighting factor
Uranium-233	Y	4.0
Neptunium-237	W	1.0
Plutonium-236	W	3.1
Plutonium-238	W	1.1
Plutonium-239	W	1.0
Plutonium-240	W	1.0
Plutonium-241	W	52.0
Plutonium-242	W	1.1
Americium-241	W	1.0
Americium-243	W	1.0
Curium-242	W	29.0
Curium-244	W	1.9
Californium-252	Y	3.5

[a] W = Weekly; Y = Yearly.

F.3 DESCRIPTIONS OF ACCIDENT SCENARIOS ANALYZED IN THE SEIS

Operations at the WIPP and accident scenarios postulated in the FEIS have been re-evaluated. This SEIS, consistent with the draft FSAR (DOE, 1989), discusses eleven potential accidents involving CH waste (accidents C0 through C10) and six involving RH waste (accidents R1 through R6). These accidents are derived from potential human error or equipment failures. Additional information concerning the accident scenarios described below appears in Section 7.3 of the draft FSAR. The potential extent of damage to the waste containers involved and the amount of activity released as a result of the accident scenarios are provided below.

The SEIS maintains the assumptions used in Section 7.3 of the draft FSAR (DOE, 1989) except in two areas: the SEIS considers a range of assumed waste container radioactivity content for all accident scenarios where a radioactive material release is postulated; and the SEIS evaluates worker dose assuming that workers will remain at their stations for the full duration of the postulated accidents. Consistent with established operational plans that require workers to wear respirators when handling a waste container with greater than 100 PE-Ci, worker exposure for accidents involving waste containers at higher radioactivity loadings is assumed to be mitigated by a respiratory protection factor of 50.

F.3.1 ACCIDENTS INVOLVING CH WASTE

C0: Forklift Tine Strikes TRUPACT-II in Radiological Control Area. The new TRUPACT-II design necessitates the removal of the TRUPACT-II from the transport trailer in the Radiological Control Area prior to moving the TRUPACT-II into the waste handling building. It is postulated that the forklift may be misaligned and that the forklift tine may strike the TRUPACT-II and cause it to fall off the transporter. Such a fall is not postulated to cause any release because the test conditions for the TRUPACT-II are more severe than this accident.

C1: Vehicle Collision with a Shipping Container in Off-Loading Area. Vehicles transporting waste from offsite will travel at a very low speed (5 to 10 miles/hour) in the off-loading area. A vehicle collision accident would cause less damage to shipping containers on the vehicle than if the containers fell 30 ft, since a 30-ft free fall would result in an impact velocity of 30 mile/hour. DOE regulations specify that a Type B package must be capable of withstanding a 30-ft drop without releasing radioactive material. Since the shipping container is a Type B package, no activity is postulated to be released in this vehicle collision accident.

C2: Drum Drop from a Forklift in the Inventory and Preparation Area. It is postulated that during the handling process a bundle of CH TRU waste drums is dropped from a forklift in the inventory and preparation area. Since the waste drums are Type A packages (per 49 CFR), they are designed and tested to withstand a 4-ft drop onto an unyielding surface without being damaged enough to release any activity. However, since the vertical lift exceeds the rated design, it is assumed that the drop and subsequent crushing by the weight of the drum bundle causes the lid of one drum to be knocked off and the inner plastic liner to tear.

Because of the short distance of the drop, it is assumed that 25 percent of the drum contents is spilled. Of the spilled fraction, 0.1 percent is assumed to be resuspended in the room air. It is conservatively assumed that 5 percent of the total radioactivity contained in the drum is contained in the allowed fraction (1 weight percent) that is less than 10 microns in diameter.

Consistent with the assumed frequency of this event, the drum is assumed to contain the average drum content of 12.9 PE-Ci of radioactivity. Since depletion of activity in the room air was considered to be equivalent to resuspension, the total amount of suspended radioactivity in the room air is 1.6×10^{-4} PE-Ci. Credit was taken for the permanently installed on-line high-efficiency particulate air (HEPA) filters, which reduce the total source term to the environment by a factor of 10^{6}. Thus, the total activity released to the environment is 1.6×10^{-10} PE-Ci.

To assess the adequacy of facility design and operating procedures with respect to worker safety, the dose consequences to workers have been estimated. Workers in the immediate vicinity of the postulated accident were assumed to respond as trained and immediately exit the work area. Due to the expected slow rate of contamination spread, internal deposition was therefore not estimated for these workers. Although it is unlikely that other workers in the inventory and preparation area would not be made aware of the accident, it was assumed that a worker would remain. The total activity inhaled by this worker is, therefore, related to how long he/she remains in the area before becoming aware of the incident and exits, the distance from the location of the accident and how rapidly the release spreads. For the purpose of this analysis, the spread of activity was modeled as a hemisphere with an initial volume corresponding to that of a 55-gal drum. The hemisphere is assumed to expand in all directions at a rate equivalent to the ventilation flow rate for the inventory and preparation area (about 25 cm/s). This expanding "cloud" was assumed to spread to a worker in the neighboring work area, conservatively estimated to be about 20 ft away and to remain at that location indefinitely. At an assumed breathing rate of 20 liters per minute (ICRP-23, 1974), the total activity calculated to be inhaled by the worker is 1.4×10^{-9} PE-Ci. Because workers are trained to leave the work area in the event of an accident that could damage a waste container, this estimate is considered to be conservative.

To evaluate more severe but less likely accident scenarios involving a drum drop, two variations on the above scenario have been postulated. These scenarios assume that the drum involved contains 100 PE-Ci of activity and 1,000 PE-Ci of activity, respectively. These are considered to be limiting events, i.e., not expected to occur during the operational lifetime of the WIPP. The 1,000 PE-Ci case is based upon the maximum allowable activity content of a waste container, as provided by the WIPP Acceptance Criteria (see Appendix A). The former case results in an environmental release of 1.3×10^{-9} PE-Ci of activity from the waste handling building and a maximum theoretical exposure to a worker of 1.1×10^{-8} PE-Ci inhaled. The latter case results in an environmental release of 1.3×10^{-8} PE-Ci, and a maximum theoretical exposure to a worker of 2.1×10^{-9} PE-Ci inhaled, reduced as a result of the protection factor of 50 offered by his/her respirator.

<u>C3: Drum(s) Punctured by a Forklift in the Inventory and Preparation Area.</u> An operator error may result in a forklift hitting a stack of CH TRU waste drums. It was conservatively assumed that two drums were punctured as a result of the collision, and that the lid of a third drum was knocked off as it fell from the stack. Operating procedures caution the operator not to back away from a puncture, but it was assumed that the drums would become disengaged and spill some of the waste. Since not all of the waste would fall out of the damaged drums, it was assumed that 10 percent of the radioactive content was released form each punctured drum and 25 percent of the radioactive content was released from the drum that lost a lid. As for accident C2, of the spilled fraction that is less than 10 microns in diameter, 0.1 percent was assumed to be resuspended in the room air. Consistent with previous analyses, it was assumed that 5 percent of the total radioactivity contained in the drums was contained int he allowed fraction (1 weight percent) that is less than 10 microns in diameter. Consistent with the frequency of the event, it was further assumed that the drums would contain an average loading of 12.9 PE-Ci each. Therefore, 2.9×10^{-4} PE-Ci of radioactivity was suspended in the

room air. Credit was taken for the continuously operating on-line HEPA filters, which reduced the total release to the environment by a factor of 10^6. The consequence of this postulated accident was a discharge to the environment of 2.9 x 10^{-10} PE-Ci.

A worker in an adjacent area could inhale 2.5 x 10^{-9} PE-Ci based on the exposure model described in C2. Again, the worker's dose commitment is expected to be much smaller than that projected in the SEIS because workers will be trained to evacuate the work area immediately after any accident that could damage a waste container.

As with accident scenario C2, more severe and less likely variations on accident scenario C3 have been evaluated for this SEIS. These variations assume that the drum with the highest release fraction, the one that loses its lid, contains 100 PE-Ci and 1,000 PE-Ci, respectively. The other two drums are assumed to contain an average activity content of 12.9 PE-Ci per drum. For the 100 PE-Ci case, an environmental release of 1.4 x 10^{-9} PE-Ci is calculated with a worker exposure of 1.2 x 10^{-8} PE-Ci inhaled. The 1,000 PE-Ci case results in an environmental release of 1.3 x 10^{-8} PE-Ci and a worker exposure of 2.2 x 10^{-9} PE-Ci inhaled.

C4: Transporter Hits a Pallet in the Underground Waste Disposal Area. Operator error may result in the transporter striking a pallet of CH TRU waste drums in the underground waste disposal area causing the drums to fall. Although it is unlikely that such an incident would cause sufficient damage to the drums to result in an activity release, it was conservatively assumed that the lid of one of the drums would be knocked off because of the fall and the inner liner tears.

This accident scenario results in a release from the drum identical with the release for accident C2, with the exception that it occurs within the underground waste disposal area. Because of the long distance from the location of the accident to the release point, particle deposition and resuspension were considered. The net result is a conservative estimate of depletion of the released activity by only 20 percent prior to reaching the outside environment. Although they are designed to be activated in case of an accidental release of radioactivity underground, no credit was taken for HEPA filters because they are not continuously on-line and require activation manually or by radiation detection instruments. For the purpose of this analysis, the detection instruments were not assumed to activate the HEPA filters. Occurrence of this postulated accident resulted in a release of 1.3 x 10^{-4} PE-Ci from the exhaust shaft stack.

Due to the longer distance of travel between the point of release and the worker and the higher rate of airflow within the mine, the release and subsequent exposure were modeled somewhat differently than a release in the waste handling building. For this accident, the release to the drift was assumed to be homogeneously distributed within a segment of the mine volume equivalent to a 4.0 x 3.4 x 6.1 meter cloud volume (8.3 x 10^7 cm^3). The worker was subject to exposure during the cloud passage time, approximately 15 seconds based upon a linear ventilation flow rate of 300 cm per second. As a result of this postulated accident, this worker could inhale 8.6 x 10^{-10} PE-Ci. This was considered conservative since the area downstream of the active waste disposal room would normally be unoccupied.

More severe, but less likely, variations of this scenario have been evaluated in this SEIS. These result in an environmental release of 1.0 x 10^{-3} PE-Ci of activity, and a worker exposure of 6.7 x 10^{-9} inhaled under the assumption of a 100 PE-Ci drum being involved. For the 1,000 PE-Ci drum variation, the environmental release would be 1.0 x 10^{-2} PE-Ci, and a worker exposure of 1.3 x 10^{-9} PE-Ci inhaled.

C5: Drum Drop from a Forklift in the Underground Waste Disposal Area. This accident and its consequences are bounded by the accident described in C4.

C6: Drums are Punctured by a Forklift or Other Machine in the Underground Waste Disposal Area. The conditions for this accident, including drum inventories and releases, were the same as described in C3 except that no credit was taken for HEPA filters. However, since the environmental release actually occurs at some distance from the location of the accident, depletion of the released activity in the underground was considered. As discussed for accident C4, depletion accounts for removal of 20 percent of the activity released from the drums.

The release to the environment from this accident was 2.3×10^{-4} PE-Ci. Worker exposure is modeled as for accident C4. The worker was calculated to inhale 1.6×10^{-9} PE-Ci. More severe, but less likely, variations on this scenario result in environmental releases of 1.1×1.0^{-3} PE-Ci and 1.0×10^{-2} PE-Ci and worker exposures of 7.4×10^{-9} PE-Ci inhaled and 1.4×10^{-9} PE-Ci inhaled for the 100 PE-Ci and 1,000 drum variations, respectively.

C7: Spontaneous Ignition in a Drum (Waste Handling Building). Although the WIPP WAC controls the types/quantities of pyrophoric materials that could be shipped to the WIPP, and therefore reduces the likelihood of fire in a waste container, the annual probability of a spontaneous ignition occurring during the processing of a container through the waste handling building was estimated based on past operational experience. The operational database indicated that for roughly 1.8 million container-years of operation with TRU-type waste similar to that to be handled at the WIPP, there has been only one recorded instance of a container fire. Contributing circumstances to this occurrence included the drum being painted black, exposure to direct sunlight, and improper packaging material. At the WIPP, the containers are painted white, not exposed to direct sunlight, and would be certified to WAC requirements. Due to these reasons, the low historic probability of a spontaneous ignition, and the short residence time of waste containers in the waste handling building, this accident was not considered to be a reasonably foreseeable event at this location. Off-site impacts of this accident are bounded by accident C10.

C8: Hoist Cage Drop. The design features of the waste hoist and cage are discussed in Section 4.3 of the draft FSAR. The hoist cage is equipped with multiple cables, providing a safety factor that makes its failure a very unlikely event. In the absence of a detailed assessment of the probability of a hoisting system failure, the WIPP Final Environmental Impact Statement evaluated the consequences of a hoist drop accident scenario. A review of Mine Safety and Health Administration reports on hoisting systems has since been conducted. The review concludes that hoisting system failure resulting in dropping waste down the shaft has an annual probability of 1.7×10^{-8} or about one catastrophic hoist accident in 60 million years of operations. Under the complete sequence of events (see below), the DOE does not consider this scenario to be reasonably foreseeable or the exposure risks to be significant. Nevertheless, because of commenters' interest (in particular the Environmental Evaluation Group), the SEIS has evaluated the consequences of such an accident.

In order to evaluate this event, a complete scenario must be postulated which describes the details of the accident. These details include:

- whether the hoist has waste on the conveyance at the time of the accident,
- the size of the radioactive payload,
- the fraction of the radioactive material which is respirable,

- the percent of the radioactivity released in the accident,
- the percent of the radioactivity which plates out or deposits on surfaces of the mine and shaft during its passage to the atmosphere,
- whether the HEPA filtration system is activated,
- the meteorological conditions including wind speed, direction, and atmospheric stability class (relates to dispersion and mixing of materials in the air), and
- the location of the individual receiving the exposure.

The specific assumptions are critical in estimating the severity of the accident consequences. The complete scenario can use assumptions ranging from very conservative to "nominal". In general, the more conservative the assumptions, the more severe the estimated consequences and the less likely the scenario is to occur.

For example, as shown in Table F. 13, the estimate of dose to the hypothetical maximally exposed individual could range from 190 rem using very conservative assumptions to about 7 millirem using more likely or "nominal" assumptions. The likelihood of these scenarios is estimated to range from a probability of about 1×10^{-17} for the 190 rem to about 1×10^{-9} for the 7 millirem exposure.

<u>C9: Diesel Fuel Fire in CH TRU Waste Disposal Area</u>. In the interest of improved safety, engineering changes have been incorporated that render the underground diesel-fuel fire scenario in the CH TRU Waste Disposal Area a scenario that is not reasonably foreseeable. These design changes can be summarized as follows:

1) All underground diesel vehicles will have a governor that limits speed to 20 mph. This effectively limits the impact energy associated with a vehicle accident.

2) All diesel fuel tanks will comply with specification SAE J703a. This specification requires that the fuel tank survive a 30-ft drop test onto a flat nonyielding surface. (The 30-ft drop is equivalent to a 30-mph impact.) Further, all fuel tanks will be located within the vehicle structure so that they are protected from puncture.

3) All non-steel fuel lines will have braided steel armor and be mounted such that they are protected from abrasion, impact, and operating damage.

4) The fuel tank size will be limited to 60 gallons.

<u>C10: Fire Within a Drum Underground</u>. This postulated accident was similar to C7, previously described. However, due to the length of time the drums would be present in the underground relative to the time spent in the waste handling building, spontaneous ignition within a drum was more conceivable following emplacement within the waste disposal area. Should a fire occur within a drum within a waste disposal area, it is not expected to propagate to adjacent waste containers.

Since waste containers will spend essentially all of their time in the waste disposal area, the probability of a drum fire will be highest in this area and will subsequently be evaluated. For the purpose of bounding all reasonable foreseeable accident consequences, the drum involved was assumed to contain 1,000 PE-Ci of radioactivity.

Since only a small fraction of drums in the existing stored waste inventory have a radioactivity content that exceeds 100 PE-Ci, the probability that a 1,000 PE-Ci drum would be involved is very small. Based upon empirical data (Mishima and Schwendiman, 1973), the spontaneous ignition was assumed to aerosolize 0.25 percent of the radioactivity content and this entire aerosolized fraction was released to the underground drift. This release was subject to a high amount of deposition due to the heated aerosol reacting with the relatively cool surfaces within the facility. This deposition was estimated to result in a depletion fraction of approximately 80 percent (Mishima and Schwendiman, 1973). As modeled in accident C4, although the HEPA filtration system is designed to be activated in response to an accidental release of

TABLE F.13 Catastrophic hoist accident[a]

Event	Very Conservative		Nominal	
	Assumption	Probability	Assumption	Probability
Hoist drop		1.7×10^{-8}		1.7×10^{-8}
Radioactive Payload (PE-Ci)	1,350[b]	1.0×10^{-2}	360[c]	1.0
Percent respirable (%)	5	2.0×10^{-2}	0.1	1.0
Percent released (%)	100	1.0×10^{-1}	10	1.0
Percent deposition (%)	20	2.5×10^{-1}	80	1.0
Meteorology (class, speed)[d]	F,2	5.0×10^{-2}	C,2	1.0
Receptor location	Boundary[e]	1.0×10^{-2}	Mills Ranch[f]	4.3×10^{-2}
Probability (per year)		4×10^{-17}		7×10^{-10}
Maximum Individual dose[g] (rem)	1.9×10^{2}		7×10^{-3}	

[a] For consistency throughout the document no credit is taken for the HEPA filters from the underground. It is also assumed the hoist is loaded with TRU waste at the time of the accident.
[b] One maximum loaded drum (1,000 PE-Ci) and 27 average drums (12.9 PE-Ci)
[c] Twenty-eight average drums.
[d] Meteorology is expressed in terms of atmospheric stability class and wind speed in meters per second.
[e] WIPP secured area boundary.
[f] Nearest permanent residence.
[g] Committed effective dose equivalent.

radioactive materials underground, no credit for filtration was assumed in this assessment. The environmental release from the waste disposal exhaust shaft assuming the absence of HEPA filtration was 0.5 PE-Ci. Waste is emplaced and stored downstream of workers and, therefore, no dose consequence to an underground worker is postulated for this event.

F.3.2 **ACCIDENTS INVOLVING RH WASTE**

R1: Crane Impacts on a Shipping Cask in the Receiving Area. Since the crane velocity and travel distance are limited, and the distance available for a shipping cask drop is less than 30 ft, a postulated accident involving a crane hitting a shipping cask is less severe than that of a cask free-falling 30 ft to an unyielding surface. A Type B package must withstand a 30-ft free-fall without significant damage. Since the shipping cask is a Type B package, no significant activity was considered to be released as a result of this postulated accident.

R2: Shipping Cask Drops in the Receiving Area. A cask dropped in the receiving area will fall less than 30 ft. Since the shipping cask is a Type B package, no activity was considered to be released for this postulated accident.

R3: Shipping Cask Drops in the Cask Preparation Area. A cask dropped in the cask preparation area will drop less than 30 ft. Since the shipping cask is a Type B package, no significant activity is considered to be released for this postulated accident.

R4: RH TRU Waste Canister Drops from Hot Cell into the Transfer Cell. It is possible that a canister containing RH TRU waste could be dropped into the transfer cell from the hot cell (a distance of about 36 ft) in the event that a grapple fails. Even with a drop over this distance, it is unlikely that a canister would be damaged enough to result in any release of radioactivity. However, for this SEIS analysis, it is assumed that the canister does breach and one percent of its total radioactive contents is released. Five percent of the radioactivity released is assumed to be less than 10 microns in diameter and 0.1 percent of this is assumed to be resuspended in the transfer cell. Depletion and resuspension are traded off equally, and the total amount of radioactivity that becomes airborne is assumed to be reduced by the HEPA filters, which provide a 10^{-6} reduction in the source term. The canister is assumed to contain a total of 2.5×10^{3} Ci of radioactivity, including 1,000 PE-Ci of transuranics. Based on these assumptions, 3.4×10^{-10} PE-Ci of fission and activation products and 5.0×10^{-10} PE-Ci of transuranics would be released to the environment. Since the transfer cell and hot cell are not occupied during canister transfer operations, doses to workers inside the facility are not calculated.

R5: Hoist drop with a Canister of RH TRU Waste. As discussed in accident C8, catastrophic failure of the hoisting system is not a reasonably foreseeable scenario. A beyond design basis accident involving a RH TRU waste canister is considered to be bounded by C8 because only a single canister is permitted on the hoist, and this would be contained within a thick-walled facility cask.

R6: Fire Involving RH TRU Waste. RH TRU waste is transferred from the waste shaft to an appropriate waste disposal area by the diesel-powered RH waste transporter. The waste is contained within a sealed steel canister and the canister is transported inside a shielded cask. The waste disposal operation consists of horizontally emplacing an RH TRU waste canister into a borehole and then plugging the borehole with a shield plug; experimental waste canisters are emplaced in vertical boreholes, which are subsequently backfilled. One canister is handled at a time, and after emplacement in complete, the contents are isolated from all credible accidents. Prior to emplacement, the canister is contained within the facility cask and the combination of this cask and the steel canister prevents the waste from becoming involved in any credible fire during a handling accident. Therefore, a fire involving a TRU waste canister would not result in any significant release of radioactivity to the environment or exposure to operating personnel.

F.3.3 **ACCIDENTS INVOLVING FLAMMABLE OR DETONABLE GASES**

As an extension to the question of operational safety issues that may be associated with gas generation, an assessment was conducted of the potential mechanisms and rates of generation of potentially flammable and/or detonable gases, the conditions under which accumulation could conceivably occur, the ignition sources that could lead to burning or detonation of accumulated gas, and the implications to operational safety of the WIPP, both during the Test Phase and Disposal Phase.

Background. The principal means of gas generation in TRU wastes are hydrogen generated through the radiolytic degradation of organic matrix wastes and hydrogenous materials, hydrogen generation through anaerobic corrosion of metals, and flammable gas production (principally methane) by microbial activity. These methods of gas generation are highly variable and closely associated with the composition of waste in individual waste containers and the environmental conditions to which the wastes and containers are subjected. Hydrogen generation through anaerobic corrosion of metals in the wastes or the waste containers is predominantly of concern in the long term and only if brine has accumulated in sufficient quantity within the decommissioned underground. However, gas generation associated with radiolytic and microbial degradation of the wastes is expected during the Disposal Phase, as well as in the long term.

Based upon existing laboratory data, it is estimated that radiolysis would produce hydrogen in CH TRU wastes at a rate of about 0.05 moles per drum per year. Microbial degradation realistically would produce gas at an average rate of 0.5 moles per drum per year, one-half of which is conservatively assumed to be methane. Therefore, for the purpose of assessing operational safety concerns with handling and storage of waste containers, flammable and/or explosive gas generation rates of 0.05 and 0.25 moles per drum per year were used to evaluate radiolytic and microbial degradation mechanisms, respectively (Slezak and Lappin, 1990).

Accumulation of flammable or detonable gases is principally of concern when sufficient oxygen is also present, i.e., at least 5.0 percent oxygen by volume in the case of hydrogen and 12.1 percent oxygen in the case of methane. If insufficient oxygen exists, a fire or detonation of the gas mixture is not a reasonably foreseeable event. Significantly, the very mechanisms for generation of hydrogen can also consume oxygen as is the case with radiolytic- and corrosion-produced hydrogen, and anaerobic production of methane requires the near absence of oxygen. Flammability and detonability of these gases also requires the presence of an ignition source. Since the presence of any potential ignition source such as a static electric charge cannot be completely ruled out, the assessment was conducted assuming that an ignition source could exist.

Individual Containers. All containers of CH TRU waste proposed to be shipped to the WIPP would be fitted with a carbon composite filter vent to prevent the overpressurization of the containers due to gas generation. Measurements of the diffusion rate of hydrogen through these vents have been conducted, the lowest measured rate being 1.9×10^{-6} moles per mole fraction per second. These measurements indicate that hydrogen generation rates below 2.4 moles per drum per year will maintain the hydrogen content of a container below the lower flammability limit for hydrogen, 4.0 percent by volume. This rate exceeds the expected upper generation rate due to radiolysis in WIPP waste. The lower detonability limit for hydrogen gas is approximately four times higher and, thus, is of even lesser concern.

The lower flammability limit for methane is 5.3 percent by volume. A methane generation rate below 3.2 moles per drum per year is sufficient to preclude a potential methane-induced fire. This rate exceeds any methane generation rate observed for TRU waste. The lower detonability limit for methane gas is slightly higher than the flammability limit and, thus, is also of lesser concern.

Detonation of methane gas within a waste container is further precluded by the geometry requirements for a methane detonation, i.e., detonation requires the existence of an unobstructed open space at least one-half the volume of a 55-gallon drum in size. More significantly, anaerobic bacteria, the principal potential source of methane gas in CH TRU waste, cannot tolerate or thrive in the presence of free oxygen, a condition guaranteed by the vents. Therefore, there are no reasonably foreseeable operational safety concerns associated with gas accumulation within drums (including during retrieval if that becomes necessary).

TEST PHASE

The Proposed Action includes a Test Phase of approximately 5 years during which experiments would be conducted to monitor and collect data on the rate of gas generation under a variety of conditions. These experiments are described in Appendix O of the final SEIS. The experiments include alcove-scale tests where drums of waste are to be emplaced and bin-scale tests involving the equivalent of six drums of waste per bin. By design, these tests are intended to accumulate gases within sealed alcoves and bins for periodic sampling and analysis. As such, conditions where accumulation of hydrogen and methane gas to levels approaching their lower flammability limits could occur. The majority of the experiments, four out of the five waste-containing alcoves and the preponderance of the bins, would be in anoxic environments, that is, with little or no oxygen present. As such, these anoxic experiments are not of apparent operational safety concern.

Alcove Tests. Calculations of the rate of accumulation of hydrogen and methane in alcove 2, which would simulate the waste storage conditions expected during the 20-year Disposal Phase and has an air atmosphere, indicate that, after 5 years, radiolytic-produced hydrogen and microbial-produced methane could be as high as 0.7 percent and 3.4 percent by volume, respectively. These results are below the lower flammability limits of each gas. Moreover, the alcove would be maintained at a slightly positive pressure to prevent inflow of air. If the design basis leak rate for the alcove, 1 percent of the volume per week, is factored into the calculations, the residual hydrogen and methane concentrations after 5 years would be 0.004 and 0.02 percent by volume, respectively. These calculations also conservatively ignore the depletion of oxygen associated with the gas production. As such, alcove 2 is not considered to be of safety concern.

Bin-Scale Tests. Gas generation calculations for individual oxic bins of test waste have also been made. These results indicate that the radiolytic-produced hydrogen concentration could reach 6.6 percent by volume over the approximate 5-year test period, even when depletion by bin sampling and periodic pressure relief is taken into account. This level exceeds the lower flammability limit for hydrogen, although depletion of oxygen in the process of hydrogen generation may prevent a flammable or detonable mixture from occurring. (As previously indicated, methane generation by anaerobes would not be expected in these bins, while free oxygen still exists. As such, methane accumulation is also not believed to be a significant safety concern.)

As a primary purpose of the Test Phase, the internal hydrogen, methane, oxygen, and other gases within the bins, as well as the alcoves, would be closely monitored. Any approach to a flammable or detonable gas mixture would be quite evident and would be mitigated to prevent the gas concentrations from reaching that level. To minimize possible ignition sources, all bins would also be electrically grounded. Additional available mitigation measures, if deemed necessary, include purging of the atmosphere with inert gas, the capability for which has been designed into the tests. The tests are intended to generate data necessary for the long-term Performance Assessment of the WIPP and to determine operational safety requirements during the Disposal Phase. Although it is important to ensure that these tests are not prematurely terminated, operational safety requirements and limiting conditions for operation during the Test Phase would be established to ensure that safety is not compromised.

DISPOSAL PHASE

The Disposal Phase involves the emplacement of waste containers in rooms mined within the salt formation and backfilling over the containers as emplacement proceeds. Seven rooms are constructed within a waste panel and eight panels are sufficient to dispose of all CH TRU waste proposed to be disposed of for the WIPP. Operational plans intended to minimize the potential for worker exposure require the use of ventilation diversion bulkheads at either end of a filled room during subsequent waste emplacement operations in other rooms within the panel. These bulkheads, while not designed to contain pressure, could isolate a space within which gases released from the waste containers could accumulate.

Conservatively ignoring diffusion of hydrogen and methane from the rooms, but crediting the displacement past the bulkheads of an equal volume of the gas/air mixture from the open space within the room as hydrogen and methane are produced, calculations of the gas concentrations in each of the seven rooms were made as a function of time. At the time the panel is filled, the first room filled within the panel would have the highest concentration of gas, estimated to be 3.4 percent methane and 0.7 percent hydrogen, by volume. The last room filled would have just been isolated and consequently would have no accumulated gases. Both gases are well below their respective lower flammability limits. These results are particularly conservative with reference to the methane percentage since ample free oxygen would still be available to all rooms. Based on these results, accumulation of hydrogen and methane within an active waste panel is not a significant operational safety concern.

Upon filling all rooms within a panel, it is planned to seal each entrance to the panel with massive plugs consisting of a truncated cone-shaped concrete structure "keyed" into the salt (see Figures 6.1 and 6.2). Salt is then used to fill most of the remaining length of the panel access drift, and a second concrete structure is set into place. The use of these 130-foot long panel seals would isolate a significant volume of initially open space within which gases can accumulate. This open space is associated with the average void fraction within WIPP waste containers, considered to be 50 percent void, and the headspace above the backfilled waste, approximately 1.5 feet throughout the entire panel. Assuming the initial concentrations of hydrogen and methane to be the average of the seven rooms at the time the panel was just filled, subsequent gas deneration and pressurization of the panel was calculated as a function of time. These calculations demonstrate that the concentration of hydrogen in the open space would not reach its lower detonability limit during the 20-year Disposal Phase, but the methane concentration could reach and eventually exceed its detonability range over years 4 through 8.

There are several factors which tend to reduce the likelihood of a gaseous detonation. A detonation requires a mixture of oxygen and methane or hydrogen in proper proportions. First, as discussed above, the anerobic generation of methane requires very low concentrations of oxygen and corrosion, which could generate hydrogen, consumes oxygen. Second, closure of the unobstructed free spaces by salt creep and the relative smoothness of the repository walls also reduce the probability of a detonation. Third, compaction of wastes and backfill in response to a pressure pulse within the headspace would also tend to damp out propagation of a pressure pulse. Finally, there are several measures such as purging with inert gas, active ventilation, delay of seal emplacement, and the use of intentional ignitors which could be used, if warranted, to further reduce the likelihood of a detonation.

However, in order to assess the potential consequences of a detonation within the sealed panel, if such a detonation were to occur, the optimal concentration for methane detonation was assumed (bounding or "worst-case" assumption) and the resulting pressures calculated. The ignition was postulated to occur at the farthest point from the seal in order to calculate the maximum possible wave acceleration in the headspace and overpressure at the seal plug. (A detonation within the backfilled waste itself would not proceed as a single event but rather as a series of small detonations and only if sufficient free and open voids existed throughout the waste stack.) It was also conservatively assumed that the reaction transitioned from a deflagration to detonation to maximize the calculated pressure, even though the surfaces of the walls, ceiling, and backfill are considered too smooth (development of a detonation is enhanced by turbulent flow) to allow this transition to occur. The dissipation of the energy of the detonation that would occur through crushing of the backfilled drums below the headspace was also ignored, since such crushing would likely terminate the detonation. A time history of the pressure at the seal plug was developed with an initial resulting impulse load on the exposed concrete face of the seal plug of 800 pounds per square inch (psi) dropping to 120 psi within one-third of 1 second.

A structural evaluation of the impulse loading on the seal plug was conducted by ignoring all but the innermost concrete structure of the seal plug. Because of its size and material of construction, no movement even of this initial component of the plug is predicted, and rapid dissipation of the energy of the detonation would occur within the concrete and surrounding salt. The far face of the innermost concrete structure would see pressures of, at most, several pounds per square inch. Minor cracking within the first several feet of the surrounding salt is possible, as is some spalling of the concrete, but it is unlikely that the event would even be audible to an individual in the main access drift at the far end of the seal. Such an event would also consume the available oxygen within the panel, precluding the possibility of a subsequent detonation. Based upon these results, the accumulation of hydrogen and methane following sealing of a panel is not considered a significant operational safety issue, and release of radioactive material is not reasonably foreseeable. (For further discussion, see Slezak and Lappin, 1990.)

Long Term. As discussed in Subsection 4.3.2.4, fractures of 3 to 15 feet are expected in the Disturbed Rock Zone. Although it is not certain that a gaseous explosion could occur in the repository, the fractures that could occur as a result of gas detonation (1 or 2 feet) would not, therefore, pose additional long-term performance concerns.

REFERENCES FOR APPENDIX F

Baes et al. (Baes, C. F., R. D. Sharp, A. L. Sjoreen, and R. W. Shor), 1984. A Review and Analysis of parameters for Assessing Transport of Environmentally Released Radionuclides Through Agriculture, ORNL-5786, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Baes, C. F., and T. H. Orton, 1979. "Productivity of Agricultural Crops and Forage," in Hoffman, F.O., and C. F. Baes ed., 1979, A Statistical Analysis of Selected Parameters for Predicting Food Chain Transportation and Internal Dose of Radionuclides, ORNL/NUREG/TM-282

Briggs, G. A., 1969. "Plume Rise," AEC Critical Review Series, TID-25075.

DOE (U.S. Department of Energy), 1989. Final Safety Analysis Report, Waste Isolation Pilot Plant (Draft), DOE/WIPP 89-XXX, Carlsbad, New Mexico.

DOE (U.S. Department of Energy), 1985. "Radiation Standards for the Protection of the Public in the Vicinity of DOE Facilities," memorandum from William A. Vaughan, Assistant Secretary for Environment, Safety and Health, U.S. Department of Energy (August 5, 1985), Washington, D.C.

Dunning, D. E., 1986. Estimates of Internal Dose Equipment from Inhalation and Ingestion of Selected Radionuclides, WIPP-DOE-176, Rev. 1, Westinghouse Electric Corporation - Advanced Energy Systems Division, Albuquerque, New Mexico.

Gifford, F. A., 1961. "Use of Routine Meteorological Observations for Estimating Atmospheric Dispersion," Nuclear Safety 2 (4):47-57.

ICRP (International Commission on Radiological Protection), 1979. Limits for Intakes of Radionuclides by Workers, ICRP Publication 30 (ICRP 79a), Pergammon Press, Elmsford, New York.

ICRP (International Commission on Radiological Protection), 1977. Recommendations of the International Commission on Radiological Protection, ICRP Publication 26 (ICRP 77), Pergammon Press, Elmsford, New York.

ICRP (International Commission on Radiological Protection), 1974. Report of the Task Group on Reference Man, ICRP Publication 23 (ICRP 74), Pergammon Press, Elmsford, New York.

Kocher, D. C., 1981. Dose-Rate Conversion Factors for External Exposure to Photons and Electrons, NUREG/CR-1918, ORNL/NUREG-79, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Miller, C. W., 1979. "The Interception Fraction," in Hoffman, F.O. and C.F. Baes eds., A Statistical Analysis of Selected Parameters for Predicting Food Chain Transport and Internal Dose of Radionuclides, Final Report, ORNL/NUREG/TM-282.

Mishima, J., and Schwendiman, L. C., 1973. Fractional Airborne Release of Uranium (Representing Plutonium) During the Burning of Contaminated Wastes, Battelle Pacific Northwest Laboratory, BNWL-1730.

Moore et al. (R. E. Moore C. F. Baes III, L. M. McDowell-Boyer, A. P. Watson, F. O. Hoffman, J. C. Pleasant, and C. W. Miller), 1979. AIRDOS-EPA: A Computerized Methodology for Estimating Environmental Concentrations and Dose to Man for Airborne Release of Radionuclides, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

NRC (Nuclear Regulatory Commission), 1977. Regulatory Guide 1.109, Calculation of Annual Doses of Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix 1, Rev. 1. Office of Standards Development.

Pasquill, F., 1961. "The Estimation of the Dispersion of Windborne Material," Meteorological Magazine, Vol. 90, pp. 33-53.

Rupp et al. (A. F. Rupp, L. P. Bornwasser, and D. F. Johnson), 1948. Dilution of Stack Gases in Cross Winds, USAEC Report AECD-1811, CE-1620, U.S. Atomic Energy Commission.

Slade, D. H., ed., 1968. Meteorology and Atomic Energy - 1968, TID-24190, Atomic Energy Commission.

Slezak and Lappin, 1990. Memo to Darrell [Daryl] Mercer and Craig Fredrickson, DOE/SEIS from Scott Slezak and Al Lappin, Sandia National Laboratories, "Potential for and Possible Impacts of Generation of Flammable and/or Detonable Gas Mixtures During the WIPP Transportation, Test, and Operational Phases," Albuquerque, New Mexico.

TABLE F.2 Frequency of atmospheric stability classes for each direction

	Fraction of time in each stability class						
SECTOR[a]	A	B	C	D	E	F	G
1	0.5740	0.0084	0.0042	0.0391	0.0705	0.0517	0.2521
2	0.3376	0.0084	0.0038	0.0287	0.0738	0.1937	0.3540
3	0.2030	0.0071	0.0034	0.0240	0.0907	0.1979	0.4740
4	0.1869	0.0098	0.0045	0.0548	0.1209	0.1794	0.4437
5	0.2813	0.0246	0.0086	0.1044	0.1597	0.1413	0.2801
6	0.2627	0.0208	0.0091	0.1053	0.1756	0.1144	0.3121
7	0.2320	0.0044	0.0132	0.0485	0.1498	0.1175	0.4347
8	0.2981	0.0154	0.0154	0.0615	0.1231	0.0712	0.4154
9	0.3701	0.0168	0.0037	0.0299	0.1252	0.1121	0.3421
10	0.4469	0.0163	0.0041	0.0265	0.0898	0.0714	0.3449
11	0.5295	0.0153	0.0088	0.0306	0.0722	0.0481	0.2954
12	0.4420	0.0122	0.0020	0.0326	0.0570	0.0855	0.3686
13	0.5465	0.0178	0.0076	0.0293	0.0561	0.0726	0.2701
14	0.5657	0.0046	0.0061	0.0428	0.0413	0.0428	0.2966
15	0.5731	0.0134	0.0134	0.0403	0.0538	0.0336	0.2723
16	0.6558	0.0061	0.0048	0.0400	0.0461	0.0218	0.2255

[a] Wind directions are numbered counterclockwise beginning with 1 for due north.

TABLE F.10 Organ dose correction factors (unitless)[a]

Radio-nuclide	T.body	R.mar.	Lungs	Endost.	S.wall	Lli wall	Thyroid	Liver	Kidneys	Testes	Ovaries
Co-60	.570	.540	.530	.560	.490	.490	.660	.500	.530	.700	.480
Sr-90	0	0	0	0	0	0	0	0	0	0	0
Ru-106	0	0	0	0	0	0	0	0	0	0	0
Sb-125	.539	.511	.502	.582	.461	.451	.631	.467	.489	.678	.447
Cs-137	0	0	0	0	0	0	0	0	0	0	0
Ce-144	.515	.388	.459	.721	.407	.390	.655	.414	.440	.674	.355
Pu-239	.074	.039	.049	.077	.042	.041	.068	.041	.042	.087	.037

[a] From Kocher, 1981.

TABLE F.11 Radionuclide specific parameters 50-year commited dose factors[a]

Sol[b]	Radio-nuclide	Effective	Red marrow	Lungs	Endosteal	Stomach wall	Lli wall	Thyroid	Liver	Kidneys	Testes	Ovaries
Y[c]	Co-60	2.2×10^{-1}	6.4×10^{-2}	1.3	5.0×10^{-2}	1.0×10^{-1}	3.0×10^{-2}	6.0×10^{-2}	1.2×10^{-1}	5.8×10^{-2}	9.9×10^{-3}	1.8×10^{-2}
Y	Sr-90	1.3	1.1	1.1×10	2.5	8.6×10^{-3}	7.6×10^{-2}	8.5×10^{-3}	8.8×10^{-3}	8.5×10^{-3}	8.5×10^{-3}	8.5×10^{-3}
Y	Ru-106	4.8×10^{-1}	5.1×10^{-2}	3.8	5.1×10^{-2}	5.3×10^{-2}	1.4×10^{-1}	5.1×10^{-2}	5.2×10^{-2}	5.2×10^{-2}	5.2×10^{-2}	5.2×10^{-2}
W	Sb-125	1.2×10^{-2}	2.2×10^{-3}	8.0×10^{-2}	1.2×10^{-2}	2.3×10^{-3}	1.2×10^{-2}	1.2×10^{-3}	3.9×10^{-3}	1.2×10^{-3}	9.0×10^{-4}	1.3×10^{-3}
D	Cs-137	3.2×10^{-2}	3.1×10^{-2}	3.2×10^{-2}	3.0×10^{-2}	3.2×10^{-2}	3.3×10^{-2}	2.9×10^{-2}	3.2×10^{-2}	3.2×10^{-2}	3.2×10^{-2}	3.0×10^{-2}
Y	Ce-144	3.8×10^{-1}	9.5×10^{-2}	2.9	1.7×10^{-1}	1.0×10^{-2}	1.3×10^{-1}	6.9×10^{-3}	9.4×10^{-1}	8.2×10^{-3}	6.9×10^{-3}	7.1×10^{-3}
W	Pu-239	5.2×10^{2}	7.3×10^{2}	1.2×10^{3}	9.1×10^{3}	5.6×10^{-3}	1.1×10^{-1}	3.3×10^{-3}	2.0×10^{3}	3.4×10^{-3}	1.2×10^{2}	1.2×10^{2}

[a] Dose factors are presented in rem per microcurie inhaled. (Dunning, 1986).

[b] Solubility class yielding highest effective dose for particle size of 1 micron. All other organ dose factors are those yielding highest dose irrespective of solubility class.

[c] D, W, and Y refer to lung clearance rate in days, weeks, or years.

APPENDIX G

TOXICITY PROFILES, RISK ASSESSMENT METHODOLOGY,
AND MODELS FOR CHEMICAL HAZARDS

TABLE OF CONTENTS

G.1 **INTRODUCTION**

Toxicity profiles are provided to give the reader a brief introduction and understanding of the chemical components and their potential health effects. It is important to remember that any chemical can cause health effects if individuals are exposed to high enough doses.

The profiles are intended to provide information on:

- Physical/Chemical Properties: A description of properties that aid in predicting how the chemical will behave in the environment.

- Fate and Transport: Indicates, where possible, what happens to the chemical within the environment.

- Health Effect: Background information on potential health effects in humans or animals from acute or chronic exposures. In addition, information is provided on various exposure routes (i.e., inhalation, oral, or dermal).

- Effects on Wildlife: Includes a discussion of the toxic effects of the chemicals on aquatic and terrestrial organisms.

- Regulatory Standards and Guidelines: A description of various parameters that have been developed to protect human health and the environment.

The toxicology profiles are intended to be brief overviews of individual chemicals and not extensive reviews. They are, however, intended to include the major health effects (i.e., toxicity) and other aspects of the chemical in question.

Section G.8 includes a description of the air dispersion models used to estimate concentrations of hazardous chemicals released during routine operations and accidents. The various input parameters and assumptions used in the models are provided.

The exposure parameters and methods used to estimate the daily intakes of hazardous chemicals are provided in Section G.9. The methodology and calculations for determining the long-term or short-term risks associated with exposure to carcinogenic and noncarcinogenic chemicals are given in Section G.10.

G.2 **1,1,2-TRICHLORO-1,2,2-TRIFLUOROETHANE**

SUMMARY

1,1,2-Trichloro-1,2,2-trifluoroethane (Freon 113) is a chlorine-containing, non-hydrogenated fluorocarbon. Exposure to high doses can affect the central nervous system, heart, and liver. 1,1,2-Trichloro-1,2,2-trifluoroethane is mainly of environmental concern due to its ability to destroy atmospheric ozone.

IDENTIFICATION

CAS Number: 76-13-1

Chemical Formula: $CCl_2 F CCl F_2$

Synonyms: Halocarbon 113
Refrigerant 113
TTE
Freon 113
FC-113
Fluorocarbon 113

CHEMICAL AND PHYSICAL PROPERTIES

Molecular Weight	197.5
Boiling Point	47.6° C
Melting Point	-35° C
Specific Gravity	1.5635 at 25° C
Vapor Pressure	284 torr at 20° C
Solubility	Insoluble in water, soluble in alcohol, ether, and benzene

FATE AND TRANSPORT

1,1,2-Trichloro-1,2,2-trifluoroethane is highly volatile. It is more likely to reach the stratosphere than hydrogenated fluorocarbons (Clayton and Clayton, 1981). There it photodissociates, producing chlorine atoms which destroy the ozone layer (National Research Council, 1976; Council on Environmental Quality, 1975; National Science Foundation, 1975).

HEALTH EFFECTS

1,1,2-Trichloro-1,2,2-trifluoroethane is a weak narcotic. Impairment of psychomotor abilities (e.g., loss of ability to concentrate, mild lethargy) have been observed in human volunteers (ACGIH, 1986; Stopps and McLaughlin, 1967). The threshold for impairment is approximately 2,500 ppm (Stopps and McLaughlin, 1967). Exposure to massive doses also produces irritation of the respiratory tract and liver cell enlargement (ACGIH, 1986). Slight diffuse degenerative fatty infiltration of the liver has been observed in rats after seven, 19-hour exposures to 5,000 ppm (Kniskern and Pittsman, 1952).

1,1,2-Trichloro-1,2,2-trifluoroethane may cause degreasing of the skin. Frostbite can occur which if not properly attended to, can result in gangrene (Clayton and Clayton, 1981).

1,1,2-Trichloro-1,2,2-trifluoroethane has cardiac sensitization potential (ACGIH, 1986; Gosselin, 1984; Clayton and Clayton, 1981). At concentrations greater than 25,000 ppm, dogs, monkeys, and rats (exposed under various conditions) experienced tachycardia, hypotension or myocardial depression (Aviado, 1975). Abuse of fluorocarbon-containing aerosol products has led to death due to cardiac sensitization to endogenous catecholamines, resulting in ventricular fibrillation (Gosselin, 1984). Doses below maternal toxicity produced no changes in the offspring of pregnant rabbits (both oral and inhalation exposures) (Busey, 1967).

The Ames test shows 1,1,2-Trichloro-1,2,2-trifluoroethane to be nonmutagenic. Carcinogenic studies have not been reported (ACGIH, 1986).

REGULATIONS, STANDARDS, AND GUIDELINES

Human Health

OSHA	TWA	1,000 ppm	(NIOSH, 1985) (29 CFR 1910.1000, Table Z-1) (54 FR 12, 2923-2959, Table Z-1-A)
ACGIH	TLV-TWA	1,000 ppm	
	TLV-STEL	1,250 ppm	

(The ACGIH guidelines should provide a margin of safety in preventing systemic effects and cardiac sensitization [ACGIH, 1986]).

Reference Dose (RfD) 30 mg/kg-day (based on an oral NOAEL of 273 mg/kg-day in humans with psychomotor impairment as the most sensitive end point [IRIS, 1989])

IDLH 34,200 mg/m^3

Aquatic Organisms

No regulations, standards, or guidelines are presently available governing the exposure of aquatic organisms to 1,1,2-Trichloro-1,2,2-trifluoroethane (IRIS, 1989).

G.3 **METHYLENE CHLORIDE**

SUMMARY

Methylene chloride is irritating to the skin, eyes, and mucous membranes. Short-term inhalation produces narcosis, and long-term exposure produces symptoms of neurotoxicity. Methylene chloride causes liver, lung and mammary gland tumors in mice and rats. Hepatotoxicity in experimental animals has been demonstrated.

IDENTIFICATION

CAS Number: 75-09-2

Chemical Formula: CH_2Cl_2

Synonyms: Dichloromethane
Methane dichloride
Methylene bichloride

CHEMICAL AND PHYSICAL PROPERTIES

Molecular Weight	84.93	(ACGIH, 1986)
Specific Gravity	1.3255 at 20° C	(ACGIH, 1986)
Melting Point	-96.7° C	(ACGIH, 1986)
Boiling Point	39.75° C at 76 torr	(ACGIH, 1986)
Vapor Pressure	440 torr at 25° C	(ACGIH, 1986)
Refractive Index	1.4237 (20° C)	(Clayton and Clayton, 1981)
Solubility	2 g/100 ml water at 20° C; soluble in ethanol, ethyl ether, acetone	
Flash Point	none	(Clayton and Clayton, 1981)
Log Octanol/Water Partition Coefficient	1.25	(EPA, 1985a)

FATE AND TRANSPORT

Because of its high vapor pressure, methylene chloride is easily volatilized. However, atmospheric accumulation is not of great concern due to scavenging from the troposphere by hydroxyl radicals (EPA, 1985b). This reaction produces carbon dioxide and small amounts of carbon monoxide and phosgene. Phosgene is hydrolyzed to hydrochloric acid and carbon dioxide (EPA, 1979). Methylene chloride has moderate water solubility; therefore, rain washout from the atmosphere may be important in the fate process (ATSDR, 1987). Absorption to soil is unlikely (Dilling et al., 1975). Biodegradation occurs aerobically and anaerobically (EPA, 1985, cited in ATSDR, 1987). Bioaccumulation and bioconcentration are not important fate processes (Hansch and Leo, 1979).

HEALTH EFFECTS

Methylene chloride is a skin, eye, and mucous membrane irritant (ACGIH, 1986). Upper respiratory tract irritation occurs at levels of approximately 100 ppm in humans (Welch, 1987). Short term inhalation (300-800 ppm) leads to decreases in auditory functions and impairment of various psychomotor tasks (Stewart et al., 1972). Longer term inhalation causes neurotoxicity, including headache, dizziness, nausea, memory loss, paresthesia, tingling in the hands and feet, and narcosis (Welch, 1987). These central nervous system effects could be partially due to the metabolism of methylene chloride producing carboxyhemoglobin (Cherry et al., 1983). Burns may result if methylene chloride liquid is placed on the skin (Welch, 1987).

Animal studies have shown methylene chloride to be hepatotoxic. Fatty infiltration of the liver was evident in guinea pigs exposed to 5200 ppm for 6 hours (Morris et al., 1979). Weinstein and Diamond (1972) observed transient fatty changes in the livers of mice exposed to 5,000 ppm for 7 days. A significant increase in liver cytochrome p-450 was induced in rats exposed by inhalation to 500 ppm (Norpoth et al., 1974). Longer term exposures by inhalation cause centrilobular fat accumulation in livers of mice (Weinstein and Diamond, 1972). Haun et al. (1972) showed cytoplasmic vacuolation and the presence of fat droplets in the livers of rats and dogs. An increased incidence of hemosiderosis, cytomegaly, and cytoplasmic vacuolization has been reported in livers of rats and mice exposed to methylene chloride by inhalation (NTP, 1986a). Only very high exposure concentrations will cause serious liver effects in humans (ATSDR, 1987). Cardiac sensitization is seen in animals, but only when given epinephrine (Clark and Tinston, 1973).

Methylene chloride causes cancer in laboratory animals. Increases in liver tumors (NTP, 1986a), lung tumors (NTP, 1986a) and mammary gland tumors (Burek et al., 1980 and 1984; Nitschke et al., 1982; NTP, 1986a) have been reported in rats and mice. There have been no statistically different tumor occurrences between exposed and non-exposed humans (ATSDR, 1987). It has been concluded that methylene chloride is weakly mutagenic in mammalian systems (EPA, 1987a,b). There is no evidence that methylene chloride is a teratogen (an agent causing fetal malformities) (Clayton and Clayton, 1981).

EFFECTS ON WILDLIFE

In the flow-through and static method tests, fathead minnows (*Pimephales promelas*) exposed to methylene chloride experienced loss of equilibrium, melanization, narcosis, and swollen, hemorrhaging gills. These effects are reversible, caused by short exposures to sublethal levels (Alexander et al., 1978). At concentrations of 100 μg/L, developmental stages of some amphibian species may be affected. Concentrations of
1 mg/L and above may cause substantial reproductive impairment in amphibians (Black et al., 1982). High concentrations (approximately 21 percent) of methylene chloride reduces photosynthesis in alfalfa seedlings by 82 percent (Lehmann and Paech, 1972).

REGULATIONS, STANDARDS, AND GUIDELINES

Human Health

OSHA TWA	500 ppm	
	1,000 ppm ceiling	(NIOSH, 1985)
		(54 FR 12, 2923-2959, Table Z-1-A)
		(29 CFR 1910.1000, Table Z-2)
ACGIH TLV-TWA	100 ppm	
TLV-STEL	500 ppm	(NIOSH, 1985)
NIOSH TLV-TWA	75 ppm	
	500 ppm 15 min ceiling (NIOSH, 1985)	

IARC Group 3 - indefinite animal carcinogen

EPA Group B2 - probable human carcinogen

Reference Dose (RfD) 0.05 mg/kg-day (ATSDR, 1987) [based on a NOAEL in rats of 5 mg/kg-day with hepatic histological changes as the most sensitive endpoint]

IDLH 17,500 mg/m^3

Drinking Water Equivalent Level (DWEL) 1.75 mg/L (IRIS, 1989)

Aquatic Organisms

Lowest Effect Concentration (LEC) (IRIS, 1989)

Freshwater:

- Acute LEC 1.1 x 10^4 μg/L
- Chronic Toxicity No data

Saltwater:

- Acute LEC 1.2 x 10^4 μg/L
- Chronic Toxicity 6.4 x 10^3 μg/L

G.4 **1,1,1-TRICHLOROETHANE**

SUMMARY

Exposure to 1,1,1-trichloroethane can cause central nervous system depression, as well as damage to the cardiovascular system, lungs, liver, and kidneys. It is an irritant to the eyes, skin, and mucous membranes. The primary process of elimination of 1,1,1-trichloroethane from the environment is through photo-oxidation in the atmosphere. Neither IARC nor the EPA classify 1,1,1-trichloroethane as a carcinogen. However, some of the animal studies are inconclusive.

IDENTIFICATION

CAS Number: 71-55-6

Chemical Formula: $C_2H_3Cl_3$

IUPAC Name: 1,1,1-trichloroethane

Synonyms: Methyl chloroform
1,1,1-TCA
Chlorothene

CHEMICAL AND PHYSICAL PROPERTIES

Molecular Weight	133.40
Specific Gravity	1.3376 at 20° C (Sax, 1984)
Melting Point	-32.5° C
Boiling Point	74.1° C
Vapor Pressure	100 mm at 20° C, 155 mm at 30° C
Vapor Density	4.63 (Verschueren, 1983)
Solubility	44 mg/L in water at 25° C; soluble in acetone, benzene, carbon tetrachloride, ether, methanol (Sax, 1984)

FATE AND TRANSPORT

The most important route of elimination of 1,1-1-trichloroethane (1,1,1-TCA) from the environment is by reaction with hydroxyl radicals in the atmosphere (photo-oxidation). 1,1,1-TCA is eliminated from surface water primarily through volatilization. It is able to adsorb onto organic matter in the sediment; however, this is probably not a major route of elimination from surface water. 1,1,1-TCA readily migrates from soil to groundwater, the rate of transport through soil depending on the soil composition (EPA, 1987c).

HEALTH EFFECTS

A number of toxic effects of 1,1,1-TCA have been observed. The most common of these is central nervous system depression manifested by dizziness, incoordination, and impaired judgment at low concentrations, and anesthesia and death at high concentrations. Cardiovascular effects such as cardiac arrest ventricular fibrillation, and sensitization to epinephrine-induced arrhythmias have been observed, as well as damage to the lungs, liver, and kidneys. 1,1,1-TCA is moderately irritating to the skin and mucous membranes and is a severe eye irritant in rabbits (Sax, 1984). The oral LD_{50} value for 1,1,1-TCA in rats in approximately 11,300 mg/kg (IRIS, 1989), and the inhalation LC_{50} in rats is approximately 14,000 mg/kg for a 7 hour exposure (Verschueren, 1983).

1,1,1-TCA has been found to be mutagenic in *Salmonella typhimurium* (Farber, 1977) and to cause transformation in rat embryo cells (Price et al., 1978). Although several carcinogenic studies have been performed, the doses administered resulted in direct mortality or no significant increase in tumor formation (IRIS, 1989).

EFFECTS ON WILDLIFE

1,1,1-Trichloroethane has a relatively low acute toxicity in aquatic species, with LC_{50} values ranging from 52.8 ppm for the most sensitive species to 113.0 ppm for the least sensitive species reported (Verschueren, 1983). In aquatic species, 1,1,1-TCA has an elimination half-life of 2 days and a bioconcentration factor of nine, and therefore has only a slight bioaccumulation potential (EPA, 1984).

At this time, there are no data available on the chronic toxicity of 1,1,1-TCA in aquatic species (IRIS, 1989).

REGULATIONS, STANDARDS, AND GUIDELINES

Human Health

OSHA TWA 350 ppm (NIOSH, 1985)
(29 CFR 1910.1000, Table Z-1)
(54 FR 12, 2923-2959, Table Z-1-A)

ACGIH TLV-TWA 350 ppm
TLV-STEL 450 ppm (ACGIH, 1986)

NIOSH 350 ppm-15 min ceiling (NIOSH, 1985)

IARC Group 3 indefinite animal carcinogen (IARC, 1987)

EPA Group D not classifiable as to human carcinogenicity (IRIS, 1989)

Reference Dose (RfD) 0.09 mg/kg-day (based on a NOAEL in guinea pigs of 90 mg/kg-day with hepatic histological changes as the most sensitive endpoint (Torkelson et al., 1958)

IDLH 5,429 mg/m^3

MCL	200 mg/L (IRIS, 1989)	
Ambient Water Quality Criterion	water and fish consumption	218.4 mg/L
	fish consumption only	1,030 mg/L
	(IRIS, 1989)	

Aquatic Organisms

The EPA (1988a) has reported the following lowest effect levels (LECs) for 1,1,1-trichloroethane on aquatic organisms:

Freshwater:

- Acute Toxicity 18.0 mg/L
- Chronic Toxicity No data

Saltwater:

- Acute Toxicity 31.2 mg/L
- Chronic Toxicity No data

These data, however are not adequate for establishing water quality criteria.

G.5 **CARBON TETRACHLORIDE**

SUMMARY

Carbon tetrachloride is a suspect human carcinogen and is known to produce liver tumors in laboratory animals. Carbon tetrachloride causes liver damage, as well as damage to the kidneys, skin, and eyes. Carbon tetrachloride may be readily absorbed through the skin. Acute exposure to carbon tetrachloride by inhalation, ingestion, or skin absorption may cause central nervous system depression.

IDENTIFICATION

CAS Number: 56-23-5

Chemical Formula: CCl_4

IUPAC Name: Tetrachloromethane

Synonyms: Tetrachloromethane
Perchloromethane

CHEMICAL AND PHYSICAL PROPERTIES

Molecular Weight	153.84
Specific Gravity	1.589
Melting Point	-23° C
Boiling Point	76.7° C
Vapor Pressure	90 mm at 20° C
Vapor Density	5.32
Solubility	800 mg/L H_2O at 20° C; miscible in alcohol, benzene, chloroform, ether, carbon disulfide (Merck, 1983; Sax, 1984; Verschueren, 1983)

FATE AND TRANSPORT

Volatilization is the major transport process for the removal of carbon tetrachloride from aquatic systems (EPA, 1979). Carbon tetrachloride in the troposphere degrades very slowly by reaction with hydroxyl radicals (EPA, 1979; EPA, 1987c). It can diffuse to the stratosphere where it is degraded by exposure to higher energy ultraviolet light to form CCl_3 radicals, chlorine atoms, and phosgene. This photolysis reaction is thought to be the predominant environmental fate process for carbon tetrachloride. There is no clear evidence of selective concentration (adsorption) of carbon tetrachloride in soils or sediments (EPA, 1979). Carbon tetrachloride migrates readily to groundwater and may be expected to remain there for months to years (EPA, 1987c).

HEALTH EFFECTS

Carbon tetrachloride is a suspect human carcinogen and is known to be carcinogenic in rats, mice, and hamsters (IARC, 1979). Liver tumors are most commonly seen, but adrenal tumors have also been observed (Weisburger, 1977). Data on mutagenicity, teratogenicity, and reproductive effects have been inconclusive (Verschueren, 1983).

Carbon tetrachloride has been shown to produce nonmalignant liver damage as well as kidney damage in animals and humans (ACGIH, 1986). Symptoms of liver dysfunction may include nausea, anorexia, vomiting, stomach-ache, and jaundice, but dysfunction may also be asymptomatic (ACGIH, 1986).

Central nervous system depression has been experienced in cases of acute and chronic exposure to carbon tetrachloride. Atmospheric levels of 45-97 ppm have reportedly produced dizziness and headaches (Kazantzis, 1960).

Substances such as barbiturates and chlorinated biphenyls have been shown to enhance the effects of carbon tetrachloride. Consumption of alcoholic beverages is known to markedly increase the toxicity of carbon tetrachloride (Maling, 1975; Cornish, 1973; Carlson, 1975).

Carbon tetrachloride is a skin and eye irritant due to its defatting action and appreciable blood levels of carbon tetrachloride have been reported due to skin absorption.

EFFECTS ON WILDLIFE

Carbon tetrachloride has been shown to be toxic to freshwater aquatic life at acute exposure concentrations as low as 35,200 μg/liter, and as low as 50,000 μg/liter in saltwater aquatic species (EPA, 1980). LC_{50} values ranging from 67 ppm to 150 ppm have been reported for subacute exposures to carbon tetrachloride in aquatic species (Verschueren, 1983). A bioconcentration factor of 19 has been reported for carbon tetrachloride in fish (EPA, 1986).

No data on the chronic toxicity of carbon tetrachloride in aquatic species, or its effects on terrestrial wildlife are available at this time.

REGULATIONS, STANDARDS, AND GUIDELINES

Human Health

OSHA	TWA	10 ppm 25 ppm ceiling	(29 CFR 1910.1000, Table Z-2) (54 FR 12, 2923-2959, Table Z-1-A)
ACGIH	TLV-TWA	5 ppm	(ACGIH, 1986)
NIOSH	TWA	10 ppm 25 ppm ceiling	(NIOSH, 1985)
IARC	Group 2B	Sufficient evidence of animal carcinogenicity (IARC, 1987)	

Inadequate evidence of human carcinogenicity (IARC, 1987)

EPA Group B2 Probable human carcinogen (IRIS, 1989)

Reference Dose (RfD) 0.0007 mg/kg-day (IRIS, 1989)

IDLH 1,800 mg/m^3

Drinking Water
Equivalent Level
(DWEL) 25 μg/L (IRIS, 1989)

10^{-6} Cancer Risk 0.3 μg/L (IRIS, 1989)

Aquatic Organisms

There are inadequate data for establishing ambient water quality criteria for aquatic organisms (EPA, 1980).

G.6 **LEAD**

SUMMARY

Inorganic lead is found in the earth's crust at about 15 ppm. Lead isotopes are the stable products of decay of three natural radioactive elements: from the uranium series-^{206}Pb, from the thorium series-^{208}Pb, and from the actinium series-^{207}Pb. Lead forms two series of compounds corresponding to the oxidation states of +2 and +4, the most common being +2 (Kirk-Othmer, 1985).

Lead and its compounds are persistent in the environment and are cumulative poisons. High doses of lead result in damage to the central nervous system and loss of kidney function (Sittig, 1979).

IDENTIFICATION

CAS Number: 7439-92-1 (lead as inorganic fumes and dust)

Chemical Symbol: Pb

Synonyms: C.I. 77575
Lead Flake
Lead s2

CHEMICAL AND PHYSICAL PROPERTIES

Molecular Weight	207.19
Boiling point	1740° C
Melting point	327.5° C
Specific gravity	11.35
Vapor pressure	1 mm @ 973° C
Solubility	Slightly soluble in H_2O in presence of nitrates, ammonium salts, and CO_2

FATE AND TRANSPORT

Lead exhibits the +2 oxidation state in aqueous systems. Natural compounds of lead are not usually mobile in ground or surface water since it tends to combine with carbonate or sulfate ions to form insoluble compounds under oxidizing conditions and forms extremely insoluble lead sulfide under reducing conditions.

Sorption processes are effective in reducing the concentration of soluble lead in natural waters and result in enrichment of bed sediments near the source. The tendency for lead to form complexes with naturally occurring organic material increases its adsorptive affinity for clays and other mineral surfaces. Removal of lead by sorption and precipitation occurs more rapidly in alkaline waters; therefore, lead is considerably more mobile in acidic waters.

Benthic microbes can methylate lead to form tetramethyl lead which is volatile and more toxic than inorganic lead. Biomethylation may provide a mechanism for remobilization of lead in the bed sediments. Bioaccumulation of weakly sorbed lead phases also may result in remobilization (EPA, 1979).

HEALTH EFFECTS

Lead enters the body through inhalation and ingestion, is absorbed into the circulatory system from the lungs and digestive tract, and is excreted via the urine and feces. About 90 percent of the ingested lead passes through the gastrointestinal tract unabsorbed. About 10 percent of the ingested lead is absorbed by the body and a portion is excreted in urine with lesser amounts in sweat, hair, and nails. Under conditions of approximately steady state, more than 90 percent of absorbed lead in the body is in the skeleton, where it remains in a relatively inert state (Lee, 1972).

Particle size and chemical composition affect the readiness with which lead is absorbed from the lungs and digestive tract. Small particles and highly soluble compounds are more readily absorbed, hence more hazardous, than larger particles and compounds with lower solubility.

Lead and its compounds are cumulative poisons. The most common signs of lead exposure are gastrointestinal: anorexia, nausea, vomiting, diarrhea, and constipation followed by colic. Lead can also affect hemoglobin synthesis and red blood cell survival as well as the central and peripheral nervous systems (Kirk-Othmer, 1985).

An early effect of lead on the kidney is the development of intranuclear inclusion bodies in the renal tubular lining. With continued exposure, swelling and mitochondrial changes occur in proximal tubular lining cells (Ratcliffe, 1981).

Epidemiological investigations on exposed population groups and experiments on rats have shown that the placenta does not represent an important barrier to lead. Experiments on female mice have shown that ingested lead may cause, depending on the dose, a reduction of pregnancies, a decrease in embryo weight, or abortion. At high doses of lead and low calcium diet chromosomal abnormalities were found in primates (DiFerrante, 1979).

REGULATIONS, STANDARDS, AND GUIDELINES

Human Health

OSHA TWA	0.05 mg/m^3 (Hazline, 1989)
ACGIH TLV-TWA	0.15 mg/m^3 (ACGIH, 1986)
EPA Acceptable Intake	1.40 E-03 mg/kg-day (Oral Route) (EPA, 1986) 4.30 E-04 mg/kg-day (Inhalation Route) (EPA, 1986)
Safe Drinking	0.05 mg/L (MCL) (IRIS, 1989)

Water Act (ARAAR)	0.02 mg/L (Proposed MCLG) (IRIS, 1989)
EPA Ambient Water Criteria	50 μg/L (Aquatic Organisms & Drinking Water) (IRIS, 1989)
Clean Air Act (ARAAR)	1.5 (90-day) ($\mu g/m^3$) (IRIS, 1989)

Aquatic Organisms

Bioconcentration factors for aquatic organisms range from 60 in marine and freshwater fish to 200 in marine and freshwater plants and invertebrates (EPA, 1979). Decreasing pH increases the availability of divalent lead, the principal form accumulated by aquatic animals.

G.7 **TRICHLOROETHYLENE**

SUMMARY

Trichloroethylene is an industrial solvent that induces central nervous system depression, adversely affects the liver, kidneys, and hematological systems, and sensitizes the heart to endogenous catecholamines. Lung, liver, and kidney tumors are increased significantly in exposed rats and mice.

IDENTIFICATION

CAS Number: 79-01-6

Chemical Formula: C_2HCl_3

Synonyms: Trichloroethene
TCE
Ethylene trichloride

CHEMICAL AND PHYSICAL PROPERTIES

Molecular Weight	131.4
Specific Gravity	1.45560 at 25/4° C
Melting Point	-86.8° C
Boiling Point	87.0° C
Vapor Pressure	77 torr at 25° C
Refractive Index	1.4777 at 20° C
Solubility	0.1 g/100 ml water at 20° C; soluble in ethanol and ethyl ether (Clayton and Clayton, 1981)

FATE AND TRANSPORT

Trichloroethylene vaporizes easily. However, accumulation in the atmosphere is not of critical importance. Hydroxyl radicals react with trichloroethylene to form hydrochloric acid, carbon monoxide, carbon dioxide, and carboxylic acid (EPA, 1979). The atmospheric half-life of trichloroethylene is approximately 7 days (ATSDR, 1988b). It is highly mobile in the soil, and subject to significant leaching (HSDB, 1987; EPA, 1979). Since biodegradation under aerobic conditions is slow, trichloroethylene is relatively persistent in subsurface soils and groundwater (Barrio-Lage et al., 1987; Hallen et al., 1986; Wilson et al., 1986; Fogel et al., 1986; Vogel and McCarty, 1985).

HEALTH EFFECTS

Inhalation of trichloroethylene causes irritation of the eyes and upper respiratory tract, as well as central nervous system (CNS) depression (Hazline, 1989; Sittig, 1985; Gosselin, 1984; Clayton and Clayton, 1981, Nomiyama and Nomiyama, 1977). CNS effects experienced include visual disturbances, mental confusion, fatigue (Clayton and Clayton, 1981), and at sufficiently high concentrations, euphoria, analgesia, and anesthesia (ACGIH, 1986).

Sensitization of the heart has occurred in humans at anesthetic levels (Clayton and Clayton, 1981). This has also been observed in dogs (Reinhardt et al., 1973) and rabbits (White and Carlson, 1979, 1981). If trichloroethylene is left in contact with the skin, defatting and fissuring, followed by erythema (redness) may result (Clayton and Clayton, 1981).

Trichloroethylene is hepatotoxic and nephrotoxic in experimental animals. Mice exposed to 37 ppm for 30 days had increased liver weights and vacuolated hepatocytes (Kjellstrand et al., 1983, 1981). Rats also showed an increase in liver weights when subjected to 55 ppm, 8 hr/day, 5 days/week for 14 weeks (Kimmerle and Eben, 1973) and >50 ppm continuously for 12 weeks (Nomiyama et al., 1986). Other unspecified treatment-related hepatic effects were also noted by Nomiyama et al. (1986).

Renal effects include increased kidney weights and dysfunction in rats exposed to >150 ppm for 12 weeks (Nomiyama et al., 1986) and renal tubular meganucleocytosis in rats exposed to >300 ppm for 7 hr/day, 5 days/week for 104 weeks (Maltoni et al., 1986).

Hematological system alterations have been experienced by experimental animals subjected to trichloroethylene. Rats treated for 10 days with >50 ppm had concentration-dependent inhibition of delta-aminolevulinate (ALA) dehydrogenase activity in the liver and bone marrow cells, increased ALA synthetase, decreased heme saturation of tryptophan pyrrolase, and decreased cytochrome p-450 in the liver (Fujita et al., 1984). Nomiyama et al. (1986) exposed rats to >50 ppm for 12 weeks and observed dose-related changes in hemoglobin, hematocrit, and erythroblast count. Myleotoxic anemia has been shown in rabbits (Mazza and Brancaccio, 1967).

Teratogenicity data for trichloroethylene in humans are inconclusive. In rat pups, skeletal ossification anomalies were produced by dams subjected to >100 ppm for 4 hr/day on days 8 and 21 of gestation (Healy et al., 1982).

Trichloroethylene is weakly mutagenic in some microbial test systems (Clayton and Clayton, 1981). These data are inadequate to assess human carcinogenicity caused by trichloroethylene (ATSDR, 1988b). However, trichloroethylene has been proven to be carcinogenic in rats and mice, causing renal adenomas and carcinomas, lung adenomas, and hepatomas (Maltoni et al., 1986; NTP, 1986b, 1982; Fukuda et al., 1983; Bell et al., 1978).

EFFECTS ON WILDLIFE

The availability of data pertaining to the toxic effects of trichloroethylene on wildlife is limited. Chlorophyll-containing algae and plants exposed to trichloroethylene lose their color at 600 mg/L (Verschueren, 1983). LC_{50} values of approximately 50 mg/L were noted for three freshwater species tested.

The EPA has reported lowest effect levels (LECs) for acute exposure to trichloroethylene of 45 mg/L, and 2 mg/L for freshwater and saltwater organisms, respectively. No LECs for chronic exposures have been reported.

REGULATIONS, STANDARDS, AND GUIDELINES

Human Health

OSHA	TWA	100 ppm 200 ppm ceiling(29 CFR 1910.1000, Table Z-2) (54 FR 12, 2923-2959, Table Z-1-A)
ACGIH	TLV-TWA TLV-STEL	50 ppm 200 ppm (ACGIH, 1986)
NIOSH	TWA	25 ppm (10 hr) (Hazline, 1989)
IDLH	5,400 mg/m^3	
EPA	Group B2	probable human carcinogen (IRIS, 1989)
IARC	Group 3	indefinite animal carcinogen (IARC, 1987)

Aquatic Organisms

There are inadequate data for establishing ambient water quality criteria for aquatic organisms.

G.8 **AIR DISPERSION MODEL**

The U.S. Environmental Protection Agency's (EPA's) Users Network for Applied Modeling of Air Pollution (UNAMAP) 6 version of the Industrial Source Complex Model (ISC) (EPA, 1988b) was used to estimate ambient concentrations of materials released from emission sources (stacks) at the WIPP site. Releases resulting from routine operations (long-term) and postulated on-site accident events (short-term), aboveground and underground, were modeled.

LONG-TERM MODEL

The long-term version of the model (ISCLT) projected the annual average aboveground concentrations, based on the annual average of meteorological data recorded at Carlsbad during the 5-year period 1950 to 1954. Input mixing heights and ambient temperatures were obtained from Holzworth (1972) and National Weather Service records, respectively. The model was run in the "regulatory default" mode. For convenience, the emission rate was assumed to be 10 grams per second (5 gms/sec from each stack). To determine the ambient concentrations resulting from a different emission rate, a ratio of the actual and assumed emission rates was taken and applied to the predicted ambient concentrations.

The receptor field consisted of a rectangular grid extending 50,000 meters north, east, south, and west from the originating point of the emission. The physical location of the point of origin was the centerline of the vertical ventilation exhaust duct.

Ambient concentrations for underground workers were estimated manually using the following assumptions:

- Waste disposal room dimensions are 10 meters by 91 meters by 4 meters
- Air velocity is 0.4 m/sec
- Air flow is parallel to the long axis of the chamber
- Hazardous chemicals are uniformly mixed in the air stream.

Using these assumptions, ambient concentration ($\mu g/m^3$) is the quotient of the release rate ($\mu g/sec$) and the ventilation volumetric flow rate (m^3/sec).

SHORT-TERM MODEL

Short-term concentrations were estimated by the ISC model (ISCST) running in the short-term mode. Short-term releases were assumed to be discharged from the emergency filtration system with a flow rate of 60,000 cfm to a single stack. Generic meteorological data (48 combinations of wind speed and stability customarily used in UNAMAP screening models) were used in the ISCST model. For each of the 48 combinations, a 1-hour duration was arbitrarily assigned. In all instances, wind was assumed to be blowing from the south.

Hypothetical exposed individuals were located due north of the stack at distances out to 50,000 meters.

The assumptions inherent in generic meteorological data are:

- The 48 combinations cover the entire spectrum of meteorological conditions that could be obtained.

- Each of the 48 combinations can occur at some time or other.

Thus, the highest potential short-term exposure can be identified as well as the distance at which this exposure occurs. This is a health-protective approach, since there is a low probability of all the necessary conditions occurring simultaneously. The emission rate was again assumed to be 10 grams per second. To determine the ambient concentrations resulting from a different emission rate, a ratio of the actual and assumed emission rates was taken and applied to the predicted ambient concentrations.

Manual calculations were used to estimate ambient air concentrations of hazardous chemicals affecting workers in the waste handling building and underground during postulated on-site accidents. Some accident-specific assumptions are described in Appendix F. For each accident event, it was assumed that the total release was equal to the total mass of volatile organics in the void volume of breached containers. Concentrations in the air in the vicinity of each accident were estimated using these assumptions.

For the aboveground accidents, the release was assumed to disperse into a hemisphere which expands at a given rate for a given time based on the air flow into the waste handling building. Concentrations of organics within the volume of the hemisphere were assumed to be uniform. For underground accidents, a similar procedure was followed. However, the underground release was assumed to disperse into an underground mined out area which was 4.0 meters x 3.4 meters x 6.0 meters in dimension. Again, concentration within this volume was assumed to be uniform. Estimations of particulate releases of lead during a single drum fire underground were made based on the vapor pressure of elemental lead. The ISCST model was used to predict the maximum aboveground air concentration on-site.

G.9 **EXPOSURE ASSESSMENT**

Consistent with the health-protective approach to risk assessment, potential exposures to releases of hazardous chemicals resulting from routine operations are estimated for hypothetical workers located at the points of maximum on-site concentrations above and below ground identified by the air dispersion modeling. Estimates of potential exposures were also made for a hypothetical resident located at the point of maximum concentration at the WIPP site boundary.

EXPOSURE PARAMETERS

The potential exposed individual was assumed in each case modeled to weigh 70 kg (about 154 lbs). Adults are used as the model residential receptor since no actual individual exists at the site boundary. In fact, the actual resident nearest to the facility is more than 3 miles from the boundary. The increased sensitivity of the elderly or very young individual from considerations such as body weight is mitigated by the additional dilution of the already very low predicted concentrations at the site boundary (see Section 5.0).

The daily respiratory volume was assumed to be 20 cubic meters (m^3) for a 24-hour period (residential exposures) (EPA, 1986) and 12 m^3 for an 8-hour period (occupational exposures) (EPA, 1985c). Due to a lack of chemical-specific data for volatile organics, a transfer coefficient of 1.00 was used to model uptake and absorption via the lungs for these chemicals.

The rate of lead deposition in the lungs was assumed to range from approximately 30 to 50 percent of particulates inhaled, while up to 70 percent of deposited lead was assumed to be absorbed within 10 hours of exposure (ATSDR, 1988a). To maintain a health-protective approach, a transfer coefficient of 0.35 (i.e., 70 percent x 50 percent) was used to represent deposition and absorption in the exposure estimates for lead.

Potential exposures from the inhalation of hazardous chemicals during routine operations are estimated for occupational and hypothetical residential individuals during above- and belowground operations. The concentrations of hazardous chemicals in air that are predicted at each exposed individual location are evaluated to determine if, based on the postulated scenario, the concentrations will remain constant or increase with time during the exposure period. The aboveground worker and hypothetical residential individual are continually exposed to 42-drum units from the waste handling building during the Test Phase and the Disposal Phase and 6,000-drum units from underground emissions during the Disposal Phase. Similarly, the underground worker is continually exposed to 6,000-drum units during the Test Phase and the Disposal Phase.

The concentration of hazardous chemicals in air from underground operations does not remain constant during the Test Phase because the rooms will not be backfilled and sealed. During the Test Phase, the number of drums increases by 17,600 drums, or 1-drum unit, per year. The concentration of hazardous chemicals in air at the aboveground worker and the residential individual are averaged over the 5-year period by multiplying the predicted air concentration by a weighting factor. A weighting factor (WF) of three was calculated using the following equation:

$$WF = (U_1 + U_2 + U_i ... U_n) / n, \; n = 5$$

where:

U_i = number of drum units present per year, i = 1,...n

This method conservatively assumes that the drums will be emitting volatile organic compounds over the entire 5-year period. Based on calculations of the emission period, this is unlikely. For example, the entire mass of methylene chloride would be emitted in 2 years if it continuously diffused through the carbon composite filter at the calculated emission rate provided in Subsection 5.2.4.2, Table 5.35. Therefore, an additional measure of conservatism is added by assuming the organics are emitted over the entire 5-year period.

Concentrations available to individuals potentially exposed as a result of accident events were based on the total void volume gas concentrations and short-term modeling employing the specific dispersion characteristics of a given accident area.

ESTIMATION OF DAILY INTAKES OF HAZARDOUS CHEMICALS

The TLV-based, or IDLH-based estimated intakes (I_{ai}) for the accident scenarios are estimated by the following formula:

$$I_{ai} = (C_i)(V)(A_i)(E)(f_a)$$

where:

I_{ai} = TLV- or IDLH-based estimated intake (mg/exposure),

C_i = concentration of constituent in air at the receptor location (mg/m^3),

V = respiratory volume (m^3/day),

A_i = transfer coefficient for ith chemical,

E = seconds or minutes per exposure,

f_a = conversion factor.

The respiratory volume of 20 m^3/day and transfer coefficients of 0.35 for lead and 1.0 for all volatile organic compounds are used in the upper-bound transportation accident to estimate intake of a hypothetical exposed individual located 50 meters from the accident. An exposure of 30 minutes is postulated during the accident. The conversion factor is 1 day per 1,440 minutes.

The estimated intakes for the accident scenarios postulated to occur during operations at the WIPP are also calculated using the above equation. Because the exposure to a worker is estimated, a respiratory volume of 12 m^3/workday is used in the calculation of intake. The transfer coefficients of 0.35 for lead and 1.0 for volatile organic compounds were utilized as above. Each exposure period in minutes was then converted, using the factors of 1 hour per 60 minutes and 1 workday per 8 hours. Based on the air modeling, the exposure period for workers during accidents in the waste handling building is 1 minute and in the underground is 15 seconds. A conservative 30-minute exposure period is assumed during the underground fire scenario at the WIPP. For the defined time period of each accident, the concentration of chemicals in air at the location of the worker is assumed to be constant.

For routine operation, the annualized averages for each chemical for both the Test Phase and the permanent Disposal Phase were used to estimate the chemical-specific daily intakes for the residential, aboveground occupational, and underground occupational receptors. The daily intake was estimated by

$$I_{ri} = (C_i)(V)(A_i) / (f)(W), \ i = 1, ..., 6, \qquad \text{(G-2)}$$

where:

I_{ri} = estimated daily intake of the i^{th} chemical (mg/kg-day), i = 1,...6,

C_i = concentration of the i^{th} chemical (μg/m^3),

V = scenario-specific respiratory volume (m^3/day),

A_i = transfer coefficient for the i^{th} chemical, i = 1,...6,

f = conversion factor (1,000 μg/mg),

W = body weight (kg).

G.10 **RISK ESTIMATION**

While the estimation of human health risks for this assessment employed a quantitative evaluation of the data available on waste characterization, these estimates are more meaningful when viewed in a relative, and therefore more qualitative sense. The precision of these estimates was limited by the uncertainties associated with the size and quality of the data base. In this assessment, these limitations were partially mitigated by defining a range of extremes. However, overriding uncertainties still persist. An analysis of these uncertainties is given in Section 5.0.

LONG-TERM RISK ESTIMATION FOR CARCINOGENS: ROUTINE OPERATIONS

For any Class A or B carcinogen (by the classification of the EPA's Carcinogenic Advisory Group) that is projected to average greater than 1 percent by weight of the waste, predicted air pathway exposures that may result from emissions associated with routine facility operations are compared to unit cancer risks (EPA, 1986). Excess incremental lifetime cancer risks resulting from inhalation of vapors are estimated for the exposed individuals associated with each scenario. These estimates are based on guidance provided by the SPHEM and the Air Toxics Assessment Manual (California Air Pollution Control Officers Association [CAPCOA], 1987).

Of the representative chemicals for the waste, there are three volatile organics that are Class A or B carcinogens: carbon tetrachloride, methylene chloride (dichloromethane) and trichloroethylene (TCE). The estimated daily intakes for these chemicals were used to estimate the risk of the occurrence of one excess case of cancer as a result of the estimated exposures to these chemicals. This lifetime incremental excess cancer risk is given by

$$R_i = q_1^* I_{ri} LC\ ,\ i = 1,...,3, \qquad \text{(G-3)}$$

where:

R_i = excess incremental lifetime cancer risk for the i^{th} chemical, i = 1,...,3,

q_1^* = chemical-specific cancer potency factor $(mg/kg\text{-}day)^{-1}$,

I_{ri} = estimated daily intake of the i^{th} chemical for a given individual (mg/kg-day), i =1,...,3,

LC = lifetime correction factor.

The cancer potency factors used for carbon tetrachloride, methylene chloride, and trichloroethylene were 1.36×10^{-1}, 1.40×10^{-2}, and 1.30×10^{-2} $(mg/kg\text{-}day)^{-1}$, respectively. (IRIS, 1989).
The lifetime correction factor was used to adjust the risk estimates to the specific length of the exposure period. The resulting estimate was interpreted as the lifetime risk of a single excess cancer occurrence based on the specific exposure period. An average lifetime is defined as 70

years (EPA, 1986). For the WIPP, four LCs were required. These are:

- Residential: 5/70 and 20/70, because residential exposures are assumed to be for 24 hours per day, 365 days per year for the two exposure periods.

- Occupational: (8/24)(240/365)(5/70) and (8/24)(240/365)(20/70), since occupational exposures are assumed to be 8 hours per day, 240 days per year for the entire 5-year and 20-year period.

LONG-TERM RISK ESTIMATION FOR NONCARCINOGENS: ROUTINE OPERATIONS

Potential risks were estimated for noncarcinogens projected to average greater than 1 percent by weight of the waste (Rockwell, 1988). Estimates of daily intakes for each chemical were compared with acceptable daily levels for chronic intake (AIC) according to procedures for deriving "hazard indices" described in the SPHEM (EPA, 1986).

The hazard index (HI) for a given chemical may be defined as the ratio between the daily intake of that chemical and an acceptable reference level. Clearly, an HI less than unity (one) implies that the exposure to the given chemical is acceptable.

Hazard indices were calculated for each of these based on the estimated daily intakes. The chemical-specific hazard index was estimated as follows:

$$HI_i = I_{ri} / RL_i, \quad i = 1,...,3, \tag{G-4}$$

where:

HI_i = hazard index for the ith chemical, i = 1,...,3,

I_{ri} = estimated daily intake of the ith chemical for a given individual (mg/kg-day), i = 1,...,3,

RL_i = reference level for the i^{th} chemical (mg/kg-day), i = 1,...,3.

Here the reference level is the AIC, since exposures for the routine operations scenario are assumed to be over periods of 5 continuous years and 20 continuous years. The AICs for 1,1,1-trichloroethane and 1,1,2-trichloro-1,1,2-trifluoroethane used in the assessment are 6.3 and 30 mg/kg-day (IRIS, 1989). The oral AIC was used for 1,1,2-trichloro-1,2,2-trifluoroethane because the inhalation AIC was unavailable.

RISKS ASSOCIATED WITH ACCIDENT SCENARIOS

Accident events as defined in Appendix F are short-term events with respect to potential exposures and associated risks. To estimate these risks, hazard indices were calculated as described previously. The accident scenarios during operations at the WIPP are assumed to involve potential exposures to only the occupational population because all hypothetical accidents occur either in the waste handling building or underground. Because the risks to workers associated with the release of hazardous chemicals from accidents at the WIPP are well below health-based levels, risks to the public are not estimated. Short-term exposures to the public from these events will be less than those to workers because of the restricted access to the facility, operational protocols for accident control and cleanup, and the decreased concentrations of chemicals from dilution and diffusion in air.

Estimates of intake per exposure were compared with reference levels derived from appropriate, short-term occupational standards instead of AICs. These standards include the time-weighted average Threshold Limit Values (TLVs) (ACGIH, 1986) and Immediately Dangerous to Life and Health (IDLH) criteria (CHEMTOX, 1988). In the case of lead, an IDLH has not been established. Therefore, lead exposures were compared to the TLV-based allowable intake only. As before, an HI less than unity implies that the exposure to the given chemical is acceptable.

The TLV-based acceptable intake ($TLV\text{-}AI_i$) is derived by the following equation:

$$TLV\text{-}AI_i = (TLV)\ (V)\ (A_i)$$

where:

$TLV\text{-}AI_i$ = TLV-based acceptable intake (mg/exposure),

TLV = Threshold Limit Value for the i^{th} chemical (mg/m^3) (ACGIH, 1986),

V = respiratory volume for an occupational receptor during an 8-hour workday (12 m^3/day),

Ai = transfer coefficient (0.35 for lead (ATSDR, 1988a) and 1.0 or 100 percent absorption for all volatile organics).

The TLV and respiratory volume are based on an 8-hour workday. Therefore, the allowable intake is considered an acceptable level for an 8-hour occupational exposure.

The IDLH-based acceptable intake ($IDLH\text{-}AI_i$) is derived by the following equation:

$$IDLH\text{-}AI_i = (IDLH)\ (V)\ (EF)\ (A_i)$$

where:

$IDLH\text{-}AI_i$ = Immediately Dangerous to Life and Health-based acceptable intake (mg/exposure),

IDLH = IDLH for the i^{th} chemical (mg/m^3) (CHEMTOX Database, 1988),

V	=	respiratory volume for a worker during an 8-hour workday (12 m^3/day),
EF	=	exposure period and conversion factors (30 minutes per exposure, one hour per 60 minutes and one workday per 8 hours),
A_i	=	transfer coefficient (1.0 or 100 percent absorption for all volatile organics).

The IDLH is based on a 30-minute exposure. However, the respiratory rate is the volume breathed during an 8-hour day. The exposure period and conversion factors are used to determine the amount that can be taken into the body (i.e., acceptable intake) during a 30-minute exposure period.

REFERENCES FOR APPENDIX G

ACGIH (American Conference of Governmental Industrial Hygienists), 1986. Documentation of the Threshold Limit Values and Biological Exposure Indices, fifth edition, Cincinnati, Ohio.

ATSDR (Agency for Toxic Substances and Disease Registry), 1988a. Toxicology Profile for Lead, Oak Ridge National Laboratory, Oak Ridge, Tennessee, draft.

ATSDR (Agency for Toxic Substances and Disease Registry), 1988b. Toxicological Profile for Trichloroethylene, draft, prepared by Technical Resources, Inc. under contract, No. 68-03-23618.

ATSDR (Agency for Toxic Substances and Disease Registry), 1987. Toxicological Profile for Methylene Chloride, draft, prepared by Life Systems Inc.

Alexander et al. (H. C. Alexander, W. M. McCarty, and E. A. Bartlett), 1978. Toxicity of perchlorethylene, trichloroethylene, 1,1,1-trichloroethane, and methylene chloride to fathead minnows. Bull. Environ. Contam. Toxicol, 20:344-352, (cited in EPA, 1985).

Aviado, D. M., 1975. Toxicology 3:321.

Barrio-Lage et al. (G. Barrio-Lage, F. Z. Parsons, and R. S. Nassar), 1987. "Kinetics of the depletion of trichloroethylene," Environ. Sci. Technol. 21:366-370.

Bell et al. (Z. G. Bell, K. H. Oson, and T. J. Benya), 1978. "Final report of audit findings of the Manufacturing Chemists Association," administered trichloroethylene chronic inhalation study at Industrial Biotest Laboratories, Inc., Decatur, Illinois, unpublished, (cited in ATSDR, 1988b).

Black et al. (J. A. Black, W. J. Birge, W. E. McDonnell, A. G. Westerman, B. A. Ramey and D. M. Bruser), 1982. "The Aquatic Toxicity of Organic Compounds to Embryo-Larval Stages of Fist and Amphibians," Res. Report No. 133, Water Resource Institute, University of Kentucky, (cited in U. S. EPA, 1985).

Burek et al. (J. D. Burek, K. D. Nitschke, T. Bell, D. L. Wackerle, A. C. Childs, J. Beyer, D. A. Dittenber, L. W. Rampy, and M. J. McKenna), 1984. Methylene chloride: a two year inhalation toxicity and oncogenicity study in rats and hamsters, Fund. Appl. Toxicology 4 (1): 30-47, (cited in ATSDR, 1987).

Burek et al. (J. D. Burek, K. D. Nitschke, T. Bell, D. L. Wackerle, A. C. Childs, J. Beyer, D. A. Dittenber, L. W. Rampy, and M. J. McKenna), 1980. "Methylene chloride: a two year inhalation toxicity and oncogenicity study in rats and hamsters," Toxicology Research Laboratory, Health and Environmental Sciences, Dow Chemical Company, Midland, Michigan, (cited in ATSDR, 1987).

Busey, W. M., 1967. Unpublished data, Hazelton Labs., Falls Church, Virginia, sponsored by

Haskill Lab., DuPont Co., Wilmington, Delaware (February 1967), (cited in ACGIH, 1986).

CAPCOA (California Air Pollution Control Officer Association), 1987. Air Toxics Assessment Manual, August.

CHEMTOX Database, 1988. Resource Consultants, Brentwood, Tennessee.

Carlson, G. P., 1975. Toxicol. 5:69, (cited in ACGIH, 1986).

Cherry et al. (N. Cherry, H. Venable, and H. Woldron), 1983. "The Acute Behavioral Effects of Solvent Exposure," J. Soc. Occup. Med. 33:13-18, (cited in ATSDR, 1987).

Clark, D. G. and D. J. Tinston, 1973. Br. F. Pharm. 49:355, (cited in Clayton and Clayton, 1981).

Clayton, G. D., and F. E. Clayton, eds., 1981. Patty's Industrial Hygiene and Toxicology, third edition, John Wiley and Sons, New York.

Cornish, H. H. et al., 1973. Am. Ind. Hyg. Assoc. J., 34:487.

Council on Environmental Quality, 1975. Fluorocarbons and the Environment, (cited in Clayton and Clayton, 1981).

DiFerrante, E., 1979. Trace Metals: Exposure and Health Effects, Pergamon Press.

Dilling et al. (W. L. Dilling, N. B. Tefertiller and G. J. Kallos), 1975. Evaporation rates of methylene chloride, chloroform, 1,1,1-trichloroethane, trichloroethylene, tetrachloroethylene, and other chlorinated compounds in dilute aqueous solutions, Environ. Sci. Technol. 9 (a): 833-838, (cited in ATSDR, 1987).

EPA (U.S. Environmental Protection Agency), 1988a. "Solid Waste Disposal Facility Criteria," 53 Federal Register, 33421-2.

EPA (U.S. Environmental Protection Agency), 1988b. Users Network for Applied Modeling of Air Pollution (UNAMAP) 6 Version: Industrial Source Compled Model, dated 88167.

EPA (U.S. Environmental Protection Agency), 1987a. Technical analysis of new methods and data regarding dichloromethane hazard assessments. Draft document.

EPA (U.S. Environmental Protection Agency), 1987b. Update to Health Assessment Document and Addendum for Dichloromethane (Methylene Chloride): Pharmacokinetics, Mechanism of Action, and Epidemiology, draft report, EPA 600/8-87/1030A.

EPA (U.S. Environmental Protection Agency), 1987c. Health Advisories for 25 Organics, Office of Drinking Water, PB87-235578.

EPA (U.S. Environmental Protection Agency), 1986. Superfund Public Health Evaluation Manual. Office of Emergency and Remedial Response, Office of Solid Waste and Emergency Response, U.S. Environmental Protection Agency.

EPA (U.S. Environmental Protection Agency), 1985a. Criteria Document on Dichloromethane, Final Draft, Office of Drinking Water (cited in ATSDR, 1987).

EPA (U.S. Environmental Protection Agency), 1985b. Health Assessment Document for Dichloromethane (methylene chloride), final report, Environmental Protection Agency, Research Triangle Park, North Carolina.

EPA (U.S. Environmental Protection Agency), 1985c. "Development of Statistical Distribution or Ranges of Standard Factors Used in Exposure Assessments," Office of Research and Development, Washington, D.C.

EPA (U.S. Environmental Protection Agency), 1984. Health Effects Assessment for 1,1,1-Trichloroethane, Environmental Criteria and Assessment Office, Cincinnati, Ohio, ECAO-CIN-HOO5.

EPA (U.S. Environmental Protection Agency), 1980. Ambient Water Quality Criteria for Carbon Tetrachloride. Office of Water Regulations and Standards, Criteria and Standards Division, Washington, D. C., EPA 440/5-80-026.

EPA (U.S. Environmental Protection Agency), 1979. Water-Related Environmental Fate of 129 Priority Pollutants, Vol. I and II, 440/4-79-029, Office of Planning and Water Standards, Washington, D.C.

Farber, H., 1977. Manager of Environmental Affairs, Dow Chemical, Letter to James Price, Chief of Air Quality Data Analysis, Texas Air Control Board, Austin, Texas, (cited in IRIS, 1989).

Fogel et al. (M. M. Fogel, A. R. Taddio, and S. Fogel), 1986. "Biodegradation of chlorinated ethenes by a methane-utilizing mixed culture," App. Environ. Micro. 51:720-4, (cited in ATSDR, 1988).

Fujita et al. (H. Fujita, A. Koizumi, M. Yamamoto, M. Kumai, T. Gadamoto, and M. Skeda), 1984. "Inhibition of delta-azminoleuvliante dehydratase in trichloroethylene-exposed rats, and the effects on heme regulation," Biochem. Biophysi. Acta 800:1-10, (cited in ATSDR, 1988).

Fukuda et al. (K. Fukuda, K. Takemota, and H. Tsuruta), 1983. "Inhalation carcinogenicity of trichloroethylene in mice and rats," Ind. Health 21:243-254, (cited in ATSDR, 1988).

Gosselin, R. E., 1984. Clinical Toxicology of Commercial Products, fifth edition, Williams and Wilson, Baltimore, Maryland.

Hallen et al. (R. T. Hallen, J. W. Pyne, Jr., and P. M. Malton), 1986. "Transformation of chlorinated ethenes and ethanes by anaerobic microorganisms," in ACS Div. Environ. Chem., 192nd National Mt. 25: 344-6 (cited in ATSDR, 1988).

Hansch, C. and A. J. Leo, 1979. Substituent Constraints for Correlation Analysis in Chemistry and Biology. John Wiley and Sons, New York, New York, p. 173 (cited in ATSDR, 1987).

Haun et al. (C. C. Haun, E. H. Vernot, K. I. Darmer, and S. S. Diamond), 1972. Continuous animal exposure to low levels of dichloromethane, AMRL-TR-72-130, Paper No. 12, In: Proceedings of the Third Annual Conference on Environmental Toxicology, Wright-Patterson Air Force Base, Ohio, Aerospace Medical Research Laboratory, pp. 199-208 (cited in ATSDR, 1987).

Hazline (Hazard Line), January 1989. On line.

Healy et al. (T. E. Healy, T. R. Poole, and A. Hooper), 1982. "Rat fetal development and maternal exposure to trichloroethylene at 100 development and maternal exposures to trichloroethylene at 100 ppm," Dr. J. Anaesth: 54:337-341 (cited in ATSDR, 1988).

Holzworth, G. C., 1972. "Mixing Heights, Wind Speeds, and Potential for Urban Air Pollution Throughout The Contiguous United States," EPA.

HSDB (Hazardous Substances Data Bank), 1987. September 1987, on line.

IARC (International Agency for Research on Cancer), 1987. "IARC Monographs on the Evaluation of Carcinogenesis Risks to Humans," Overall Evaluations of Carcinogenicity: An updating of IARC Monographs, Volumes 1-42. (Supplement 7) International Agency for Research on Cancer.

IARC (International Agency for Research on Cancer), 1979. "IARC Monographs on the Evaluation of Carcinogenic Risk of Chemicals to Humans," vol. 20: Some Halogenated Hydrocarbons, World Health Organization, Lyon, France, pp. 371-399.

IRIS (Integrated Risk Information System), 1989. U.S. EPA., January.

Kazantzis, G. and R. R. Bomford, 1960. Lancet Vol. 1, p. 360 (cited in ACGIH, 1986).

Kimmerle, G., and A. Eben, 1973. "Metabolism, Excretion: 1. Experimental Exposures on Rats," Arch. Toxicol., Vol. 30, pp. 1115-126 (cited in ATSDR, 1988, p. 14).

Kirk-Othmer ed., 1985. Concise Encyclopedia of Chemical Technology, Wiley-Interscience, New York.

Kjellstrand P., B. Holmquist, P. Alm, M. Kanje, S. Romare, I. Jonsson, L. Mannson, and M. Bjerkamo, 1983. "Trichloroethylene: Further Studies of the Effects on Body and Organ Weights and Plasma Butyryl Cholinesterase Activity in Mice," Acta. Pharmacl. Toxicology. Vol. 53, pp. 375-384 (cited in ATSDR, 1988, pp. 14 and 53).

Kjellstrand P., M. Kanje, L. Mansson, M. Bjerkamo, I. Mortenson, J. Lanke, and B. Holmquist, 1981. "Trichloroethylene: Effects on Body and Organ Weights in Mice, Rats, and Gerbils," Toxicology, Vol. 21, pp. 105-115 (cited in ATSDR, 1988, p. 53).

Kniskern, L. A. and W. E. Pittsman, 1952. Industrial Test Lab., No. 2335, Progress Report NS-041001, Philadelphia Naval Shipyard, Pennsylvania (February 1952), cited by ACGIH, 1986.

Lee, D. H. K. ed., 1972. Metallic Contaminants and Human Health, Academic Press.

Lehmann, J., and D. Paech, 1972. "Effects of Some Vaporized Lipophilic Solvents on Carbon Dioxide Fixation, Experentia Vol. 28, pp. 1415-1416 (cited in U.S. EPA 1985, pp. 3-26).

Maling, H. H. et al, 1975. Tox. Appl. Pharm. Vol. 33, p. 291 (cited in ACGIH, 1986).

Maltoni et al. (C. Maltoni, G. Lefemine, and G. Cotti), 1986. "Experimental Research on Trichloroethylene Carcinogenesis," Archiv. Res. Industrial Carcinogenesis Series, eds. C. Maltoni and M. A. Mehlman, Vol. V, Princeton Scientific Publishing Co., Inc., Princeton, New Jersey (cited in ATSDR, 1988, p. 15).

Mazza, V., and A. Brancaccio, 1967. "Characteristics of the Formed Elements of the Blood and Bone Marrow in Experimental Trichloroethylene Intoxication," Folia Med., Vol. 50, pp. 318-324 (cited in ATSDR, 1988, p. 51).

Merck and Co., Inc., 1983 The Merck Index, 10th edition, Rahway, New Jersey.

Morris et al. (J. B. Morris, F. A. Smith, and R. H. Garman), 1979. "Studies on Methylene Chloride-Induced Fatty Liver," Exp. Molec. Path. Vol. 30, pp. 386-393 (cited in ATSDR, 1988, p. 55).

NIOSH (National Institute for Occupational Safety and Health), 1985. NIOSH Pocket Guide to Chemical Hazards, U.S. Department of Health and Human Services, Washington, D.C.

NTP (National Toxicology Program), 1986a. "Toxicology and Carcinogenesis Studies of Dichloromethane (Methylene Chloride) (CAS No. 75-09-2) in F344/N Rats and B6C3F1 Mice (Inhalation Studies)," Tech. Rep. Ser. 306 (cited in ATSDR, 1987, p. 66).

NTP (National Toxicology Program), 1986b. "Toxicology and Carcinogenesis Studies of Trichloroethylene in F344/N Rats and B6C3F1/Mice," NTP TR 243, U. S. Department of Health and Human Services, National Institutes of Health, Bethesda, Maryland (cited in ATSDR, 1988, p. 73).

NTP (National Toxicology Program), 1982. "Carcinogenesis Bioassay of Trichloroethylene in F344 Rats and B6C3F1 Mice", CAS No. 79-01-6, NTP 81-84, NIH Publication No. 82-1799 (cited in ATSDR, 1988, p. 73).

National Research Council, 1976. Halocarbons, Environmental Effects of Chlorofluoromethane Release (cited in Clayton and Clayton, 1981, p. 3090).

National Science Foundation, 1975. The Possible Impact of Fluorocarbon and Halocarbons on Ozone (cited in Clayton and Clayton, 1981).

Nitschke et al. (K. D. Nitschke, J. D. Burek, T. J. Bell, L. W. Rampy, and M. G. McKenna), 1982. "Methylene Chloride: a Two Year Inhalation Toxicity and Oncogenicity Study," Toxicology Research Laboratory, Health and Environmental Sciences, Dow Chemical Company, Final Report (cited in ATSDR, 1987, p. 66).

Nomiyama et al. (K. Nomiyama, H. Nomiyama, and H. Arai), 1986. "Re-Evaluation of Subchronic Toxicity of Trichloroethylene," Toxicol. Lett. 31 (Suppl.), 225 (Abstract) cited in ATSDR, 1988, pp. 54 and 59.

Nomiyama, K., and H. Nomiyama, 1977. "Dose-Response Relationship for Trichloroethylene in Man," Int. Arch. Occup. Environ. Health, Vol. 39, pp. 237-248 (cited in ATSDR, 1988, p. 13).

Norpoth et al. (K. Norpoth, U. Witting, M. Springoram, and C. Wittig), 1974. "Induction of Microsomal Enzymes in the Rat Liver by Inhalation of Hydrocarbon Solvents," Int. Arch. Arbeitmed., Vol. 34, pp. 315-321 (cited in ATSDR, 1987, p. 57).

Price et al. (P. J. Price, C. M. Hassett, and J. J. Mansfield), 1978. "Transforming Activities of Trichloroethylene and Proposed Industrial Alternatives," In Vitro, Vol. 14, pp. 290-293.

Ratcliffe, J. M., 1981. Lead in Man and the Environment, Ellis Horwood Ltd., Chichester, England.

Reinhardt et al. (C. F. Reinhardt, L. S. Mullen, and M. E. Maxfield), 1973. "Epinephrine-Induced Cardiac Arrhythmia Potential of Some Common Industrial Solvents," J. Occup. Med., Vol. 15, p. 953 (cited in ATSDR, 1988, pp. 60-61).

Rockwell International Corp., 1988. "Hazardous Constituents of Rocky Flats Transuranic Waste," Internal letter for distribution from J. K. Paynter, May 24, 1988, WCPO 2-31, Golden, Colorado.

Sax, N. I., 1989. Dangerous Properties of Industrial Materials, seventh edition, New Van Nostrand Reinhold Company, New York.

Sax, N. I., 1984. Dangerous Properties of Industrial Materials, sixth edition, New Van Nostrand Reinhold Company, New York.

Sittig, M., 1985. Handbook of Toxic and Hazardous Chemicals and Carcinogens, second edition, Noyes Publications, New Jersey.

Sittig, M., 1979. Hazardous and Toxic Effects of Industrial Chemicals, Noyes Data Corp.

Stewart et al. (R. D. Stewart, T. N. Fischer, M. J. Hosko, J. E. Peterson, E. D. Baretta, and H. C. Dodd), 1972. "Experimental Human Exposure to Methylene Chloride," Arch. Environ. Health, Vol. 25, pp. 342-348.

Stopps, G. J. and M. McLaughlin, 1967. American Ind. Hyg. Assoc. J. Vol. 23, p. 43 (cited by ACGIH, 1986).

Thomson, B. M., and R. J. Heggen, 1983. "Uranium and Water: Managing Common Resources," Chemtech, May, 1983.

Torkelson et al. (T. R. Torkelson, F. Oyen, D. D. McCollister, and V. K. Rowe), 1958. "Toxicity of 1,1,1-Trichloroethane as Determined on Laboratory Animals and Human Subjects," Am. Ind. Hyg. Assoc. J. Vol. 19, pp. 353-362 (cited in IRIS, 1989).

Verschueren, K., 1983. Handbook of Environmental Data on Organic Chemicals, second edition, Van Nostrand Reinhold Co., New York.

Vogel, T. M., and P. L. McCarty, 1985. "Biotransformation of Tetrachloroethylene to Trichloroethylene, Dichloroethylene, Vinyl Chloride, and Carbon Dioxide Under Methanogenic Conditions," Appl. Environ. Microbiol., Vol. 49, pp. 1030-1083 (cited in ATSDR, 1988).

Weinstein, R. S. and S. S. Diamond, 1972. "Hepatotoxicity of Dichloromethane (Methylene Chloride) with Continuous Inhalation Exposure at a Low Dose Level," in Proceedings of the Third Annual Conference on Environmental Toxicology, Wright-Patterson Air Force Base, Ohio, pp. 209-220 (cited in ATSDR, 1988).

Weisburger, E. K., 1977. "Carcinogenicity Studies in Halogenated Hydrocarbons," Environ.Health Perspec. Vol. 21, pp. 7-16.

Welch, L., 1987. "Reports of Clinical Disease Secondary to Methylene Chloride Exposure -- a Collection of 141 Cases," unpublished study submitted to ONTS/EPA 3/31 (cited in ATSDR, 1988).

White, J. F., and G. P. Carlson, 1981. "Epinephrine-Induced Cardiac Arrhythmias in Rabbits Exposed to Trichloroethylene Potentiation by Ethanol," Toxicol. Appl. Pharmacol., Vol. 60, pp. 466-471 (cited in ATSDR, 1988, pp. 60-61).

White, J. F., and G. P. Carlson, 1979. "Influence of Alterations in Drug Metabolism in Spontaneous and Epinephrine-Induced Cardiac Arrhythmias in Animals Exposed to Trichloroethylene," Toxicol. Appl. Pharmacol., Vol. 47, pp. 515-527 (cited in ATSDR, 1988, pp. 60-61).

Wilson et al. (B. H. Wilson, G. G. Smith, and J. F. Rees), 1986. "Biotransformation of Selected Alkylbenzenes and Halogenated Aliphatic Hydrocarbons in Methanogenic Aquifer Material: A Microcosm Study," Environ. Sc. Technology, Vol. 20, pp. 997-1002 (cited in ATSDR, 1988, p. 85).

APPENDIX H

PUBLIC INFORMATION AND INTERGOVERNMENTAL AFFAIRS

TABLE OF CONTENTS

Section Page

H.1 **PUBLIC INFORMATION**

Public information and participation activities undertaken during the preparation of the FEIS are described in Subsection 14.4 of the FEIS. This subsection lists the public hearings that were held and describes the notices of availability that were published. A 141-day public comment period was held on the draft EIS.

Since the completion of the FEIS, the DOE has undertaken a range of intergovernmental affairs and public information activities to inform the public of the development of the WIPP, provide opportunities for interested parties to express concerns and comments to the DOE, and keep key government agencies and interest groups informed of issues and progress related to the WIPP project. These intergovernmental and public information activities are described in detail below.

H.1.1 **ESTABLISHMENT OF THE WIPP VISITORS PROGRAM AND CENTER**

The WIPP Visitors Center was established in 1988 to provide information to area residents regarding the history, design, and plans for the WIPP. The center includes a multi-room exhibit that demonstrates the need for the WIPP, plant design, plans for waste handling, and projections for the life of the WIPP. The WIPP Visitors Center is managed by the WIPP Project Office of Public Affairs in Carlsbad, New Mexico. Staff members are available to discuss visitors' questions about the project. The center is an extension of the WIPP project tour program so that those who are unable to go to the site may receive similar information.

Interested groups may take tours of the WIPP as part of the visitors program. Between 1981 and 1984, only visits by foreign nationals were recorded as part of the tour program: there were 23 visits from 169 foreign visitors during that period. Between 1984 and 1989, all visits were recorded; 824 tours were conducted for 9,156 visitors. The visitors have included the Governors of New Mexico, Colorado, and Idaho and members of Congress from New Mexico, Colorado, Idaho, Oklahoma, and Louisiana, as well as the Secretary of the DOE.

H.1.2 **IMPLEMENTATION OF PUBLIC INFORMATION PROGRAM**

In implementing its public information program, the DOE has conducted public hearings; held public awareness tours; sponsored a speakers bureau; participated in dedications; attended professional and scientific meetings; held community update meetings; participated in community days and fairs; responded to media inquiries; sponsored media events; and prepared numerous publications addressing WIPP-related issues. These activities are described below. (Activities in support of this SEIS are discussed in Subsection H.3.)

Public Hearings. Since the FEIS was published in 1980, the DOE has participated in two sets of public hearings that have addressed environmental issues related to the WIPP. The issues these hearings have addressed include:

- Site and Preliminary Design Validation (SPDV) Program. The DOE held hearings on the results of the SPDV in Santa Fe and Albuquerque, New Mexico, April 19 and May 16, 1983, respectively. Following the hearings, a notice was published in the Federal Register that reaffirmed the 1981 Record of Decision and the decision to proceed to full construction.

- The Bureau of Land Management held administrative land withdrawal hearings for the WIPP in Albuquerque and Carlsbad, New Mexico in May, 1983.

- Land Withdrawal Bill. In 1987, Congress considered legislation that would permanently withdraw the land to be used for the WIPP from the public domain and assign administrative responsibility to the DOE. WIPP staff members testified at hearings for the bill in Washington, D.C., and Carlsbad, New Mexico. Congress did not act on this bill prior to adjournment.

Public Awareness Tours. The DOE conducted public awareness tours in 3 cities in Utah, 2 cities in Idaho, 14 cities in New Mexico, 5 cities in Colorado, 3 cities in Mississippi, 2 cities in Louisiana, and 5 cities in Wyoming. These tours informed residents and community officials along waste transportation routes about the WIPP and transportation issues. The DOE issued press releases in each of these cities having news media and the tours received extensive media coverage. Almost 3,000 people attended the exhibits and discussed issues with WIPP staff members. Thousands more were reached through press coverage.

Speakers Bureau. Since the DOE established a speakers bureau in 1987, 376 presentations have been made to 15,628 persons in civic clubs, professional organizations, schools, and other groups. These presentations have covered issues such as transportation of waste to the WIPP, waste handling operations, safety at the WIPP, the WIPP environmental programs, and overviews of the WIPP for elementary and secondary students.

Dedications. The DOE has held official dedications for the WIPP and associated facilities and has invited the public to these events. These dedications have included the following:

- The groundbreaking for the waste handling building was held in 1984 and the facility was dedicated in 1987. About 560 persons attended the two functions. The Waste Handling Building is the largest surface facility at the WIPP.

- The WIPP Visitors Center was dedicated in 1988. This facility is located at the WIPP Project Office in Carlsbad, New Mexico. Approximately 175 persons attended its dedication.

- The Alternate Emergency Operations Center (AEOC) was dedicated in 1988. Located near Carlsbad, New Mexico, the AEOC was developed to provide another location for emergency personnel to conduct emergency response activities if the primary Emergency Operations Center (EOC) at the WIPP site is inaccessible during an emergency. The DOE negotiated an agreement for joint DOE, State, county, and city use of the AEOC. About 30 persons attended its dedication.

- The Safety and Emergency Services Building, which houses the Emergency Operations Center, the First Aid Station, the emergency equipment (ambulance, fire truck, rescue vehicle), and Environmental, Safety and Health employees was dedicated in 1989.

- The DOE developed and installed a display on the WIPP project at DOE's National Atomic Museum at Kirtland Air Force Base in Albuquerque, New Mexico. About 50 persons attended the 1988 opening of the display.

- The DOE provided a regularly updated display on the WIPP project for the Carlsbad Centennial Museum which attracted hundreds of visitors in 1988.

Professional Conferences. The DOE has provided professional conferences with information about the WIPP project through professional conferences as follows:

- In 1988, the DOE's exhibit presented WIPP information to 700 radioactive and hazardous waste management professionals at the DOE Model Conference in Oak Ridge, Tennessee.

- At the 1987, 1988, and 1989 Waste Management Conferences in Tucson, Arizona, WIPP information was presented to 1,300 national and international radioactive waste management professionals each year.

- At Carlsbad, New Mexico, in May 1988 and Odessa, Texas, in December 1988, the WIPP Institutional Program gave status updates on institutional activities within the western and southern States to Defense Transuranic Waste Program participants.

- In November 1988, a presentation was made on the WIPP project and institutional and public affairs outreach to the American Society for Public Administration at El Paso, Texas.

- The Public Awareness display was exhibited at the National Conference of State Legislators in Tulsa, Oklahoma in August 1989. Approximately 1,500 persons visited the display, including legislators from every State.

- In 1989, the DOE provided a WIPP information booth at the annual meeting of the National Conference of State Legislators in Tulsa, Oklahoma. More than 6,000 legislators, legislative staff members, and other government officials attended.

Other Groups. The DOE has also provided information about the WIPP project to groups whose main interest relates to an aspect of the WIPP. These meetings included the following:

- Student Leadership Conference at New Mexico Tech's American Indian Science and Engineering Society's Student Leadership Conference in Socorro. The DOE participated in this 1989 activity, the purpose of which was to interest New Mexico Indian high school students in science and math. About 60 students attended this event.

- Operation CARE (Combined Accident Reduction Effort) in 1989 in Santa Fe, New Mexico. The DOE provided a speaker and an information booth at this meeting, which brought together about 300 law enforcement and highway patrol officials from across the nation.

- Health Physics Society Annual Meeting in 1989 in Albuquerque, New Mexico. The DOE provided an information booth at this meeting, which brought together about

3,000 national and international health physics professionals.

- Ninth International Symposium on Packaging and Transportation of Radioactive Materials (PATRAM) in 1989 in Washington D.C. The DOE provided an informational booth at this event, which drew about 800 national and international experts in the fields of packaging and transporting radioactive waste.

- National Association of Governors' Highway Safety Representatives annual meeting in 1989 in Tulsa, Oklahoma. The DOE provided an information booth at this event, which brought together about 400 State highway safety officials.

Community Activities. The DOE has held both regularly and specially scheduled community update meetings with community leaders in New Mexico. Updates on the WIPP project have been held in Carlsbad, Artesia, Roswell, Vaughn, and Hobbs. Seminars explaining how to participate in the Federal government procurement system have also been held in these locations for local businesses and contractors.

In the informal context of "community days," the DOE has provided the community with opportunities to meet with WIPP staff members and tour its facilities. These events included the following:

- WIPP Family Day at the WIPP site in 1987 and 1989. The DOE invited families of WIPP employees to tour the site. These events provided WIPP employees' family members with a general overview of the facility, a demonstration and overview on transportation, an environmental overview, and tours of the Waste Handling Building and the underground areas.

- Southeast New Mexico Community Leaders Day in 1988. The WIPP Public Affairs Office organized this event for elected officials and community leaders in southern New Mexico. The event included surface and underground tours and overviews of the WIPP project.

- Southeastern New Mexico Community Days in 1988. Organized by the WIPP Public Affairs Office, this event drew about 1,450 persons. The DOE provided overviews and surface and underground tours.

- Northern New Mexico Community Day in 1988. The WIPP Public Affairs Office organized this event, which included a general overview, transportation overview and demonstration, environmental overview, and tours of the Waste Handling Building and the underground areas. The event drew about 785 persons.

- Water Fair. The DOE assisted the State of New Mexico in gathering water samples from the Carlsbad area by co-sponsoring a Water Fair with the Environmental Improvement Division. More than 70 samples were brought to the fair by residents wishing to receive free water analyses.

- Eddy County Fair, 1985 through 1989. The DOE provided an information booth and exhibit at this fair in Carlsbad, New Mexico. About 2,500 people visited the booth.

- Lea County Fair, August 1989. The DOE provided an information booth and exhibit at this fair in Lovington, New Mexico; almost 700 people visited the booth.

- Eastern New Mexico State Fair in 1986, 1987, 1988, and 1989. The DOE provided an information booth and exhibit at this fair in Roswell, New Mexico. About 2,000 persons visited the booth.

- New Mexico State Fair in September 1988 and 1989. The DOE sponsored an information booth and exhibit at this fair in Albuquerque, New Mexico. A total of approximately 18,000 persons stopped at the booth.

- Knowles Frontier Day, July 1989. The WIPP Public Affairs Office provided an information booth and exhibit at this event which is based around fire protection and emergency response; over 100 people visited the booth.

- Science showcase. In 1987, 1988, and 1989, the DOE participated in the Carlsbad School System's Science Showcase program. The goal of this program is to encourage Carlsbad's young people to view science as a creative discipline that offers a wide range of career opportunities. Each year, more than 1,100 students, teachers, and parents learn about the WIPP at this event.

Media. The DOE, through its Office of Intergovernmental and External Affairs and the WIPP Public Affairs Office, is committed to responding to press inquiries with accurate and timely information. In addition to requests for information from southeastern New Mexico, information has been provided to regional media including The Albuquerque Journal and Tribune, Albuquerque television stations, Albuquerque radio stations (KOB and KGGM), the Boise Statesman in Idaho, and the Denver Post and Rocky Mountain News in Colorado. National requests have included inquiries from The Chicago Tribune, USA-Today News, Newsweek and Time magazines, The New York Times, Cable News Network, and The MacNeil/Lehrer Report.

Media events sponsored by the DOE were designed to provide the media with in-depth information about key issues of public interest. For example:

- The DOE exhibited the TRUPACT-II testing in Albuquerque, New Mexico. Local and national media and public officials were invited to this event. The TRUPACT-II containers were dropped from 30 feet onto an unyielding surface, dropped onto a blunted spike, and burned.

- The DOE sponsored a tour to demonstrate the TRUPACT-II full-scale model in Carlsbad, New Mexico; Idaho Falls, Idaho; and Portland, Oregon. The purpose of this tour was to answer questions from interested media about the proposed transportation routes for waste materials and about the proposed contents of the TRUPACT-II containers.

Publications. In addition to the public information activities described above, the DOE has prepared numerous publications addressing different WIPP issues. The titles of these publications are:

"Waste Isolation Pilot Plant -- WIPP"
"In Situ Testing at the Waste Isolation Pilot Plant"
"Visitor Information"*
"Certification Requirements"
"Transuranic Waste"

"Environmental Protection"
"Participants/Lines of Communication"
"Why Salt? Why Southeastern New Mexico?"*
"Raptor Studies and the WIPP Environment"
"Waste Handling Procedures at WIPP"
"Commonly Asked Questions"*
"Transportation: A Satellite Tracking System"
"Transportation: TRUPACT-II"*
"Safety Throughout the Project"
"Waste Handling Building"
"Highway Route Selection"
"States Training and Education Program"
"Public Law 96-164"
"Where Will Waste Come From?"
"WIPP Project Speakers Bureau Brochure"
"Draft Plan for Waste Isolation Pilot Plant Test Phase: Performance Assessment and Operations Demonstration"
"DOE Invites Public Comments on WIPP-SEIS Document."

* Spanish translations of these publications are being prepared.

H.2 **INTERGOVERNMENTAL AFFAIRS**

An important function related to the WIPP Project Office of Public Affairs is to keep interested government officials informed of key issues and progress related to the WIPP project. In the process, the DOE has worked closely with numerous Federal, State, and local government agencies. In some cases, the DOE has regularly attended meetings of key governmental agencies, and the WIPP project staff members have participated in the ongoing meetings of governmental groups as follows:

- The Environmental Evaluation Group (EEG) provides independent oversight of the WIPP project. The group has a professional staff and is responsible to the president of the New Mexico Institute of Mining and Technology. WIPP staff members have conducted 30 quarterly reviews of the WIPP project for the EEG and published 42 reports on their investigation and analyses of the WIPP.

- The Radioactive and Hazardous Materials Committee (RHMC) oversees WIPP project activities for the New Mexico legislature. Since 1979, WIPP staff members have attended about 50 meetings of the RHMC.

- The Radioactive Waste Consultation Task Force (RWCTF) is an executive task force that oversees the WIPP project for the Governor of New Mexico. In 1985, the DOE was invited to the meetings of the RWCTF and has attended eight meetings since then.

- The National Academy of Sciences (NAS) WIPP Panel is composed of 11 prominent scientists and has met approximately 3 times a year since 1979. WIPP project staff members were available for the 30 meetings.

- The Pacific States Alliance (PSA) is a four-state committee established to study and recommend measures to transport radioactive material safely through Washington, Oregon, Idaho, and Wyoming. The DOE participated in five meetings in 1988 and 1989 with the PSA and attends all PSA meetings to identify concerns, address questions, and provide project updates.

- The Western Governors' Association (WGA) is an alliance of governors from 11 western States dedicated to uniformly representing the western governors in intergovernmental affairs. The DOE regularly attends WGA meetings to identify concerns, address questions, and provide project updates.

- Congressional support. The WIPP Project Office has responded on numerous occasions to requests for information from different members of Congress and has conducted briefings and tours for interested members who have visited the facility.

In addition to regular involvement with these governmental groups, the WIPP Project Office of Public Affairs has met on request and initiated meetings with other governmental groups interested in the project. These meetings have included the following:

- Santa Fe Interested Citizens. Approximately 20 elected and appointed Santa Fe, New Mexico, leaders toured the WIPP site and received briefings.

- National Congress of American Indians (NCAI). The WIPP Project Office met with NCAI members on four occasions. In December 1987, WIPP staff members met with the leaders of New Mexico Indian Tribes and Pueblos. In February 1988, WIPP staff members met with officials of Indian Tribes and Pueblos from outside New Mexico. In December 1988, a WIPP representative met with tribal officials at a meeting arranged by the NCAI at a transportation coordinating group meeting. In September 1989, WIPP staff attended and participated in the NCAI-sponsored tribal seminar on nuclear waste. This seminar's purpose was to familiarize Federal officials with tribal cultural and sovereignty rights.

- All Indian Pueblo Council (AIPC). After AIPC publicly expressed opposition to the WIPP project, the DOE met with the AIPC in 1988 to hear concerns and respond to questions and comments. The AIPC represents New Mexico's 19 Indian pueblos on matters for which unity and numbers enhance the pueblos' interests.

- Interstate Route 84 Task Force. In July 1988, WIPP staff members conducted a public information tour in Oregon along the route of proposed Interstate Route 84 to provide information on the transport of TRU wastes through Oregon and to identify and address concerns. WIPP project staff members responded to media questions, provided technical expertise, and displayed the full-scale TRUPACT-II model.

- Hanford Waste Board and Advisory Committee (Oregon). This group sponsored four public information meetings along the proposed Interstate Highway 84 corridor in Oregon. The DOE attended these meetings to provide the public with information on the transport of TRU wastes through Oregon and to identify and address concerns. WIPP project staff members responded to media questions, provided technical expertise, and displayed the full-scale TRUPACT-II model.

- Western Interstate Energy Board (WIEB). WIPP project staff members attended three meetings held by the WIEB on the WIPP during 1987 and 1988. The WIEB is an interstate compact group representing 16 western States in many environmental and intergovernmental affairs.

- Southern States Energy Board (SSEB). The SSEB held a meeting on the WIPP in 1987 which WIPP project staff members attended. The SSEB is a non-profit interstate compact serving as the regional representative of 16 southern States in energy and environmental matters. The SSEB also held a meeting in Carlsbad, New Mexico and toured the WIPP site in September 1988.

- DOE Field Offices. Personnel associated with or supporting the WIPP Project Office meet with the DOE's Idaho, Oak Ridge, and Savannah River Operations Offices to plan, coordinate, and interface with the States within their regions.

- Office of Civilian Radioactive Waste Management (OCRWM). The WIPP Project Office met and worked with DOE OCRWM five times in 1987, 1988 and 1989. During these meetings, the DOE attended OCRWM's Transportation Coordination Group meetings to exchange information about transportation policy, hosted the

OCRWM Transportation Institutional Support Manager on a visit to the WIPP site, and participated in the OCRWM Institutional Planning for Transportation Activities meeting.

- Mine Safety and Health Administration (MSHA). Pursuant to a Memorandum of Understanding between MSHA and DOE, the MSHA conducts safety inspections of the underground WIPP facility.

- Other State of New Mexico Agencies. The DOE met with the State Highway Commission to discuss highway upgrading and with the Radiation Technical Advisory Council to discuss TRU waste transportation and other agenda items. The State Highway Commission has responsibility for maintenance of State roads and shipments of hazardous materials over those roads. The Radiation Technical Advisory Council is responsible for radiation protection in New Mexico.

- Local government agencies. The DOE met with the Raton, New Mexico City Council in 1988 to address concerns about waste transportation. After the meeting, the City Council defeated a resolution to restrict the transportation of radioactive waste through city limits. Instead, the council voted to support the New Mexico Municipal League's resolution. The DOE has addressed the Santa Fe City Council on the constituents in and the handling of radioactive mixed wastes and has participated in public forums sponsored by the League of Women Voters, City of Santa Fe, and Santa Fe County.

- Commercial Vehicle Safety Alliance (CVSA). The DOE met with the CVSA in 1988 to keep informed on CVSA's pilot study for the inspection of radioactive shipments. The CVSA is an alliance of States that is trying to establish uniform inspection procedures for all hazardous materials shipments.

- Confederated Tribes of the Umatilla Indian Reservation (CTUIR). The DOE attended a CTUIR sponsored workshop on transportation of radioactive materials in 1988. The DOE gave a WIPP update to the CTUIR Board of Trustees in August 1989. The CTUIR is composed of the Umatilla, Cayuse, and Walla Indian Tribes in northeastern Oregon.

- Eight Northeast Tribes of Oklahoma. The DOE met with this group in 1988 to inform the tribes about WIPP issues. This group is a State-chartered forum that represents the Eastern Shawnee, Seneca-Cayuga, Quapaw, Peoria, Wyandot, Miami, Modoc, and Ottawa Indian Tribes on issues of common concern.

H.3 INTERGOVERNMENTAL AFFAIRS AND PUBLIC INFORMATION PLAN FOR THE WIPP SEIS

In conjunction with the preparation of the WIPP final SEIS, the DOE Albuquerque Operations Office has established an Office of Intergovernmental Affairs and Public Information (IAPI). The objective of the IAPI Office is to ensure that public information and public participation activities for the SEIS are in compliance with the CEQ's regulations implementing the NEPA and DOE's NEPA guidelines. To ensure the public has adequate opportunities for involvement in the SEIS, the DOE implemented the following activities:

- Intergovernmental Affairs. The DOE has met with 1) representatives of the States of New Mexico, Colorado, Utah, Idaho, Washington, Oregon, Wyoming, California, Arizona, Nevada, Kentucky, and Arkansas; 2) the Western Governors' Association; 3) the Southern States Energy Board; 4) the National Congress of American Indians and Council of Energy Resource Tribes; 5) Environmental Protection Agency and the Bureau of Land Management; 6) key environmental groups; 7) the Environmental Evaluation Group; and 8) Congressional representatives from the host and corridor States and from oversight committees such as the House Armed Services Committee. The purposes of these meetings were to discuss the planned content of the SEIS, to receive any input regarding environmental issues, and to review the schedule for completion of the NEPA process.

  These meetings provided important input into the development of the SEIS, particularly in the focusing of transportation issues and collection of relevant data. The meetings helped the SEIS Office of Intergovernmental Affairs and Public Information identify information needs that government officials and the interested public may have.

- Federal Register Notices. A Notice of Preparation of the SEIS appeared in the Federal Register on February 17, 1989. On April 21, a Notice of Availability for the SEIS was published that also announced the beginning of the public comment period. Subsequently, the DOE published five more Federal Register Notices announcing various changes and additions to the public hearing schedule and extensions of the public comment period (May 26, June 12, June 26, July 7, and July 11, 1989). The total public comment period was 90 days in length.

- Toll-Free Request Line. At the beginning of the pubic comment period, the DOE established a toll-free telephone line connected to an answering machine at the SEIS Project Office. This line allowed citizens from around the U.S. to call 24-hours a day, seven days a week to register to speak at the public hearings on the draft SEIS. The line was also available to request copies of the SEIS; to obtain fact sheets, summaries, or other informational materials on the SEIS; to be placed on the SEIS mailing list; or to receive a return phone call from someone on the SEIS Project Office staff.

- Mailing List. The DOE developed a comprehensive mailing list for distribution of the SEIS and other materials. The mailing list is a compendium of approximately 2,000 interested citizens; Federal, State, and local agencies; elected officials; tribal officials; public interest groups; and others. Sources for this mailing list consisted of those responding to the February 17, 1989, Federal Register notice, lists from the 10 waste generator or storage facilities, the FEIS distribution list, telephone requests received on the SEIS toll-free telephone line, the DOE Office of Intergovernmental and External Affairs, and others. In response to informational materials prepared by the SEIS Project Office during the early public information efforts on the SEIS, numerous interested parties asked to be added to the mailing list.

- Public Hearings. During the 90-day public comment period, the DOE held a total of nine public hearings on the draft SEIS in seven States, including:

Atlanta, Georgia	May 25, 1989
Pocatello, Idaho	June 1, 1989
Denver, Colorado	June 6, 1989
Pendleton, Oregon	June 8, 1989
Albuquerque, New Mexico	June 13-14, 1989
Santa Fe, New Mexico	June 15-17, 1989
Artesia, New Mexico	June 22, 1989
Odessa, Texas	June 26, 1989
Ogden, Utah	July 10, 1989

The DOE's approach for notifying the public of an upcoming public hearing included public service announcements, display ads, press releases, and press conferences. For example, prior to the public hearing in Atlanta on May 25, the DOE sent public service announcements to 27 radio and television stations in Georgia, South Carolina, Tennessee, Kentucky, and Ohio. In the same States, the DOE took out display-type advertisements in 16 newspapers of general circulation. Two days before the hearing, the DOE issued a press release, and on the day before and the day of the hearing, the DOE held press conferences.

Similar efforts were undertaken for all of the hearings. As a result of these types of activities, the DOE succeeded in attracting close to a thousand commenters to the nine hearings, in addition to the almost 900 written comments it received.

- Others. A variety of press releases and public service announcements regarding the SEIS have been prepared and distributed to the media and to others on the mailing list.

APPENDIX I

METHODS AND DATA USED IN LONG-TERM CONSEQUENCE ANALYSES

TABLE OF CONTENTS

LIST OF FIGURES

LIST OF TABLES

I.0 INTRODUCTION

This appendix describes the analytical methods, codes, and exposure calculations used to calculate the impacts from the postulated long-term release scenarios discussed in Subsection 5.4. It also presents the basis for selecting the input data values used in the codes.

COMPARISON WITH THE DRAFT SEIS

Two principal changes have been made for this final SEIS since the draft SEIS was published in April 1989. In Case I, a model describing the potential for release from an undisturbed repository, a third scenario has been added, Case IC. This scenario assumes a near-complete failure of tunnel and shaft seals, letting some radionuclide-bearing brine move through those tunnels and shafts to the Culebra aquifers, whence they move to the hypothesized stock well 5 km downstream.

In addition, the earlier Cases IIA and IIC have been recalculated as Cases IIA(rev) and IIC(rev). These two were chosen for recalculation because they were the extremes of the earlier analyses. Those scenarios were analyzed using a one-dimensional, stream-tube, single-point-injection version of the SWIFT-II code. For this final SEIS, these two calculations have been repeated with a more realistic version of that code, one that incorporates two-dimensional transport with lateral diffusion, allows for a time-dependent width of the injection plume, and uses radionuclide-specific diffusivities. The code also had available an improved description of the transmissivity field of the Culebra based on more data (i.e., the results of the H-11 multipad tests) than had been available for inclusion in the draft. The more important inputs used in the analyses reported in this final SEIS are compared below with those used in the draft SEIS.

Brine reservoir. The description of the brine reservoir under the site is based on measurements made on the WIPP-12 brine reservoir. Somewhat higher initial pressures have since been observed in a brine reservoir at the Belco well to the south, but the brine reservoir description in the revised Case II has not been changed. All the other input parameters for Case IIC are taken at the end of their ranges. Brine reservoir parameters will be varied in the final performance assessment.

Borehole properties. The properties of the deteriorated drill hole are already at the extremes of their ranges as given in Subsection I.2.4. No new data have come to the DOE's attention to warrant changing these inputs further.

Waste properties. A few changes were made in the properties of the waste and the waste disposal panels. The quantities of radionuclides present are larger, because the mass inventory for the whole repository has been scaled up to fill the entire repository to its design volume (Appendix B).

Also, the inventory was aged for 175 years instead of 100 years before starting the calculations, this being the sum of the time to the end of the institutional control period (100 years) and the time (75 years) until the borehole plug starts to deteriorate.

Salado brine inflow. The brine inflow to the panels was increased from 1.3 m^3/yr to 1.4 m^3/yr, as a result of a modification of the Salado lithostatic pressure value (from 14 MPa to 14.8 MPa) used in estimating long-term brine inflow rates.

Brine properties and inflow into the Culebra. The density of the Castile groundwater was increased from 1.0 g/cm^3 to 1.24 g/cm^3 in the calculations to be consistent with its load of solutes. The net effect has been to decrease the rate at which brine enters the Culebra from the borehole by 30 percent (Table 5.65). For example, in Case IIA(rev), the inflow from the borehole at early times is reduced from 11.2 m^3/yr to 8.7 m^3/yr; and in Case IIC(rev) at early times from 99 m^3/yr to 74 m^3/yr.

Groundwater transport. An important difference from the draft SEIS has been to build increased capabilities into the SWIFT II code, allowing it to make more realistic predictions. The original Case II calculations used a one-dimensional stream-tube approach for simulating the transport of contaminants in the Culebra. The revised Case II transport calculations presented in this final SEIS use a two-dimensional system: 1) to provide estimates of breakthrough concentrations for the contaminants at the stock well that more realistically incorporate lateral dispersion and species-specific effects, and 2) to provide quantitative estimates of the cumulative release of radionuclides at distances from the waste panel coincident with the present land-withdrawal boundary and with the stock well location. The added capability for calculations in two dimensions permits an explicit time-dependent size of the initial injection disturbance shown in Figure 5.7.

Species-specific diffusivities. Separate diffusivities have been included for each radionuclide as opposed to one figure for all. Thus in Case IIA(rev), the former figure of 1×10^{-6} cm^2/s now ranges from 1.0 to 3.8×10^{-6} cm^2/s; and in Case IIC(rev) the former diffusivity figure of 5×10^{-7} cm^2/s now ranges from 5×10^{-7} cm^2/s to 2.0×10^{-6} cm^2/s (Tables I.2.12 and I.2.13). The net effect is to increase the diffusion into the matrix on either side of the fractures.

Culebra transmissivity distribution. The Case II calculations reported in the draft SEIS used a Culebra groundwater flow model calibrated to data collected approximately through October 1987 (LaVenue et al., 1988). An additional modeling effort has been completed that includes an expanded area covered and an expanded and revised data base of transmissivities and fluid heads. The new model differs from the previous one in that it is calibrated to all significant transient events (shaft construction, and the H-3 and H-11 multipad tests) near the off-site transport pathway between the waste disposal panel and the stock well. (See Subsection 4.3.3.3.)

Stock well location. Transmissivity data imply more fracturing south of the site. This results in a flow path that flows first to the east, then south, rather than almost straight south. As a result, the hypothetical stock well has moved about 540 m to the southeast. The distance along the flow path to the site boundary is now 3,610 m instead of 2,860 m, and to the stock well the distance is now 5,960 m instead of 4,840 m. The straight line distance from the center of the southwest panel to the stock well is 5.04 km.

Integrated releases. A principal purpose for including a two-dimensional flow model instead of a one-dimensional one was to be able to make realistic evaluations of the integrated releases of contaminants past the site boundary and past the stock well. These results are presented in Subsection 5.4.2.8.

I.1 **METHODS**

I.1.1 **THE NEFTRAN CODE**

The NEFTRAN code (Network Flow and Transport) (Longsine et al., 1987) is used to calculate radionuclide releases from an undisturbed repository in Cases IA, IB, and IC. It is a groundwater flow and radionuclide transport code developed by Sandia National Laboratories for the U.S. Nuclear Regulatory Commission. Codes that preceded NEFTRAN are NWFT (Campbell et al., 1980) and NWFT/DVM (Campbell et al., 1981). It was designed with the assumption that all significant flow and radionuclide transport progresses along discrete one-dimensional legs or paths. A flow field is represented by the assemblage of these legs forming a network. The solution of the flow equations in NEFTRAN requires pressure boundary conditions and it is required that these conditions be specified as part of the input data.

NEFTRAN first solves the flow equations for the network using Darcy's Law. From this, the average interstitial fluid and radionuclide velocities for each leg are calculated. The code then uses a Distributed Velocity Method (DVM) applied over the entire length of the migration path using an average velocity for each isotope calculated from the isotopic velocities in all legs. The DVM technique treats convective-dispersive transport by simulating the movement of an ensemble of representative particles. Dispersion is treated by assigning a velocity distribution to these particle ensembles (Campbell et al., 1986).

The user can set up and input any network in the generalized network scheme through a specification of the number of legs, the number of junctions, the junctions bounding each leg, and the junctions where boundary conditions are specified. The hydraulic head gradient provides the driving force for fluid flow through the leg. Conservation of mass at each junction is the assumption that allows the flow network to represent a flow system. This conservation law is given by

$$\sum_j M_j = 0 \qquad \text{(I-1)}$$

where j is the index of summation over all legs that are connected at the given junction, and M_j is the mass flow rate for the j^{th} leg in units of mass per unit time. For the case when the j^{th} leg is bounded by junctions j1 and j2, the mass flow rate in the leg is represented by the equation

$$M_j = \rho_j K_j A_j \left[\frac{(P_{j1} - P_{j2})}{Z_j \rho_j g} + \frac{(E_{j1} - E_{j2})}{Z_j} \right] \qquad \text{(I-2)}$$

where A_j is the cross-sectional area, K_j is the hydraulic conductivity, E_{ji} is the elevation of the i^{th} junction, g is the acceleration due to gravity, P_{ji} is the pressure at the i^{th} junction, Z_j is the length of the leg, and ρ_j is the fluid density.

To account for the effects of brine concentration on the flow, the hydraulic conductivity is weighted as

$$K_j = K'_j \left[\frac{\mu_f}{\mu_j}\right] \left[\frac{\rho_j}{\rho_f}\right] \tag{I-3}$$

K_j is the fresh-water hydraulic conductivity for the j^{th} leg, μ_f and ρ_f are the respective viscosity and density of fresh water at approximately 20 degrees C, μ_j and ρ_j are the respective actual viscosity and density in the j^{th} leg.

A matrix equation is developed by applying Equation (I-1) to a boundary junction, substituting Equation (I-2) for M_j with $\Theta_j = A_j\ K_j\ /Z_j\ g$, and repeating this procedure for each junction in the network. The resulting matrix equation is

$$\underline{\underline{\Theta}}\ \underline{p} = \underline{e} \tag{I-4}$$

where $\underline{\underline{\Theta}}$ is a matrix of coefficients containing functions of $\Theta_j = A_j\ K_j/Z_j\ g$, $\underline{p}$ is a vector of unknown pressures, and $\underline{e}$ is a vector of junction elevations and boundary pressures.

NEFTRAN calculates the mass flow rate in each leg using Equation (I-2) and divides it by the corresponding density to determine the volumetric flow rate. This flow rate is then used to calculate the fluid velocity for the j^{th} leg

$$v_{fj} = \frac{Q_j}{A_j \cdot \varphi_j} \tag{I-5}$$

where φ_j is the porosity of the leg and Q_j is the volumetric flow rate.

If j=1,2,...,n is the number of legs along a given radionuclide migration path, NEFTRAN uses the weighted average fluid velocity v_f over the migration path given by

$$\bar{v}_f = \sum_{j=1}^{n} Z_j \Bigg/ \left[\sum_{j=1}^{n} \frac{Z_j}{v_{fj}}\right] \tag{I-6}$$

for the transport simulation such that it preserves total migration time. This approach results in a combination of all legs into a single one-dimensional segment having average properties. This approach has been shown to be sufficient provided the legs in the migration path represent either porous media or transport through fractures with no diffusion into the adjacent matrix blocks.

The Distributed Velocity Method (DVM) is the direct simulation technique used in NEFTRAN to treat the convective-dispersive transport of chains of radionuclides. The DVM approach can treat radionuclide chains of arbitrary length and distribution coefficients. Some numerical dispersion can result from the DVM technique. This dispersion, however, can be controlled while still retaining the efficiency required for risk analysis (Campbell et al., 1981).

The DVM technique is based on the concept that, due to heterogeneity of the flow field, several alternative paths exist for migration of particles from position x_ to x where x is the receiver point and donor points are located at coordinates x_. If the density of an ensemble of particles at time t_ is given by $\rho(x\_,t\_)$, the density $\rho(x,t)$ at x for $t > t\_$ can be determined by introducing a velocity distribution P(v). The equation describing the density of particle at point x is obtained by summing over all possible donor points in the following manner

$$\rho_o(x, t) = \int_{-\infty}^{\infty} dv P(v) \rho(x - v\Delta t, t - \Delta t) \qquad \text{(I-7)}$$

where

$$\Delta t = t - t\_$$

The propagation of the initial conditions from time t_ to time t is given by Equation (I-7). An integration over "injection" time must be performed in addition to that over velocity, if a source $S(x,\tau)$ is included. Sources could result from either transport of wastes from the repository or decay of a radioactive parent. The propagation of the density function from time t_ to t (Equation I-7) is implemented numerically in DVM by discretizing time and space. Also, the velocity-space domain is discretized by dividing the velocity dimension into a few intervals based on equal probability. The propagation of particles is then implemented by simulating the migration of particles in each velocity interval. For the latter, the location of the source is time dependent.

NEFTRAN provides for every species to have a different retardation factor in each leg of the migration path. The average species velocity for each leg is treated separately. The mean species velocity caused by dispersion in the leg for the k^{th} species in the j^{th} leg is given by

$$\bar{v}_{kj} = v_{fj}/R_{kj} \qquad \text{(I-8)}$$

NEFTRAN maintains a mean velocity for each species while calculating distributed velocities about the mean in each leg. When particles begin a time step as a parent species and end the time step as a daughter, NEFTRAN calculates the average velocity by weighting species velocities with the average time spent as each species

$$V_m(1,...\rho) = \frac{1}{} \sum^{p} TS_j\, v_j \qquad \text{(I-9)}$$

(Δt) $j=1$

The output of NEFTRAN consists of the following:

1. Pressure at each junction of the flow network

2. Volumetric flow rate at each leg of the flow network

3. Discharge rate (in curies/day) of each radionuclide as a function of time at the end of the transport path specified by the user.

In the calculation of Cases IA, IB, and IC, the arrival times of radionuclides at the top of shaft or any other point of interest were determined by the times at which the discharge rates rose to 10^{-10} Ci/day. The threshold used for the arrival of stable lead was 8×10^{-9} mg/L.

I.1.2 **THE SWIFT II GROUNDWATER TRANSPORT CODE**

The SWIFT II (Sandia Waste Isolation Flow and Transport) Code is used to calculate releases from a disturbed repository (Cases IIA through IID, including Cases IIA[rev] and IIC[rev]). This code requires specification of the time-varying flow out of a brine reservoir and up the borehole to the Culebra. This flow rate is calculated by analytical models described in this subsection. SWIFT II is a fully transient, three-dimensional code that has been under development and maintenance since 1975. The program has been comprehensively documented and extensively tested. Calculational comparisons to experimental data have resulted in a program that is both accurate and versatile.

SWIFT II solves the coupled equations for transport in geologic media. This code considers the following processes:

- fluid flow
- heat transport
- dominant-species miscible displacement (brine)
- trace-species miscible displacement (radionuclide chains).

The first three processes indicated above are coupled by means of the fluid density and viscosity. This coupling results in a determination of the velocity field that is needed for a calculation of the third and fourth processes.

I.1.2.1 **Implementation of Brine-Reservoir and Borehole Submodels**

Figure I.1.1 is a drawing of a brine-reservoir breach. It represents a borehole that passes through the repository and connects a brine reservoir to the Culebra. LaVenue et al. (1988) have detailed the most recent model of the Culebra, having calibrated the steady-state flow field to the field data using SWIFT II. The analyses for cases IIA(rev) and IIC(rev) use the transmissivity distribution Culebra model of LaVenue et al. (1988), updated as described in Subsections 4.3.3.2 and 5.4.2.6, with the pressurized brine reservoir specified analytically as a source term.

Figure I.1.1

In terms of its initial and hydraulic properties, the brine-reservoir submodel is represented by the form

$$Q = A_Q + B_Q \delta p \qquad \text{(I-10)}$$

where δp is the change in pressure within the Culebra source block m (i.e., the block where the breach will penetrate the Culebra Dolomite) during time-step δt. Quantity Q is the volumetric rate of water injection into block m during time-step δt. Q, as well as the flow-rate parameters A_Q and B_Q, are assumed constant during δt. A_Q and B_Q are defined by equations I-34 and I-35, respectively. Q varies as a function of time step to reflect depletion of the brine reservoir.

The brine-reservoir submodel is discussed in the following three subsections. The first subsection describes the influence functions P_I and W_I used to characterize pressure and flow rate, respectively, at the borehole-reservoir interface. The second subsection specifies brine-reservoir response in terms of P_I and its time derivative P'_I. The third and final subsection couples the Culebra and the reservoir to determine a Culebra source term of the form specified in Equation (I-10).

Influence Functions. Van Everdingen and Hurst (1949) consider two basic influence functions useful in determining pressure drawdown and flow rate at the borehole-reservoir interface. W_I represents a constant-pressure condition at $r = r_w$ (Figure I.1.1). This term is called the terminal-pressure influence function. The second influence function P_I represents a constant-rate condition at $r = r_w$. This term is the terminal-rate influence function. These functions provide basic functions that, through superposition, result in a general solution.

P_I and W_I are derived from a dimensionless flow equation assumed to have cylindrically symmetric form

$$\frac{1}{r_D}\frac{\partial}{\partial r_D}\left[k_D r_D \frac{\partial \Delta p}{\partial r_D}\right] = \frac{\partial \Delta p}{\partial t_D} \qquad \text{(I-11)}$$

where Δp is the pressure drawdown.

For well radius r_w, porosity φ, total compressibility c, viscosity μ, and reference permeability k_o, the dimensionless quantities in Equation (I-11) are defined as follows:

$$r_D = r/r_w, \quad t_D = t/t_w, \quad t_w = \varphi c r_w \mu^2/k_o, \text{ and } k_d = k/k_o \qquad \text{(I-12)}$$

The reference permeability k_o is set equal to k for an homogeneous system. The result is $k_d = 1$, which is the form of the flow equation given in Van Everdingen and Hurst (1949).

Initial conditions assuming a state of equilibrium in the borehole and reservoir result in the equation

$$\Delta p(r_D, t_D = 0) = 0 \qquad \text{(I-13)}$$

The boundary condition at the wellbore-reservoir interface distinguishes two influence functions. For P_I,

$$\frac{\partial \Delta p}{\partial r_D}(r_D = 1, t_D) = -1 \; . \qquad \text{(I-14)}$$

For W_I,

$$\Delta p(r_D=1,t_D) = 1 \qquad \text{(I-15)}$$

The constant-rate influence function, P_I, is obtained as a solution of Equation (I-11) evaluated at the wellbore interface

$$P_I = \Delta p(r_D=1,t_D) \qquad \text{(I-16)}$$

The dimensionless flow rate at the wellbore interface, W_I, is given by $\partial \Delta p_D/\partial r_D(r_D=1,t_D)$. Integration over dimensionless time yields the constant-pressure influence function

$$W_I = \int_0^{t_D} \left[\frac{\partial \Delta p}{\partial r_D} \right]_{r_D = 1} dt_D \qquad \text{(I-17)}$$

Van Everdingen and Hurst (1949) assumed homogeneity and derived analytic expressions for P_I and W_I. Frick and Taylor (1962) tabulated these functions. Observations indicate that brine reservoirs at the WIPP site have heterogeneous hydraulic properties. The brine reservoir properties are based on WIPP-12 data. These data indicate that a relatively high-permeability region k_1 located near the well serves as a collection area for a larger region having a lower permeability k_2 (Figure I.1.1).

Lappin et al. (1989, Section 3.4.3) present interpretations of the WIPP-12 brine-reservoir test data that result in two permeability regions k_1 and k_2 surrounding the borehole. The assumption is made that yet a third low-permeability zone k_3 provides an effectively infinite source of pressurized brine. Its distance $r > r_3$ is sufficiently great, however, and its permeability k_3 (equal to the permeability of the intact rock) is so small that it does not participate within the time scale of observations from the WIPP-12 field testing. For the three-zone characterization of the brine reservoir, the dimensionless permeability function assumes the form

$$k_D(r_D) = \begin{cases} k_1/k_o & 1 \le r_D \le r_{D2} \\ k_2/k_o & r_{D2} < r_D \le r_{D3} \\ k_3/k_o & r_D > r_{D3} \end{cases} \qquad \text{(I-18)}$$

The radii r_w, r_2, and r_3 are specified in Figure I.1.1. For this heterogeneous system, the reference permeability has been arbitrarily set to $k_o = k_1$. Assuming heterogeneous properties makes an analytic solution difficult. As a result, the study uses the numerical algorithms of the GTFM model (Pickens et al., 1987) to generate the desired influence functions. A tabulation of these functions provides input for SWIFT II.

Generalized Brine-Reservoir Response. The influence function W_I represents the total flow that occurs in response to a pressure drop of unity. If the pressure drop at the wellbore $\Delta p_w = \Delta p_w(r_D{=}1)$ is constant, but differs from unity, then the flow rate is $\Delta p_w W_I$. If Δp_w varies as a function of time, then the principle of superposition (Carslaw and Jaeger, 1959) yields the cumulative fluid flow

$$W_D(t_D) = \int_0^{t_D} \Delta p'_w(\lambda) W_I(t_D - \lambda) d\lambda \tag{I-19}$$

where Δp_w denotes the pressure drop at the wellbore-reservoir interface and the prime denotes differentiation with respect to the argument. Carter and Tracy (1960) approximate Equation (I-19) with a form more suitable for numerical computations by assuming a linear variation within a given time step $t_D^n \leq t_D \leq t_D^{n+1}$

$$W_D^{n+1} = W_D^n + Q_D (t_D - t_D^n) \tag{I-20}$$

where a superscript denotes the time level and Q_D represents an average rate of flow during the time step.

Carter and Tracy (1960) evaluate the flow rate Q_D by equating the right-hand sides of Equations (I-19) and (I-20). Through the use of a step-function Laplace transforms with respect to t_D the equation becomes

$$s\overline{\Delta p_w}\;\overline{W_I} = [(W_D^n - Q_D t_D^n)/s] + [Q_D/s^2] \tag{I-21}$$

where s is the Laplace-transform variable, and the bars denote transformed quantities. The analysis of Carter and Tracy becomes approximate with Equation (I-21). The identity $1/s^2 = sP_{IWI}$ (VanEverdingen and Hurst, 1949, p. 316) allows one to solve forp $_w$. Performing an inverse Laplace transform and solving the resulting equation for Q_D gives

$$Q_D = (\Delta p_w^{n+1} - W_D^n P'^{n+1}_I)/(P_I^{n+1} - t_D^n P'^{n+1}_I) \tag{I-22}$$

This equation gives the flow rate as a function of the pressure drop Δp_w at the wellbore. The injection volume W can be accumulated numerically as a function of time, and P_I and P'_I can be evaluated from tables. However, Equation (I-22) applies only to the brine reservoir. The hydraulic coupling to the Culebra is presented below.

Reservoir-Borehole-Aquifer Coupling. The following equations characterize the pressure response of the brine reservoir.

$$Q = A_I + B_I \Delta p_{bw} \tag{I-23}$$

where the subscript $_b$ is used to distinguish brine-reservoir quantities, and

$$A_I = -(Q_w/W_w)W^n P_I'^{n+1}/(P_I^{n+1} - t_D{}^n P_I'^{n+1}) \tag{I-24}$$

and

$$B_I = Q_w/(P_I^{n+1} - t_D{}^n\, P_I'^{n+1}) \tag{I-25}$$

In order to characterize the borehole, the analysis assumes a finite transmissibility T_w in the plugs and rubble. The borehole flow is governed by the equilibrium condition

$$Q= T_w\,(p_{bw} - p_w - \rho_s g\Delta h) \tag{I-26}$$

Saturated brine of density ρ_s is assumed to occupy the wellbore with a vertical distance Δh separating the centroids of the Culebra and the brine reservoir.

The static pressure difference $\Delta p_o = p_{bo} - \rho_s\, g\Delta h - p_o$ can be substituted into Equation (I-26), giving the equation

$$Q = T_w\,(\Delta p_w - \Delta p_{bw} - \Delta p_o) \tag{I-27}$$

where Δp_{bw} and Δp_w represent pressure drops of the brine reservoir and the aquifer, respectively. For the pressurized release considered here, Δp_w is inherently negative and Δp_{bw} inherently positive.

Hydraulic coupling to the Culebra focuses on the grid-block m that was penetrated by the wellbore. The pressure p of this grid block, as determined by the finite-difference method, represents an average over the pore volume V of the block. This pressure is influenced by several factors. These include the pore value of the block, its transmissive connections to neighboring blocks, and the hydraulic connection between the wellbore and the grid block. To characterize the latter, the following relation between the borehole flow and pressure differences is assumed

$$Q = M\,(p_w - p) \tag{I-28}$$

which indicates a proportionality between flow rate and pressure drop between the wellbore and the grid-block center.

M, the mobility, is given by

$$M = 2\pi(\mu_o/\mu)(K/\rho_o\, g)\Delta z/\ln(r_1/r_w) \tag{I-29}$$

where K is the hydraulic conductivity of the grid block, Δz is the thickness of the Culebra, and ρ_o and μ_o are reference values of density and viscosity, respectively. These parameters are used to convert hydraulic conductivity to permeability. The quantities ρ and μ vary as functions of the average salinity of the fluid in the grid block.

The distance r_1 of Equation (I-29) refers to the Culebra Dolomite and should not be confused with the radius (cf. Equation [I-18]) used to characterize the permeability distribution of the brine reservoir. After defining Δr as a pseudo-grid-block radius, $\Delta r = (\Delta x \; \Delta y/\pi)^{\square}$, and after determining the average pressure of the cone of influence in the Culebra Dolomite over the range $r_w \leq r \leq \Delta r$, Reeves et al. (1986, pp. 26-27) define r_1 as the radius at which the pressure of the cone of influence equals the average pressure:

$$\ln(r_1/r_w) = r_w(\,1+\Delta r/r_w)\,[\ln(\Delta r/r_w)-1]/(\Delta r-r_w) \qquad \text{(I-30)}$$

Equations (I-29) and (I-30) provide a definition of the mobility as the hydraulic conductance from the wellbore radius to the radius of the average pressure. Stated in terms of pressure drops below static pressure, Equation (I-28) can be written in the form

$$Q = M(\Delta p - \Delta p_w) \qquad \text{(I-31)}$$

Equations (I-23), (I-27), and (I-31) provide a set of three equations in the three unknowns Δp_w, Δp_{bw}, and Q. Solved simultaneously, they yield the desired relations. The flow rate injected into the Culebra can be represented as

$$Q = [A_I + B_I\,(\Delta p_o + \Delta p)]\;T/(T + B_I) \qquad \text{(I-32)}$$

The net transmissibility due to borehole-aquifer coupling is

$$T^{-1} = T_w^{-1} + M^{-1} \qquad \text{(I-33)}$$

The assumption has been made that the well skin of the brine reservoir is sufficiently high in permeability relative to T_w and M that it may be neglected in Equation (I-33).

Expressed in terms of the incremental change δ_p for time-step n, the pressure drop becomes $\Delta p = \Delta p^n - \delta p$, and the flow rate Q can be expressed in the form of Equation (I-10), where

$$A_Q = [A_I + B_I\,(\Delta p_o + \Delta p^n)]\;T/(T + B_I) \qquad \text{(I-34)}$$

and

$$B_Q = -\,TB_I/(T + B_I) \qquad \text{(I-35)}$$

Equations (I-10), (I-34), and (I-35) are the equations necessary for a determination of the flow rate.

I.1.3 CALCULATIONS FOR RADIATION EXPOSURE PATHWAYS

This subsection describes the conversion from amounts of released radionuclides to human radiation exposures (Cases IIA through IID, including Cases IIA[rev] and IIC[rev]). The 1980 FEIS analyzed the effects of radioactivity release from the WIPP through consideration of the consequences of five different hypothetical scenarios that would result in the movement of radionuclides to the biosphere. The analysis of these scenarios followed a pathway that led from radionuclide movement through the geosphere to transport through the biosphere after discharge into the Pecos River at Malaga Bend and ultimately predicted radiation doses to the people living in the area. Direct-access releases to the surface from an intrusion borehole were also included. Human dose estimates in the FEIS used information from the International

Commission on Radiological Protection (ICRP, 1959).

The SEIS concentrates on the effects of release of radioactivity from the WIPP through an estimate of the consequences of two different hypothetical cases. These are a release from an undisturbed repository (Case I) and a release as a result of a borehole passing through the repository into a pressurized brine reservoir below. Human dose estimates in this SEIS are based on the new ICRP philosophy in ICRP 26 and 30 (ICRP, 1977, and ICRP, 1979, respectively). Indications are that analyses with the new ICRP philosophy for internal dose assessment are less restrictive than the previous methods (ICRP, 1959) for about 25 percent of the radionuclides considered, more restrictive for about 25 percent, and about the same for the remaining 50 percent (Poston, 1985).

With the exception of this somewhat changed philosophy, the radionuclide-transport pathways calculations in the SEIS repeat the FEIS pathway calculations with a minimum of change. This approach responds to changes in repository design and improved understanding of local geohydrology rather than to changes in biological pathway parameters.

I.1.3.1 **Philosophy of Dose Limitations in ICRP**

The International Commission on Radiological Protection recommends a system of dose limitations based on three principles (IRCP, 1977). The first of these is that no practice shall be adopted unless it results in a net positive benefit. The second is that all exposures shall be kept as low as reasonably achievable (ALARA). The third principle is that the dose equivalent to an individual shall not exceed the ICRP recommended limits.

In addition, the ICRP also suggests two other methods of controlling exposure. It recommends controlling exposure on an annual basis through an annual dose equivalent limit and also with a "committed effective dose equivalent." This is the dose equivalent received from internally deposited material integrated over a 50-year working life. The "committed effective dose equivalent" is the concept that is used for calculating internal doses in this SEIS.

A discussion of the possible pathways for Cases IIA through IID, including Cases IIA(rev) and IIC(rev), now follows. The pathway begins as a release to the surface at the top of the intruding borehole.

I.1.3.2 **Release at the Head of the Intruding Well**

The release at the top of the intrusion well consists of two elements. A repository panel is breached by a borehole, and cuttings are removed directly from the panel. Later, the drillhole penetrates a brine reservoir in the Castile Formation and more material is brought to the surface. The time required to drill from the repository level down to the brine reservoir is about 15 hours. During this time radioactive material continues to be eroded from the consolidated waste by the swirl of the drilling fluid.

Penetration of the Castile brine pocket results in pressurized brine mixing with the drilling fluid in the borehole and flowing with it up to the wellhead. About 1,000 barrels of brine-pocket fluid are assumed to mix with the drilling fluid and recirculate through the panel to the surface. If CH TRU waste is encountered, the equivalent of three drums of consolidated wastes is removed in the form of cuttings and eroded material. If RH TRU waste is encountered, all the contents of a single RH container is brought to the surface. The drilling operation ends, the borehole is plugged and capped, and the immediate supply of radioactive material to the surface ceases.

I.1.3.3 **Geologist Exposure**

The approach used to calculate the highest individual external dose received by a member of the drilling crew is the same as that used in the FEIS Subsection 9.7.1.5. The highest individual external dose is received by a geologist who examines cuttings for a period of 1 hour at a distance of 1 meter (about 1 yard). The samples are treated as point sources with no self-shielding effects. Elements considered are plutonium-238, plutonium-239, plutonium-240, uranium-233, uranium-235, americium-241, and neptunium-237. For RH TRU waste, strontium-90 and cesium-137 are also considered.

The calculation uses the equation (USPHS, 1970)

$$Exp = 0.5 \bullet n \bullet E \bullet C \tag{I-36}$$

where Exp is the gamma exposure rate at 1-meter distance from the source (mrem/hr), n is the number of gamma quanta per disintegration, E is the gamma ray energy (MeV), and C is the activity of the sample (mCi). As indicated above, the geologist examines a sample for 1 hour. The sample is assumed to have a volume of 526 cm^3. After the disposal room is fully compacted, a single consolidated drum of CH TRU waste will occupy a volume of about 21.5 gal (81 L). The ratio of volumes implies that the sample occupies 1/155 of the consolidated drum; the radioactivity in a single sample is obtained by dividing the inventory-per-drum values by 155 (Lappin et al., 1989, Tables 5-1, 7-1). The dose to the geologist from exposure to CH TRU waste on a per sample basis is presented in Table I.1.1.

A similar calculation was made for the drill hole intercepting RH TRU waste. In this case it was assumed that the contents of the whole canister (Table B.2.12) was brought to the surface. The resulting dose to the geologist on a per sample basis is presented in Table I.1.2. The exposure at 100 years after site closure is seen to be dominated by cesium-137 at 90 mrem dose. However, because cesium-137 has only a 30-year

TABLE I.1.1 Maximum dose received by a member of the drilling crew (CH TRU waste)

Nuclide	C (mCi/sample)	E (MeV)	n (γ-q/dis)[a]	Exposure (mrem/hr-sample)
Plutonium-238	35.0	0.099	8.0×10^{-5}	1.4×10^{-4}
Plutonium-239	4.0	0.0		
Plutonium-240	1.0	0.65	2.0×10^{-7}	6.5×10^{-8}
Uranium-233	0.06	0.029	1.7×10^{-4}	1.5×10^{-7}
Uranium-235	3.2×10^{-6}	0.143	0.11	
		0.185	0.54	
		0.204	0.05	3.0×10^{-7}
Americium-241	7.1	0.06	0.36	0.077
Neptunium-237	7.3×10^{-5}	0.0		
Total				0.077

[a] γ-q/dis = gamma quanta per disintegration.
Cf. Lappin et al., 1989, Table 7-2.

TABLE I.1.2 Maximum dose received by a member of the drilling crew (RH TRU waste) at 100 years after site closure

Nuclide	C (mCi/sample)	E[a] (MeV)	n (γ-q/dis)[b]	Exposure (mrem/hr-sample)
Strontium-90	340	(no gamma)		
Cesium-137	320	0.662	.85	90
Plutonium-238	1950	0.099	8.0×10^{-5}	7.7×10^{-3}
Plutonium-239	5050	(no gamma)		
Plutonium-240	740	0.650	2.0×10^{-7}	4.8×10^{-5}
Plutonium-241	74	0.160	6.7×10^{-8}	4.0×10^{-7}
Americium-241	130	0.060	3.6×10^{-1}	1.4
Total				91

[a] From ICRP Publication 38, 1983.
[b] γ-q/dis = gamma quanta per disintegration.

half-life, its contribution to the geologist's dose falls to 1.4 mrem in just 180 years. His dose from cesium has fallen to the level of the dose from the next most important radionuclide, americium-241. These results apply to all six variants of Case II.

I.1.3.4 **Doses Received by Indirect Pathways**

The inventory in the analysis described above involves the equivalent of three CH drums or one RH canister of waste material brought to the surface during the drilling operation. The material (cuttings and particles eroded from the room contents by drilling fluid) are deposited into a settling pond at the top of the drillhole. After the drilling operations end, the radioactive material present in the settling pond is available for transport through airborne or surface-water pathways.

A ranch family hypothetically resides at a distance of 500 meters (550 yd) downwind from this settling pond. Exposure to the family is through two pathways:

- Inhalation of contaminated air

- Ingestion of foods (meat, milk, and above- and below-surface food crops) produced on the ranch.

The settling pond is assumed to be 14 ft wide, 35 ft long, and 12 ft deep. The pond contains 44,000 gal of mud and has a surface area of 500 ft^2. There is also a second pit, called the suction pit, downstream of the settling pit. The volume of these two pits totals about three times the volume of the borehole. It is assumed that all waste materials are discharged into the settling pit. Radionuclide concentrations in the dry mud pit are shown in Table I.1.3.

For example, there are 16.5 Ci of Pu-238 in the equivalent of three drums. That much Pu-238 in a volume of 44,000 gal (167 m^3), with a density of 1.4 yields

$$\frac{16.5\ \text{Ci}}{167\text{m}^3} \times \frac{\text{m}^3}{10^6\text{cm}^3} \times \frac{\text{cm}^3}{1.4\text{g}} = 7.1 \times 10^{-8}\ \text{Ci/g}$$

and when the 50 percent of the water evaporates, the concentration doubles, becoming 1.42 x 10^{-7} Ci/g.

A similar set of calculations was made to determine the amounts of different radionuclides in the mud pit, if RH TRU waste had been intercepted starting from Table B.2.12. The results are given in Table I.1.4.

TABLE I.1.3 Radionuclide concentrations in the dry mud pit from CH TRU waste contributions

Nuclide	Concentration (Ci/g)
Americium-241	2.83×10^{-8}
Neptunium-237	2.91×10^{-13}
Plutonium-238	1.42×10^{-7}
Plutonium-239	1.54×10^{-8}
Plutonium-240	3.86×10^{-9}
Uranium-233	2.57×10^{-10}
Uranium-235	1.29×10^{-14}

Cf. Lappin et al., 1989, Table 7.3.

TABLE I.1.4 Radionuclide concentrations in the dry mud pit from RH TRU waste contributions

Nuclide	Concentration (Ci/g)
Strontium-90	3.86×10^{-9}
Cesium-137	3.69×10^{-9}
Plutonium-238	2.21×10^{-8}
Plutonium-239	5.81×10^{-8}
Plutonium-240	1.87×10^{-8}
Plutonium-241	8.30×10^{-10}
Americium-241	4.51×10^{-9}

A procedure called the squared Gaussian plume model (FEIS, subsection K.3.1) was used to calculate the downwind surface air concentration at a distance of 500 m (550 yd) and the resulting dry-deposition flux. Provided the area of the mud pit is small (less than 100 square meters [120 yd^2]), the suspended material transported to distances greater than about 100 meters (110 yd) from the pit may be assumed to come from an upwind point source. The Gaussian plume model for air concentration downwind is given by the expression

$$\chi = \frac{2Q}{\sqrt{2\pi}\, 3\sigma_y \sigma_z u} \qquad \text{(I-37)}$$

where

χ = ground-level air concentration (Ci/m^3)
Q = source strength (Ci/sec)
$3\sigma_y$= lateral width of assumed uniform distribution (m)
σ_z= vertical standard deviation (m)
u = average wind speed (m/sec).

These air concentrations and deposition fluxes for CH TRU waste are shown in Table I.1.5. Table I.1.6 contains these values for RH TRU waste.

TABLE I.1.5 Air concentration and deposition flux values for CH TRU waste

Nuclide	Concentration (Ci/m^3)	Deposition Flux (Ci/m^2-s)	Deposition Flux (Ci/m^2-yr)
Americium-241	3.07×10^{-18}	3.07×10^{-20}	9.70×10^{-13}
Neptunium-237	3.16×10^{-23}	3.16×10^{-25}	9.96×10^{-18}
Plutonium-238	1.54×10^{-17}	1.54×10^{-19}	4.85×10^{-12}
Plutonium-239	1.68×10^{-18}	1.68×10^{-20}	5.29×10^{-13}
Plutonium-240	4.19×10^{-19}	4.19×10^{-21}	1.32×10^{-13}
Uranium-233	2.79×10^{-20}	2.79×10^{-22}	8.82×10^{-15}
Uranium-235	1.40×10^{-24}	1.40×10^{-26}	4.41×10^{-19}

Source: Lappin et al., 1989, Table 7-4.

TABLE I.1.6 Air concentration and deposition flux values for RH TRU waste

Nuclide	Concentration (Ci/m^3)	Deposition Flux (Ci/m^2-s)	Deposition Flux (Ci/m^2-yr)
Strontium-90	4.21×10^{-19}	4.21×10^{-21}	1.33×10^{-13}
Cesium-137	4.03×10^{-19}	4.03×10^{-21}	1.27×10^{-13}
Plutonium-238	2.41×10^{-18}	2.41×10^{-20}	7.60×10^{-13}
Plutonium-239	6.34×10^{-18}	6.34×10^{-20}	2.00×10^{-12}
Plutonium-240	2.03×10^{-18}	2.03×10^{-20}	6.42×10^{-13}
Plutonium-241	9.04×10^{-20}	9.04×10^{-22}	2.85×10^{-14}
Americium-241	4.92×10^{-19}	4.92×10^{-21}	1.55×10^{-14}

Parameters involved in these calculations include the following:

1. resuspension rate = $10^{-13} (u/u_0)^3 s^{-1}$ (u_0 = 1 m/s)
2. wind velocity = 3.73 m/s
3. density of dry drilling mud = 1.4 g/cm^3
4. mud pit surface area = 46.45 m^2
5. depth available for resuspension = 1.0 cm
6. deposition rate = 1.68×10^{-18} Ci/m^2–s
7. particle size.
8. plume vertical standard deviation = σ_z = 40.92 m
9. plume lateral standard deviation = σ_y = 57.68 m

The source area is approximated by choosing a vertical standard deviation and lateral width of the assumed Gaussian distribution and identifying a virtual point source 20.6 m (22.5 yd) upwind of the leeward side of the pit. Steady-state soil concentrations at 100 years (within 2 percent of steady state) appear in Table I.1.7 for CH TRU waste. RH TRU waste steady-state soil concentrations appear in Table I.1.8.

Transfer factors used in the dose calculations are given in Table I.1.9.

Data on human food consumption per capita are required for the four pathways. Data for the United States were taken from Till and Meyer (1983, Table 6.8). They are 508 g/day for milk, 86 g/day for meat products, 103 g/day for below-surface crops, and 202 g/day for above-surface crops. Each steer eats 15 kg of fresh forage per day.

TABLE I.1.7 Steady-state soil concentrations (CH TRU waste)

Nuclide	Concentration (Ci/kg(soil))
Americium-241	8.62×10^{-14}
Neptunium-237	8.85×10^{-19}
Plutonium-238	4.31×10^{-13}
Plutonium-239	4.70×10^{-14}
Plutonium-240	1.17×10^{-14}
Uranium-233	7.84×10^{-16}
Uranium-235	3.92×10^{-20}

Cf. Corrected from Lappin et al., 1989, Table 7-5.

TABLE I.1.8 Steady-state soil concentrations (RH TRU waste)

Nuclide	Concentration (Ci/kg(soil))
Strontium-90	1.18×10^{-14}
Cesium-137	1.13×10^{-14}
Plutonium-238	6.76×10^{-14}
Plutonium-239	1.78×10^{-13}
Plutonium-240	5.70×10^{-14}
Plutonium-241	2.54×10^{-15}
Americium-241	1.38×10^{-15}

TABLE I.1.9 Soil-to-plant and forage-to-food-product transfer factors (Case II)

	Nuclide	Soil-to-Plant (kg-soil/kg-plant)	Forage-to-Food Product (day/kg-food or day/liter-milk)
Beef:			
	Americium-241	4.2×10^{-2}	3.6×10^{-6}
	Neptunium-237	9.2×10^{-2}	5.0×10^{-6}
	Plutonium-238	1.4×10^{-2}	1.0×10^{-6}
	Plutonium-239	1.4×10^{-2}	1.0×10^{-6}
	Plutonium-240	1.4×10^{-2}	1.0×10^{-6}
	Plutonium-241	1.4×10^{-2}	1.0×10^{-6}
	Uranium-233	1.7×10^{-2}	3.4×10^{-4}
	Uranium-235	1.7×10^{-2}	3.4×10^{-4}
	Strontium-90	1.25	8.1×10^{-4}
	Cesium-137	4.8×10^{-2}	2.0×10^{-3}
Milk:			
	Americium-241	4.2×10^{-2}	2.0×10^{-5}
	Neptunium-237	9.2×10^{-2}	5.0×10^{-6}
	Plutonium-238	1.4×10^{-2}	2.7×10^{-6}
	Plutonium-239	1.4×10^{-2}	2.7×10^{-6}
	Plutonium-240	1.4×10^{-2}	2.7×10^{-6}
	Plutonium-241	1.4×10^{-2}	2.7×10^{-6}
	Uranium-233	1.7×10^{-2}	6.1×10^{-4}
	Uranium-235	1.7×10^{-2}	6.1×10^{-4}
	Strontium-90	1.25	1.4×10^{-3}
	Cesium-137	4.8×10^{-2}	7.1×10^{-3}
Dried edible below surface crops:			
	Americium-241	6.4×10^{-5}	
	Neptunium-237		
	Plutonium-238	1.4×10^{-3}	
	Plutonium-239	1.4×10^{-3}	
	Plutonium-240	1.4×10^{-3}	
	Plutonium-241	1.4×10^{-3}	
	Uranium-233	9.0×10^{-4}	
	Uranium-235	9.0×10^{-4}	
	Strontium-90	4.7×10^{-1}	
	Cesium-137	3.2×10^{-2}	
Dried edible above surface crops:			
	Americium-241	2.8×10^{-5}	
	Neptunium-237	1.5×10^{-2}	
	Plutonium-238	1.7×10^{-4}	
	Plutonium-239	1.7×10^{-4}	
	Plutonium-240	1.7×10^{-4}	
	Plutonium-241	1.7×10^{-4}	
	Uranium-233	1.0×10^{-3}	
	Uranium-235	1.0×10^{-3}	
	Strontium-90	2.2	
	Cesium-137	2.2×10^{-2}	

Cf. Lappin et al., 1989, Table 7-6.

Note. All data are from Till and Meyer (1983), Tables 5.17, 5.18, 5.36, and 5.37. Transfer factors were selected assuming that vegetables would be washed before being eaten.

The analysis used various computer codes to tabulate the committed effective dose equivalent for various body organs per unit activity inhaled or ingested. The organs included in these tabulations are those explicitly considered by the ICRP to be at risk. The committed dose equivalent is the total dose equivalent that an organ or tissue of the body is expected to receive over the 50-year period following exposure. It is recognized that in most environmental applications, more rigorous evaluation requires information on the time variation in the dose equivalent rates for the various tissues at risk. This information provides the time dependence of environmental conditions, and therefore, that of the intake could be assessed with consideration of the years of remaining life. It is also recognized that overestimates by factors of 2 to 3 in the risk are possible by not using the time-dependent nature of the organ dose equivalent rates and the years of life remaining.

Committed dose equivalent (CDE) and committed effective dose equivalent (CEDE) factors used in the analysis are shown in Table I.1.10.

Tables I.1.11 and I.1.12 list the maximum doses received by a person through indirect pathways for each nuclide of importance. These pathways include ingestion of foods provided by animals feeding on the land, as well as crops grown below and aboveground (root and leafy vegetables). The inhalation pathway assumes a breathing rate of 2.7 x 10^{-4} m^3/s. The tables summarize the exposure calculated for a person living on the hypothetical farm described in the subsection below for a 50-year committed effective dose equivalent.

I.1.3.5 **Exposure from Stock Well Water**

In addition to radiation exposure at the top of the intrusion borehole at the WIPP site itself in Case II, there is a possible exposure pathway through a stock well that taps the Culebra aquifers; a stock well that is at the closest point downstream for the salinity of its water to be low enough for cattle to drink (Subsection I.2.7 below). There is no radionuclide or stable lead release to the stock well until after 200,000 years, and hence no human exposure. The starting point for all six variants of Case II is the concentrations of radionuclides at the stock well (Table 5.68). Discharge rates and concentrations at 10,000 years are used because they are still rising at that time, which is the end of the calculation. The human exposure calculated is the exposure of a person who eats beef from those cattle.

The calculation assumes that eight cattle graze in the square mile (2.6 km^2) around the well. Each animal requires 13 gal/day (49 L/day) of water to drink. Therefore, allowing for rainfall at the rate of 20 cm/yr and evaporation at the rate of 200 cm/yr and a stock pond whose area is 139 ft^2 (0.0013 hectares), this well is pumped at the rate of 120 gal/day (460 L/day). The result is an evaporation-caused increase in radionuclide concentrations by a factor of 1.1635.

The maximally exposed individual is assumed to eat beef from the cattle at the rate of 86 g/day (NCRP, 1984, Table 5.3).

TABLE I.1.10 50-year committed dose equivalent (CDE) and committed effective dose equivalent (CEDE) factors (rem/μCi)

Nuclide	Ingestion CEDE (rem/μCi)	Inhalation CDE (rem/μCi)

Americium-241	4.5	10,000
Cesium-137	0.05	0.1
Neptunium-237	3.9	9,600
Plutonium-238	0.054	3,300
Plutonium-239	0.058	3,800
Plutonium-240	0.058	3,800
Plutonium-241	0.086	84
Strontium-90	0.012	11
Uranium-233	0.025	1,100
Uranium-235	0.025	1,000

Cf. Lappin et al., 1989, Table 7-7.

Note. All data are from DOE (1988b). The CEDE values are for the whole body; the CDE values are for critical organs. Lungs are the critical organ for uranium and strontium inhalation. The gastrointestinal tract is the critical organ for cesium inhalation. Bone is the critical organ in all other cases. The doses to the other tissues in the body are generally no more than a tenth of the doses to the body from radionuclides ingested and inhaled.

TABLE I.1.11 Maximum doses received by a person through indirect pathways for CH TRU waste

Committed Effective Dose Equivalents After a 1-Year Exposure
(mrem during the subsequent 50 years)

Nuclide	Beef	Milk	Vegetables	Root Crops	Inhalation
Americium-241	2.76×10^{-8}	9.06×10^{-7}	8.01×10^{-7}	9.36×10^{-7}	2.62×10^{-1}
Neptunium-237	7.48×10^{-13}	4.42×10^{-12}	3.82×10^{-9}		2.58×10^{-6}
Plutonium-238	1.54×10^{-10}	2.45×10^{-12}	3.00×10^{-7}	1.23×10^{-6}	4.37×10^{-1}
Plutonium-239	1.80×10^{-11}	2.86×10^{-13}	3.52×10^{-8}	1.44×10^{-7}	5.40×10^{-2}
Plutonium-240	4.49×10^{-12}	7.17×10^{-14}	8.80×10^{-9}	3.59×10^{-8}	1.35×10^{-2}
Uranium-233	5.34×10^{-11}	5.66×10^{-10}	1.45×10^{-9}	6.63×10^{-10}	2.62×10^{-4}
Uranium-235	2.66×10^{-15}	2.83×10^{-14}	7.23×10^{-14}	3.32×10^{-14}	1.19×10^{-8}
Total ingested dose:		4.43×10^{-6}			
Total inhaled dose:					7.66×10^{-1}

Cf. Corrected from Lappin et al., 1989, Table 7-8.

TABLE I.1.12 Maximum doses received by a person through indirect pathways for RH TRU waste

Committed Effective Dose Equivalents After a 1-Year Exposure
(mrem during the subsequent 50 years)

Nuclide	Beef	Milk	Vegetables	Root Crops	Inhalation
Strontium-90	6.76×10^{-8}	6.90×10^{-7}	2.30×10^{-5}	2.51×10^{-6}	3.94×10^{-5}
Cesium-137	2.55×10^{-8}	5.35×10^{-7}	9.16×10^{-7}	6.79×10^{-7}	3.43×10^{-7}
Plutonium-238	2.41×10^{-11}	3.84×10^{-10}	4.57×10^{-8}	1.92×10^{-7}	6.77×10^{-2}
Plutonium-239	6.80×10^{-11}	1.08×10^{-9}	1.29×10^{-7}	5.43×10^{-7}	2.05×10^{-1}
Plutonium-240	2.18×10^{-11}	3.48×10^{-10}	4.15×10^{-8}	1.74×10^{-7}	6.58×10^{-2}
Plutonium-241	1.44×10^{-12}	2.29×10^{-11}	2.73×10^{-9}	1.15×10^{-8}	6.47×10^{-5}
Americium-241	1.42×10^{-9}	1.49×10^{-8}	1.28×10^{-7}	1.49×10^{-7}	4.19×10^{-2}
Total ingested dose:		2.99×10^{-5}			
Total inhaled dose:					3.81×10^{-1}

Table I.1.13 shows the chain of logic leading from the concentrations of the various radionuclides in the well water to the concentrations of those radionuclides in the beef for cases IIA through IID. Table I.1.14 continues from the concentration in beef to the dose to humans, expressed as the 50-year committed dose from 1 year's consumption of that beef. Similarly, Tables I.1.15 and I.1.16 show these chains in logic for Cases IIA(rev) and IIC(rev).

Column A is from Table 5.68. The factor of 1.1635 used in going from Column A to Column C is the evaporation-caused nuclide enrichment factor. The factors in Column D that convert from the amount of water the steer drinks to the concentration of a radionuclide in his flesh are from Baes et al., 1984. These are actually for the forage-to-beef pathway, used here because of the lack of any similar table for the water-to-beef pathway, and as recommended in NRC Regulatory Guide 1.109 (NRC, 1976). The conversion factors of Column G are from tables for individual radionuclides in DOE/EH-0071 (DOE, 1988b). These last factors allow for all the steps from the ingestion of beef to the resultant committed effective dose equivalent, including the amount of the nuclide excreted. A similar logic applies to Tables I.1.15 and I.1.16.

The totals listed in Tables I.1.13 through I.1.16 assume that the cattle have been drinking from the stock well long enough to come to equilibrium with the radionuclides in their water. (That is, the calculations use meat transfer coefficients [Column D, Table I.1.13] that assume that steady-state conditions have been reached [Baes et al., 1984].) As the cattle continue to use this water, the radionuclide concentrations in their muscle tissue build up according to the factor

$$1 - \exp(-\lambda t)$$

where λ is equal to 1n $2/T_{1/2}$, $T_{1/2}$ being the effective or biological half-life of the radionuclide in muscle tissue, and t is the length of time the animal uses the contaminated water.

The value used by the Nevada Applied Ecology Group for the biological half-life of ^{239}Pu in muscle is 2,000 days (Martin and Bloom, 1980). The Environmental Evaluation Group suggests a value of 200 days for t (Neill, 1989). The build-up factor then becomes 0.067.

Using the larger of these two factors (0.067), and assuming the same factors apply to other radionuclides as well, the total of 27.8 mrem shown in Table I.1.16 for Case IIC(rev) reduces to 1.9 mrem.

Finally then, this 1.9 mrem dose is a 50-year committed effective dose equivalent. If the individual eats this beef for only 1 year, he or she would receive an average annual exposure of 0.4 mrem, which is approximately 1/2700 the 100-mrem average annual background present in the United States. However, this individual will continue to eat beef. It is standard procedure to calculate the total dose equivalent for radionuclides deposited in the body that will occur over a 50-year period. This is reported in the year that the radionuclide is ingested. On this basis, a committed effective dose equivalent of 1.9 mrem is about 2 percent of background. (None of the exposures in Figure 5.16, Tables 5.63, 5.64 or I.1.13 through I.1.16 include this non-equilibrium factor of 0.067.)

TABLE I.1.13 Steps in the calculation of human exposure: from radionuclide concentrations in the stock well water to their concentrations in beef (Cases IIA, IIB, IIC, and IID)

Nuclide	A Concentration in well kg (nuclide)/ kg (brine)	B Specific activity (Ci/g)	C Concentration in pond (Ci/L)	D Conversion factor (d/kg)	E Concentration in beef (Ci/kg)
			Case IIA		
Pb-210	$7.61x10^{-19}$	$7.63x10^{1}$	$7.43x10^{-14}$	$3.0x10^{-4}$	$1.11x10^{-15}$
Ra-226	$5.46x10^{-17}$	1.0	$6.99x10^{-14}$	$2.5x10^{-4}$	$8.73x10^{-16}$
Th-230	$8.21x10^{-23}$	$2.02x10^{-2}$	$2.12x10^{-21}$	$6.0x10^{-6}$	$6.37x10^{-25}$
U-234	$1.68x10^{-18}$	$6.25x10^{-3}$	$1.34x10^{-17}$	$2.0x10^{-4}$	$1.34x10^{-19}$
			Case IIB		
Np-237	$8.37x10^{-9}$	$7.05x10^{-4}$	$7.55x10^{-9}$	$5.5x10^{-5}$	$2.08x10^{-11}$
Pb-210	$1.20x10^{-13}$	$7.63x10^{1}$	$1.17x10^{-8}$	$3.0x10^{-4}$	$1.76x10^{-10}$
Pu-239	$8.36x10^{-10}$	$6.22x10^{-2}$	$6.66x10^{-8}$	$5.0x10^{-7}$	$1.66x10^{-12}$
Pu-240	$1.07x10^{-10}$	$2.28x10^{-1}$	$3.13x10^{-8}$	$5.0x10^{-7}$	$7.83x10^{-13}$
Ra-226	$8.63x10^{-12}$	1.0	$1.10x10^{-8}$	$2.5x10^{-4}$	$1.38x10^{-10}$
Th-229	$3.65x10^{-11}$	$2.13x10^{-1}$	$9.95x10^{-9}$	$6.0x10^{-6}$	$2.99x10^{-12}$
Th-230	$9.01x10^{-12}$	$2.02x10^{-2}$	$2.33x10^{-10}$	$6.0x10^{-6}$	$6.99x10^{-14}$
U-233	$2.92x10^{-7}$	$9.68x10^{-3}$	$3.61x10^{-7}$	$2.0x10^{-4}$	$3.61x10^{-9}$
U-234	$7.94x10^{-9}$	$6.25x10^{-3}$	$6.35x10^{-8}$	$2.0x10^{-4}$	$6.35x10^{-10}$
U-236	$7.71x10^{-9}$	$6.47x10^{-5}$	$6.39x10^{-9}$	$2.0x10^{-4}$	$6.39x10^{-12}$
			Case IIC		
Np-237	$2.98x10^{-8}$	$7.05x10^{-4}$	$2.69x10^{-8}$	$5.5x10^{-5}$	$7.40x10^{-11}$
Pb-210	$4.15x10^{-14}$	$7.63x10^{1}$	$4.05x10^{-9}$	$3.0x10^{-4}$	$6.07x10^{-11}$
Pu-239	$4.14x10^{-14}$	$6.22x10^{-2}$	$3.29x10^{-12}$	$5.0x10^{-7}$	$8.24x10^{-17}$
Pu-240	$2.32x10^{-14}$	$2.28x10^{-1}$	$6.77x10^{-12}$	$5.0x10^{-7}$	$1.69x10^{-16}$
Ra-226	$2.98x10^{-12}$	1.0	$3.81x10^{-9}$	$2.5x10^{-4}$	$4.76x10^{-11}$
Th-229	$1.58x10^{-11}$	$2.13x10^{-1}$	$4.30x10^{-9}$	$6.0x10^{-6}$	$1.29x10^{-12}$
Th-230	$3.57x10^{-12}$	$2.02x10^{-2}$	$9.22x10^{-11}$	$6.0x10^{-6}$	$2.77x10^{-14}$
U-233	$8.59x10^{-8}$	$9.68x10^{-3}$	$1.06x10^{-6}$	$2.0x10^{-4}$	$1.06x10^{-8}$
U-234	$2.86x10^{-8}$	$6.25x10^{-3}$	$2.29x10^{-7}$	$2.0x10^{-4}$	$2.29x10^{-9}$
U-236	$8.84x10^{-9}$	$6.47x10^{-5}$	$7.32x10^{-10}$	$2.0x10^{-4}$	$7.32x10^{-12}$
			Case IID		
Np-237	$2.57x10^{-10}$	$7.05x10^{-4}$	$2.32x10^{-10}$	$5.5x10^{-5}$	$6.38x10^{-13}$
Pb-210	$1.46x10^{-15}$	$7.63x10^{1}$	$1.43x10^{-10}$	$3.0x10^{-4}$	$2.14x10^{-12}$
Pu-239	$6.58x10^{-13}$	$6.22x10^{-2}$	$5.24x10^{-11}$	$5.0x10^{-7}$	$1.31x10^{-15}$
Pu-240	$3.83x10^{-13}$	$2.28x10^{-1}$	$1.12x10^{-10}$	$5.0x10^{-7}$	$2.80x10^{-15}$
Ra-226	$1.05x10^{-13}$	1.0	$1.34x10^{-10}$	$2.5x10^{-4}$	$1.68x10^{-12}$
Th-229	$1.52x10^{-13}$	$2.13x10^{-1}$	$4.13x10^{-11}$	$6.0x10^{-6}$	$1.24x10^{-14}$
Th-230	$1.20x10^{-13}$	$2.02x10^{-2}$	$3.10x10^{-12}$	$6.0x10^{-6}$	$9.31x10^{-16}$
U-233	$2.55x10^{-10}$	$9.68x10^{-3}$	$3.16x10^{-9}$	$2.0x10^{-4}$	$3.16x10^{-11}$
U-234	$2.56x10^{-10}$	$6.25x10^{-3}$	$2.04x10^{-9}$	$2.0x10^{-4}$	$2.04x10^{-11}$
U-236	$7.40x10^{-11}$	$6.47x10^{-5}$	$6.12x10^{-12}$	$2.0x10^{-4}$	$6.12x10^{-14}$

Column C = A x B x 1100(g/L) x 1.1635
Column E = C x D x 50(L/d)

TABLE I.1.14 Steps in the calculation of human exposure: from radionuclide concentrations in beef to committed dose to humans (Cases IIA, IIB, IIC, and IID)

Nuclide	E Concentration in beef (Ci/kg)	F Ingestion rate (Ci/d)	G CEDE (rem/μCi)	H Committed dose (mrem/yr of exposure)
Case IIA (Total = $2.09x10^{-4}$)				
Pb-210	$1.11x10^{-15}$	$9.58x10^{-17}$	5.1	$1.78x10^{-4}$
Ra-226	$8.73x10^{-16}$	$7.51x10^{-17}$	1.1	$3.02x10^{-5}$
Th-230	$6.37x10^{-25}$	$5.48x10^{-26}$	$5.3x10^{-1}$	$1.06x10^{-14}$
U-234	$1.34x10^{-19}$	$1.15x10^{-20}$	$2.6x10^{-1}$	$1.1x10^{-9}$
Case IIB (Total = $7.2x10^{1}$)				
Np-237	$2.08x10^{-11}$	$1.79x10^{-12}$	3.9	2.54
Pb-210	$1.76x10^{-10}$	$1.52x10^{-11}$	5.1	2.82
Pu-239	$1.66x10^{-12}$	$1.43x10^{-13}$	4.3	$2.25x10^{-1}$
Pu-240	$7.83x10^{-13}$	$6.74x10^{-14}$	4.3	$1.06x10^{-1}$
Ra-226	$1.38x10^{-10}$	$1.19x10^{-11}$	1.1	4.77
Th-229	$2.99x10^{-12}$	$2.57x10^{-13}$	3.5	$3.28x10^{-1}$
Th-230	$6.99x10^{-14}$	$6.01x10^{-15}$	$5.3x10^{-1}$	$1.16x10^{-3}$
U-233	$3.61x10^{-9}$	$3.11x10^{-10}$	$2.7x10^{-1}$	$3.06x10^{1}$
U-234	$6.35x10^{-10}$	$5.46x10^{-11}$	$2.6x10^{-1}$	5.18
U-236	$6.39x10^{-12}$	$5.49x10^{-13}$	$2.5x10^{-1}$	$5.01x10^{-2}$
Case IIC (Total = $1.29x10^{2}$)				
Np-237	$7.40x10^{-11}$	$6.37x10^{-12}$	3.9	9.06
Pb-210	$6.07x10^{-11}$	$5.22x10^{-12}$	5.1	9.72
Pu-239	$8.24x10^{-17}$	$7.08x10^{-18}$	4.3	$1.11x10^{-5}$
Pu-240	$1.69x10^{-16}$	$1.46x10^{-17}$	4.3	$2.28x10^{-5}$
Ra-226	$4.76x10^{-11}$	$4.09x10^{-12}$	1.1	1.64
Th-229	$1.29x10^{-12}$	$1.11x10^{-13}$	3.5	$1.42x10^{-1}$
Th-230	$2.77x10^{-14}$	$2.38x10^{-15}$	$5.3x10^{-1}$	$4.60x10^{-4}$
U-233	$1.06x10^{-8}$	$9.15x10^{-10}$	$2.7x10^{-1}$	$9.02x10^{1}$
U-234	$2.29x10^{-9}$	$1.97x10^{-10}$	$2.6x10^{-1}$	$1.87x10^{1}$
U-236	$7.32x10^{-12}$	$6.30x10^{-13}$	$2.5x10^{-1}$	$5.75x10^{-2}$
Case IID (Total = $9.15x10^{-1}$)				
Np-237	$6.38x10^{-13}$	$5.49x10^{-14}$	3.9	$7.81x10^{-2}$
Pb-210	$2.14x10^{-12}$	$1.84x10^{-13}$	5.1	$3.43x10^{-1}$
Pu-239	$1.31x10^{-15}$	$1.13x10^{-16}$	4.3	$1.77x10^{-4}$
Pu-240	$2.80x10^{-15}$	$2.40x10^{-16}$	4.3	$3.77x10^{-4}$
Ra-226	$1.68x10^{-12}$	$1.44x10^{-13}$	1.1	$5.79x10^{-2}$
Th-229	$1.24x10^{-14}$	$1.07x10^{-15}$	3.5	$1.36x10^{-3}$
Th-230	$9.31x10^{-16}$	$8.00x10^{-17}$	$5.3x10^{-1}$	$1.55x10^{-5}$
U-233	$3.16x10^{-11}$	$2.71x10^{-12}$	$2.7x10^{-1}$	$2.67x10^{-1}$
U-234	$2.04x10^{-11}$	$1.76x10^{-12}$	$2.6x10^{-1}$	$1.67x10^{-1}$
U-236	$6.12x10^{-14}$	$5.27x10^{-15}$	$2.5x10^{-1}$	$4.81x10^{-4}$

Column F = E x 0.086(kg/d)
Column H = F x G x 365(day) x 1000(mrem/rem) x 1,000,000 (μCi/Ci)

TABLE I.1.15 Steps in the calculation of human exposure: from radionuclide concentrations in the stock well water to their concentrations in beef (Cases IIA[rev] and IIC[rev])

Nuclide	A Concentration in well kg (nuclide)/ kg (brine)	B Specific activity (Ci/g)	C Concentration in pond (Ci/L)	D Conversion factor (d/kg)	E Concentration in beef (Ci/kg)
			Case IIA(rev) (Total = 7.86×10^{-7})		
Np-237	4.91×10^{-20}	7.05×10^{-4}	4.03×10^{-20}	5.5×10^{-5}	1.11×10^{-22}
Pb-210	3.12×10^{-21}	7.63×10^{1}	2.77×10^{-16}	3.0×10^{-4}	4.15×10^{-18}
Ra-226	2.40×10^{-19}	1.00×10^{0}	2.79×10^{-16}	2.5×10^{-4}	3.49×10^{-18}
U-233	3.00×10^{-22}	9.68×10^{-3}	3.38×10^{-21}	2.0×10^{-4}	3.38×10^{-23}
U-234	2.67×10^{-22}	6.25×10^{-3}	1.94×10^{-21}	2.0×10^{-4}	1.94×10^{-23}
U-236	3.02×10^{-22}	6.47×10^{-5}	2.27×10^{-23}	2.0×10^{-4}	2.27×10^{-25}
			Case IIC(rev) (Total = 27.8)		
Np-237	2.01×10^{-9}	7.05×10^{-4}	1.65×10^{-9}	5.5×10^{-5}	4.53×10^{-12}
Pb-210	7.80×10^{-14}	7.63×10^{1}	6.93×10^{-9}	3.0×10^{-4}	1.04×10^{-10}
Pu-239	6.54×10^{-10}	6.22×10^{-2}	4.73×10^{-8}	5.0×10^{-7}	1.18×10^{-12}
Pu-240	2.34×10^{-11}	2.28×10^{-1}	6.21×10^{-9}	5.0×10^{-7}	1.55×10^{-13}
Ra-226	6.12×10^{-12}	1.00×10^{0}	7.12×10^{-9}	2.5×10^{-4}	8.91×10^{-11}
Th-229	1.33×10^{-11}	2.13×10^{-1}	3.30×10^{-9}	6.0×10^{-6}	9.91×10^{-13}
Th-230	4.37×10^{-12}	2.02×10^{-2}	1.03×10^{-10}	6.0×10^{-6}	3.08×10^{-14}
U-233	6.29×10^{-9}	9.68×10^{-3}	7.08×10^{-8}	2.0×10^{-4}	7.08×10^{-10}
U-234	2.05×10^{-9}	6.25×10^{-3}	1.49×10^{-8}	2.0×10^{-4}	1.49×10^{-10}
U-236	3.47×10^{-9}	6.47×10^{-5}	2.61×10^{-10}	2.0×10^{-4}	2.61×10^{-12}

Column C = A x B x 1,000(g/L) x 1.1635
Column E = C x D x 50 (L/d)

TABLE I.1.16 Steps in the calculation of human exposure: from radionuclide concentrations in beef to committed dose to humans (Cases IIA[rev] and IIC[rev])

Nuclide	E Concentration in beef (Ci/kg)	F Ingestion rate (Ci/d)	G CEDE (rem/μCi)	H Committed dose (mrem/yr of exposure)
		Case IIA(rev) (Total = 7.86 x 10^{-7})		
Np-237	1.11 x 10^{-22}	9.53 x 24^{-24}	3.9 x 10^{0}	1.36 x 10^{-11}
Pb-210	4.15 x 10^{-18}	3.57 x 10^{-19}	5.1 x 10^{0}	6.65 x 10^{-7}
Ra-226	3.49 x 10^{-18}	3.00 x 19^{-19}	1.1 x 10^{0}	1.21 x 10^{-7}
U-233	3.38 x 10^{-23}	2.91 x 24^{-24}	2.7 x 10^{-1}	2.86 x 10^{-13}
U-234	1.94 x 10^{-23}	1.67 x 24^{-24}	2.6 x 10^{-1}	1.58 x 10^{-13}
U-236	2.27 x 10^{-25}	1.96 x 26^{-26}	2.5 x 10^{-1}	1.78 x 10^{-15}
		Case IIC(rev) (Total = 27.8)		
Np-237	4.53 x 10^{-12}	3.90 x 10^{-13}	3.9 x 10^{0}	5.55 x 10^{-1}
Pb-210	1.04 x 10^{-10}	8.93 x 10^{-12}	5.1 x 10^{0}	1.66 x 10^{1}
Pu-239	1.18 x 10^{-12}	1.02 x 10^{-13}	4.3 x 10^{0}	1.60 x 10^{-1}
Pu-240	1.55 x 10^{-13}	1.33 x 10^{-14}	4.3 x 10^{0}	2.09 x 10^{-2}
Ra-226	8.91 x 10^{-11}	7.66 x 10^{-12}	1.1 x 10^{0}	3.07 x 10^{0}
Th-229	9.91 x 10^{-13}	8.52 x 10^{-14}	3.5 x 10^{0}	1.09 x 10^{-1}
Th-230	3.08 x 10^{-14}	2.65 x 10^{-15}	5.3 x 10^{-1}	5.12 x 10^{-4}
U-233	7.08 x 10^{-10}	6.09 x 10^{-11}	2.7 x 10^{-1}	6.00 x 10^{0}
U-234	1.49 x 10^{-10}	1.28 x 10^{-11}	2.6 x 10^{-1}	1.22 x 10^{0}
U-236	2.61 x 10^{-12}	2.24 x 10^{-13}	2.5 x 10^{-1}	2.05 x 10^{-2}

Column F = E x 0.086 (kg/d)
Column H = F x G x 365 (day) x 1000 (mrem/rem) x 1,000,000 (μCi/Ci)

I.1.4 CALCULATIONS FOR CHEMICAL EXPOSURE PATHWAYS

As discussed in Subsection 5.4.2.2, lead is used as an indicator chemical parameter for the purpose of evaluating potential risks associated with the hazardous chemical component of TRU waste during the long-term (i.e., 10,000 years) performance of the WIPP. Unlike organic compounds that degrade and radionuclides that decay with time, metals will always be present in the waste. The initial concentration of metals in the waste will not change, although the prevalent chemical species may be altered with time due to changes in the repository environment. Lead is the principal metal in the waste (WEC, 1989) and its solubility is not expected to be limited by its initial concentration.

Thermodynamic data and information on stable solid phase equilibrium chemistry for other RCRA-regulated metals in brine are not available, and therefore they cannot be evaluated as lead is. Also, the scientific literature lacks information on the types and rates of reactions (e.g., radiolysis and biodegradation) in salt that would influence long-term behavior of organic chemicals in the WIPP.

I.1.4.1 Lead Solubility in WIPP Composite Brine

The concentration of heavy metals in solution is controlled by the solubility of various oxides, carbonates, sulfates, and sulfides. The solid and aqueous species present in aqueous systems is dependent on oxidation-reduction reactions (measured in terms of Eh) and acid-base reactions (measured in terms of pH). A system has reached equilibrium when forward reactions just balance reverse reactions. When substances are mixed, such as when brine comes in contact with the TRU waste in the repository, they may undergo chemical changes. In natural systems, a final equilibrium is probably never attained because chemical reactions occur at different rates and the environment may be changed by a process that alters the chemistry of the system. Regardless of the rate at which equilibrium is attained, equilibrium relationships are useful for predicting chemical changes that can or cannot occur.

The concentration of lead in WIPP composite brine (Abitz et al., 1989) was calculated using the Pitzer equations employed in the EQ3NR solubility/speciation computer code (Wolery, 1983; Jackson, 1988) by equilibrating the native brine with the mineral anglesite ($PbSO_4$) at pH = 6.1, Eh = 411 mV, T = 27°C. Eh was constrained by the NH, NO redox couple. For a system defined by these parameters, and equal concentrations of CO_3^{--} and SO_4^{--}, cerussite ($PbCO_3$) is the predicted stable phase (Brookins, 1988). However, anglesite ($PbSO_4$) was used in the solubility model because both Pb^{++} and CO_3^{--} are not mutually present in any one of the thermodynamic data bases accessed by the EQ3NR code. This presents no critical problem when evaluating solubility models for WIPP composite brine because the total inorganic carbon (estimate of CO_3^{--}) rarely exceeds 5 mg/L, whereas SO_4^{--} averages 17,000 mg/L. Therefore, it is assumed that the activity of CO_3^{--} in the brine is negligible.

Table I.1.17 contains the element concentrations entered into the EQ3NR code for the brine solubility simulation. The calculated lead concentrations, with mean ionic values for log $a(\pm)$, are shown in Table I.1.18. The solubility for lead is 116 mg/L. When using the solubility values, it should be kept in mind that they represent the maximum concentrations that can exist in solution. Actual concentrations are influenced by several factors, including dissolution rates and available surface area. The log a(±) values indicate which species complexes are likely to be present in the brine, with the most dominant complexes having the number closest to zero (e.g., PbCl species for Pb). Unfortunately, Pitzer's equations cannot evaluate the relative concentrations among species of an ion pair (e.g., $PbCl^+$ versus $PbCl_2$), so the charge on the dominant species cannot be predicted. For the purposes of further calculations, the charge on the dominant species is assumed to be zero.

TABLE I.1.17 Element concentrations entered into the EQ3NR code: brine solubility calculations

Element or complex	Concentration (mg/L)
Br^-	1,380
Cl^-	194,000
F^-	6.4
I^-	11
NO_3^-	11
SO_4^{--}	17,000
B	1,480
Ca^{++}	328
K^+	18,100
Mg^{++}	18,200
Mn^{++}	1.21
Na^+	83,400
NH_4^+	136
Sr^{++}	1.7

TABLE I.1.18 Ionic species and total lead solubility in WIPP composite brine

Ionic Species	Log a($\pm$)
Pb^{++} Cl^{-}	-1.20
Pb^{++} Br^{-}	-2.78
Pb^{++} SO_4^{--}	-3.90
Pb^{++} I^{-}	-4.21
Pb^{++} F^{-}	-4.40

Total Pb = 116.3 mg/L

I.1.4.2 **Modeling Assumptions for Calculating Lead Solubility in Culebra Groundwater**

Aqueous speciation/solubility calculations with the EQ3NR code (Wolery, 1983) were performed to estimate the lead solubility in Culebra groundwaters (using representative samples from wells H-2a, H-3b and H-14 in Randall et al., 1988). In addition to lead, the elements active in this problem were boron, carbon, calcium, chlorine, fluorine, iron, hydrogen, potassium, magnesium, manganese, sodium, oxygen, sulfur, silicon, and strontium. The number of aqueous species (178-194) and minerals (208-225) that areactive in a given problem is unique for each groundwater composition. However, the number of gas species was seven in all three cases. The solubility model is based on the following assumptions:

1. Cerussite is the stable solid phase for lead under the indicated temperature, pressure, Eh (oxidation-reduction state), and pH of the system.

2. The system oxidation-reduction reactions are considered in equilibrium with the entered Eh value based on platinum electrode measurements.

3. Thermodynamic equilibrium is evaluated with the B-dot equations of Helgeson (1969), which are applicable to solutions with ionic strengths no greater than about one molal (moles/kg H_2O).

4. No reaction-rate or biological kinetics are considered.

Assumptions 2 and 4 are necessary because of limitations in the available data. The first assumption is based on theoretical calculations of stable phases in an aqueous solution containing equal concentrations of sulfate, carbonate, and the cation of interest (lead) at a temperature of 25°C and a pressure of 1 atmosphere (Brookins, 1988).
Culebra groundwaters used in these models have ionic strengths up to 0.9 molal, which is near the indicated upper limit for valid use in the thermodynamic equilibrium equations. Solubility values reported here represent the maximum concentrations that can exist in a solution equilibrated with the indicated pure solid phases. It should be noted, however, that natural ground waters rarely equilibrate with pure solid phases (e.g., $PbCO_3$). This is especially true for

sulfate and carbonate minerals, which show extensive solid solution with calcium, magnesium, manganese, iron, zinc, and barium. If mineral solid solutions were equilibrated with the aqueous fluid, slightly lower solubilities would probably be calculated for lead. EQ3NR has the capability to model this type of scenario for carbonate solid solutions containing calcium, magnesium, manganese, iron, and zinc, but not lead. Therefore, the solubility values derived from pure mineral phases are the maximum concentrations that can exist in the solution, but not necessarily the actual concentrations.

I.1.4.3 **Lead Solubility in Culebra Groundwaters**

Groundwater in the Culebra has been sampled from several wells located within the 16-square-mile WIPP boundary and analyses from three of these (wells H-2a, H-3c, and H-14) are used in this evaluation. These wells were selected because they are generally to the south, in the direction of the hypothetical stock well location. The maximum concentration of lead that can occur in the Culebra groundwater obtained from each well was calculated using the EQ3NR code with the Debye-Huckel B-Dot equations (Wolery, 1983) by equilibrating the groundwater with anglesite ($PbSO_4$). Anglesite is the predicted stable phase at 25°C and 1 atmosphere (Brookins, 1988) for the values of Eh and pH listed in Table I.1.19. Table I.1.19 also contains the element concentrations entered into the EQ3NR code and the total element and aqueous species concentrations. The solubility range of lead is 52.7 to 54.4 mg/L. These solubility values represent the maximum concentrations that can exist in solution. Actual concentrations, as previously noted, are influenced by several factors, including dissolution and precipitation rates.

The model results indicate that lead solubility in Culebra groundwater is not increased to a large degree with increasing chloride concentration (e.g., well H-2a versus well H-3b). The dominant lead species in the Culebra groundwater was calculated to be uncharged $PbCO_3$.

I.1.4.4 **Health Effects Associated with Stable Lead from Wind Dispersion**

As described in Cases IIA and IIB (Subsection 5.4.2.3), drilling mud containing TRU waste constituents is brought to the surface in the scenario involving oil and gas exploration. These drilling fluids and associated cuttings are assumed to be disposed of in a mud pond located at the site. The abandoned mud pond eventually dries and its contents are subject to wind erosion.

TABLE I.1.19 Element concentrations entered into the EQ3NR code and Pb solubility for the dominant aqueous species: Culebra solubility calculations

Code Inputs

Parameter	Well H-2a	Well H-3b	Well H-14
T (°C)	22.4	22.4	22.0
Eh (mv)	60	199	70
pH	7.8	7.3	7.7

Concentration (mg/L)

Element or Complex	Well H-2a	Well H-3b	Well H-14
Cl^-	4800	27800	8200
Br^-	bdl[a]	27.5	14
F^-	2.1	1.6	0.8
HCO_3^-	54	47	40
SO_4^{--}	2900	4800	1500
$B(OH)_3$	57	137	63
Ca^{++}	670	1300	1800
Fe^{++}	0.42	0.14	0.4
K^+	100	450	250
Mg^{++}	160	830	530
Mn^{++}	0.07	0.14	bdl[a]
Na^+	2600	17000	3300
SiO_2(aq)	13.5	13	14
Sr^{++}	9.8	31.5	31

Resultant Output

	Well H-2a		Well H-3b		Well H-14	
Solid Phase	Aqueous Species	Concentration (mg/L)	Aqueous Species	Concentration (mg/L)	Aqueous Species	Concentration (mg/L)
$PbCO_3$	Total Pb	53.4	Total Pb	54.4	Total Pb	52.8
	$PbCO_3$	68.8	$PbCO_3$	67.9	$PbCO_3$	67.8
			$PbCl^+$	1.0		
			$PbCl_2$	0.5		

[a] bdl = below detection limit of analytical method

No exposure to humans from stable lead contained in the mud is expected prior to the mud drying. The dried mud, however, is subject to wind erosion and dispersion of lead particulates. Human exposure to the lead occurs through inhalation of airborne particulates.

The methodology used to calculate the dispersion of particulates in air is the same as that described in the FEIS, Appendix K. Additional assumptions used in calculating the air transport of particulates containing stable lead in this scenario are:

- The surface area of the mud pond is 500 ft^2 (46.45 m^2).
- The mud pond contains 22,000 gallons of dried mud, including 6 kg of stable lead (i.e., the equivalent of 3 drums of waste with an average lead content of 2 kg each).
- The resuspension rate of one-micron particulates from the mud pond is 5.19×10^{-12} s^{-1}.
- The exposed individual (i.e., receptor) is 570 yd (521 m) downwind from a virtual source, 21 m upwind of the center of the mud pond.

The particulate deposition velocity is 0.01 m/sec, resulting in a calculated ground deposition rate of 5.16×10^{-17} g/m^2-s.

Using these assumptions, the calculated ambient air concentration of stable lead at the downwind receptor location is $5.16 \times 10^{-9} \mu g/m^3$. This calculation is shown in Table I.1.20. The amount of ground surface deposition at the same location over a 1-year time period is 1.63×10^{-9} g/m^2.

The potential exposed exposed individual was assumed to weigh 70 kg, and a daily respiratory volume of 20 m^3/day was assumed (EPA, 1986). The rate of lead deposition in the lungs was assumed to be 50 percent of the particles inhaled, while up to 70 percent of this deposited lead was assumed to be absorbed (ATSDR, 1988), resulting in a transfer coefficient of 0.35 (i.e., 70 percent x 50 percent). The calculated daily intake of lead by an exposed individual is compared to the acceptable daily intake levels for chronic exposure (AIC). The calculated AIC-based hazard index as described in EPA (1986) is used for determination of potential risk to human health. The equations used to calculate lead uptake by humans are provided in Tables I.1.20 and I.1.21.

The daily intake of lead by humans in this scenario, using the calculated air concentration of $5.16 \times 10^{-9} \mu g/m^3$, is 5.16×10^{-13} mg/kg-day. The daily intake can be compared to the acceptable level for chronic intake (AIC) (EPA, 1986). This acceptable level is 4.3×10^{-4} mg/kg-day. The calculated hazard index for lead is therefore $5.16 \times 10^{-13}/4.3 \times 10^{-4} = 1.2 \times 10^{-9}$. This value is considerably less than unity, indicating that the intake of stable lead is well below the acceptable reference level. The dose calculated for ingestion represents the most direct and, therefore, the highest intake of lead by an exposed individual. Because of the small quantity of lead deposited on the ground surface (i.e., 1.63×10^{-9} g/m^2) and the even smaller amounts potentially taken up by animals and plants, it can be assumed that all other potential exposure pathways in this scenario (e.g., ingestion of vegetables, milk and meat) will be orders of magnitude below health-based levels. These results apply to all six variants of Case II.

TABLE I.1.20 Calculation of the lead ambient air concentration at the exposed individual location, human lead intake via inhalation, and lead hazard index for humans

Equation 1: Calculation of the lead ambient air concentration at exposed individual location

$$X = \frac{2C\, d_o\, A\, K\, \Omega\, (10^4)}{\sqrt{2\,\pi\, 3\, \Gamma_y\, \Gamma_z}\; u}$$

Where:

C = mud density (2.0 g/cm^3)

d_o = depth available for resuspension (1 cm)

A = area of mud pit (46.45 m^2)

K = resuspension rate (5.065 x 10^{-12} s^{-1})

(10^4) = conversion cm^2/m^2

Ω = 3.6 x 10^{-5} g/g (concentration of Pb in dried mud pit)

Γ_y = 57.68 m

Γ_z = 40.92 m

u = average wind speed (3.7 m/s)

Calculations:

$$X = \frac{2 \times 2 \times 1 \times 46.45 \times 5.065 \times 10^{-12} \times 3.6 \times 10^{-5} \times 10^{4}}{\sqrt{2\pi \times 3(57.68) \times 40.92 \times 3.7}}$$

X = 5.16 x 10^{-15} g/m^3 or 5.16 x 10^{-9} μg/m^3

TABLE I.1.20 Concluded

Equation 2: Calculation of human lead intake via inhalation

I_r = $(C_{Ai})(RV)(T_{A1})/A(W_A)$

I_r = daily Pb intake (mg/kg-day)

C_{Ai} = [Pb] in air (μg/m^3)

RV = daily respiratory volume (m^3/day)

T_{A1} = transfer coefficient across lungs

A = conversion factor (μg/mg)

W_A = average adult body weight (kg)

Assumptions:

RV = 20 m^3/day (EPA, 1986)

T_{A1} = (50% deposited in lungs)(70% absorbed) = 0.35 (ATSDR,1988)

A = 1000

W_A = 70 kg

C_{Ai} = $5.16 \times 10^{-9} \mu$g/m^3 (from Equation 1)

Calculations:

I_r = $\{(5.16 \times 10^{-9})(20)(.35)\}/(1000)(70)$

= 5.16×10^{-13} mg/kg-day

Equation 3: Human Hazard Index (HI)

HI = I_r/AIC

AIC = 4.3×10^{-4} mg/kg-day (EPA, 1986)

5.16×10^{-13}mg/kg-day/4.3×10^{-4}mg/kg-day

= 1.20×10^{-9}

TABLE I.1.21 Calculation of lead intake by humans, lead concentration in beef, lead intake by humans via beef ingestion, and human hazard index, Case IIC(rev)

Equation 1: Lead Intake by Beef Cattle

$I_c = (C_{w_i})(GPF)(Q_w)/(W_c)$ (modified from Whelan et al., 1987)

Where:

I_c = intake per day per steer (mg/kg-day)
C_{w_i} = lead concentration in water (mg/L)
W_c = steer average body weight (kg)
GPF = gut partitioning factor
Q_w = intake of water by cattle (L/day)

Assumptions:

C_{w_i} = 1.50 mg/L (See Subsection 5.4.2.6)

GPF = 0.15 (ATSDR, 1988)

Q_w = 49 L/day

W_c = 400 kg (Merck, 1979)

Calculations:

$I_c = C_{w_i}(0.15)(49)/400$

$= 1.84 \times 10^{-2}\, C_{w_i}$

$= (1.84 \times 10^{-2})(1.50) = 0.028$ mg/kg-day

Adult cattle will tolerate 6 mg/kg-day for 2-3 years (Botts, 1977)

TABLE I.1.21 Continued

Equation 2: Lead Concentration in Beef

$$C_{w_{im}} = C_{w_i} F_{m_i} f_w Q_w \exp[-\beta_{w_i} th_m] \text{ (Whelan et al., 1987)}$$

Where:

$C_{w_{im}}$ = [Pb] in meat (mg/kg)

C_{w_i} = [Pb] in water (mg/L)

F_{m_i} = water-to-meat transfer coefficient $(kg/day)^{-1}$

f_w = fraction of total water intake that is water containing Pb

Q_w = daily water intake of beef cattle (L/day)

β_{w_i} = decay constant for Pb in water $(day)^{-1}$

th_m = holdup time from slaughter to consumption (days)

Assumptions:

A steer produces 200 kg (441 lb) of beef (Baes et al., 1984)

$F_{m_i} = 3 \times 10^{-4}\ (kg/day)^{-1}$ (Baes et al., 1984)

f_w = 1 (i.e., all water consumed is assumed to contain lead)

Q_w = 49 L/day

β_{w_i} = 0 (Pb is environmentally persistent (EPA, 1986))

- th_m = 20 days (Whelan et al., 1987)

Calculations:

$$C_{w_{im}} = C_{w_i} (3 \times 10^{-4}) (1) (49) (e^{-0}) = 0.0148\ C_{w_i}$$

$= (0.0148) \times (1.50)$

$= 0.022$ mg (lead)/kg (beef)

TABLE I.1.21 Concluded

Equation 3: Human Lead Intake via Beef Ingestion

$I_r = C_{w_{im}} (IR)_m (GPF)/(W_A)$ (Modified from Envirosphere, 1987)

Where:

I_r = daily intake of Pb by consumer (mg/kg-day)

IR_m = meat ingestion rate (kg/day)

GPF = gut partition factor

W_A = average adult body weight (kg)

Assumptions:

IR_m = 0.086 kg/day (adult males 19-50) (ICRP, 1975)

GPF = 0.15 (ATSDR, 1988)

W_A = 70 kg (EPA, 1986)

Calculations:

$I_r = C_{w_{im}} (0.086)(0.15)/70 = 1.84 \times 10^{-4} C_{w_{im}}$

$= (1.84 \times 10^{-4})(.022)$

$= 4.05 \times 10^{-6}$ mg/kg-day

Equation 4: Human Lead Hazard Index

HI $= I_r/AIC$ (EPA, 1986)

AIC $= 4.30 \times 10^{-4}$ mg/kg-day (EPA, 1986)

HI $= 4.05 \times 10^{-6}/4.30 \times 10^{-4} = 0.009$

I.1.4.5 **Health Effects from Exposure to Stable Lead in Beef**

This subsection examines the potential human health impacts associated with the release of stable lead to the biosphere. The release scenario examined involves a breach of the repository by a single borehole that penetrates both a waste panel and the pressurized brine of the Castile Formation below the host formation (Case II). The scenario and the assumptions used to model the subsequent release of stable lead to the Culebra are described in Subsection 5.4.2.6. The concentrations of stable lead calculated to reach the stock well at 10,000 years, when that concentration reaches its maximum at the end of the calculations, are 4×10^{-6} mg (lead)/L (brine) in Case IIA(rev) and 1.5 m/L in Case IIC(rev). This assessment assumes that beef cattle consume water from a hypothetical stock well that contains the maximum concentration of lead and that this concentration is maintained in the stock pond throughout the lifetime of the cattle. The equations used to calculate lead uptake by humans are provided in the following pages.

The methodology for this assessment involves calculating the amount of lead uptake per unit body weight of the cattle, the concentration of lead retained in beef, and the concentration of lead ingested by humans consuming this beef.

To calculate lead intake by cattle in Case IIC(rev), it is assumed that 49 liters per day of water containing 1.50 mg/L lead is consumed. An average steer weighs 400 kg (882 lb) (Merck & Co., 1979). A gut partitioning factor of 0.15 is used to account for the fact that not all of the lead ingested by cattle is retained in the beef (i.e., a portion of the lead will be excreted) (ATSDR,1988). Thus, the cattle may take up and retain lead at the rate of 0.028 mg/kg-day (Table I.1.21, Equation 1). It has been estimated that a mature steer will tolerate 6 mg/kg-day lead for 2 to 3 years (Botts, 1977). Assuming the concentration of lead in the stock water remains constant throughout the lifetime of the steer, it is estimated that 0.022 mg of lead per kg of beef will be available for human consumption (Table I.1.21, Equation 2).

For the purposes of these calculations, it is estimated that an adult male (age 19 to 50) consumes 0.086 kg of beef daily (NCRP, 1984). Adult male body weight averages 70 kg (154 lb). The daily human retention of lead, assuming 0.022 mg/kg of lead in the beef consumed, is 4.05×10^{-6} mg/kg-day (Table I.1.21, Equation 3).

The estimate of the daily intake of lead by humans calculated in this manner can be compared to the acceptable daily level for chronic intake (AIC) according to procedures described in the Superfund Public Health Evaluation Manual (EPA, 1986) (see SEIS Appendix G). As shown in Table I.1.21, Equation 4, the acceptable daily level for chronic intake (AIC) is 4.30×10^{-4} mg/kg-day (EPA, 1986). The calculated AIC-based hazard index for lead in Case IIC(rev) is 0.009. This value is considerably less than unity, indicating that the estimated intake of lead is well below the acceptable reference level. In other words, the ingestion of this concentration of lead every day throughout the life of the consumer will not result in adverse health effects.

For Case IIA(rev), these figures are only 3×10^{-6} as large, and the hazard index is that much further below the acceptance intake level.

I.1.5 **ASSESSING COMPLIANCE WITH THE EPA STANDARDS**

On September 19, 1985, the Environmental Protection Agency (EPA) promulgated Environmental Standards for the Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes (40 CFR Part 191). In 1987, the court remanded these standards to the EPA because of differences between their groundwater protection provisions and those in the EPA drinking water standards. Strictly speaking therefore, there are at present no standards against which to judge the potential performance of the WIPP. However, the DOE has agreed with the State of New Mexico to use the remanded standards in its planning and analyses until new ones are promulgated.

A June 2, 1989, Working Draft of a possible new 40 CFR Part 191 has recently become available. The changes from the remanded standards that are relevant to the WIPP are:

- A possible alternative to the present definition of "disposal" defines disposal as placement in a disposal system, but explicitly excludes "placements for experimental purposes that include pre-established plans for the removal of the fuel or waste."

- The definitions of groundwater are changed. The proposed new definition would make groundwater in the Culebra dolomite Class IIIB groundwater. Class III groundwater is groundwater that is "saline or otherwise contaminated beyond levels that would allow use for drinking or other beneficial purposes." Class IIIB groundwater is Class III groundwater "characterized by a low degree of interconnection to adjacent groundwaters of higher class or surface waters."

- A new containment requirement extends the time period of concern for the performance of an undisturbed disposal system to cover the period of 10,000 to 100,000 years, with the projected releases over this extended time to be "not much greater" than allowed by a table reproduced here as Table I.1.22. The expected performance of the undisturbed repository as reported in Subsection 5.4.2.5 as Case IA meets this potential requirement.

- A new assurance requirement has been added, that "disposal systems shall be selected to and designed to keep releases to the accessible environment as small as reasonably achievable, taking into account technical, social, and economic considerations." This potential requirement is met by DOE's explicit commitment to comply with all applicable standards, including 40 CFR 191 as finally promulgated.

- The groundwater protection requirements (Part 191.16) are completely rewritten with several options put forward. A portion of Option 2.C might apply to the WIPP. This option, if decided upon, would call for "a reasonable expectation that, for 1,000 years after disposal, undisturbed performance shall not cause . . . any increase in the levels of radioactivity for Class IIIB groundwaters such than an individual can receive more than 25 millirems

TABLE I.1.22 Release limits for containment requirements (cumulative releases to the accessible environment for 10,000 years after disposal)

Radionuclide	Release limit per 1,000 MTHM or other unit of waste[a] (curies)
Americium-241 or -243	100
Carbon-4	100
Cesium-135 or -137	1,000
Iodine-129	100
Neptunium-237	100
Plutonium-238, -239, -240, or -242	100
Radium-226	100
Strontium-90	1,000
Technetium-99	10,000
Thorium-230 or -232	10
Tin-126	1,000
Uranium-233, -234, -235, -236, or -238	100
Any other alpha-emitting radionuclide with a half-life greater than 20 years	100
Any other radionuclide with a half-life greater than 20 years that does not not emit alpha particles	1,000

[a] For TRU waste, this unit is a million curies of alpha-emitting transuranic nuclides with half-lives greater than 20 years. The proposed new standards increase the size of the unit to ten million curies, but also increase the release limits per unit by a factor of ten; the net result is no overall change in release limits.

Note: For projected releases of several radionuclides, there is the additional requirement that

$$R = \frac{Q_a}{RL_a} + \frac{Q_b}{RL_b} + \cdots + \frac{Q_n}{RL_n} \leq 1 \text{ (or 10)}$$

where R is the normalized release, Q_i is the projected release of radionuclide i and RL_i is its release limit for that radionuclide.

annual committed effective dose equivalent from all routes of exposure from the disposal system." Option 3.B would extend this time to 10,000 years. The expected performance of the undisturbed repository as reported in Subsection 5.4.2.5 as Case IA meets this potential requirement.

Part B of both the remanded and the proposed 40 CFR Part 191 sets standards for the disposal of TRU wastes in a geological repository. Part B protects the public from significant radiation doses by requiring that no more than a predetermined amount of each radionuclide be released to the biosphere. Specifically, the disposal systems are to be designed to provide a reasonable expectation that the cumulative releases of radionuclides to the accessible environment for 10,000 years from all significant process and events shall have:

- less than one chance in 10 of exceeding the quantities calculated according to Table I.1.22, and

- less than one chance in 1,000 of exceeding 10 times those quantities.

The standard is thus one dealing with probabilities rather than certainties. It says that performance assessments do not have to provide complete assurance that these requirements will be met. Because there will be substantial uncertainties in projecting disposal system performance, actual proof of the future performance cannot be attained. Instead, the standards require a reasonable expectation that compliance will be achieved.

The EPA assumes that, whenever practicable, the DOE will summarize the results of the performance assessment into a complementary cumulative distribution function (CCDF) indicating the probability of exceeding various levels of cumulative release, written P (Release > R). The effects of the uncertainties will be incorporated into a single CCDF for each disposal system. If this CCDF meets the requirements above, then that disposal system is deemed to comply with Part B of the EPA standards.

I.1.5.1 **Performance Assessment**

A performance assessment consists of four parts:

- Scenario development and screening,

- Consequence assessment,

- Sensitivity and uncertainty analysis, and

- Regulatory compliance assessment.

Scenario development and screening examine possible future events or processes that might affect a repository, assign probabilities to them, and determine which possibilities merit detailed consideration. Consequence assessments estimate the releases that might arise from the scenarios of interest. Sensitivity and uncertainty analyses identify important processes and parameters and illuminate the sources and extent of uncertainties in the consequence assessment, thus enabling the regulator to evaluate the confidence that can be placed in the results. Finally, a regulatory compliance assessment combines the results of the scenario analyses, consequence assessments, and sensitivity and uncertainty analyses and determines whether the repository is in compliance with the requirements of the EPA standards.

In a Monte Carlo simulation using deterministic models for predicting consequences, the following approach can be used to generate a CCDF. (If a stochastic or some other model is used, another technique would be used to generate a CCDF.) This process is described in greater detail in Hunter et al., 1986.

Assume that K scenarios have been identified as important. For WIPP, K might be as large as 10. These scenarios are analyzed by choosing appropriate ranges and distributions for the model's input parameters and then statistically sampling from these ranges to obtain sets of input values for the scenarios. This sampling must be done by some means such as Latin Hypercube sampling, so that all samples have the same probability of occurring. (The same set of input parameters is used for all scenarios in order to ensure that any variation observed between scenarios is due to scenario differences and not to differences in sampling.) For WIPP, the number of sets of input parameters, N, might be 100.

Thus there will be NK sets of consequences, R_{nk}, calculated in the performance assessment. For WIPP this may amount to as many as 1,000 calculations.

For each scenario, a probability will also be estimated. The sum of these probabilities P_k cannot, statistically speaking, be greater than one. Each probability is therefore normalized by dividing them by the sum of probabilities ΣP_k and by N. Similarly, the consequences are normalized by dividing them by the release limits given in Table I.1.22.

There result NK pairs of normalized consequences and associated probabilities. These pairs are ordered by the magnitude of their consequences, with the largest consequence first. Then the CCDF [P(Release > R_{nk})] is the sum of all normalized probabilities $P_k/(N\Sigma P_k)$ down to that point on the list.

A CCDF generated in this manner is actually a step function consisting of NK steps (Figure I.1.2). The EPA containment requirements are indicated as the forbidden area in the upper right hand area in this figure. If the CCDF remains outside this forbidden area, the standard is met. This particular hypothetical example indicates a region of possible violation.

I.1.5.2 **Application to WIPP**

Case II will probably be one of the scenarios entering into the CCDF for the WIPP. Its probability of occurrence is high. Using EPA's figure of 30 holes per square kilometer of repository area, 51 holes may be drilled into the WIPP in 10,000 years (i.e., 1 every 200 years on the average). Inasmuch as the waste disposal panels subtend about 7 percent of the repository area, the probability of a hole intersecting a waste panel is

$$[1 - (1 - .07)^{51}] = 1 - .025 \approx .98$$

Then, because about half the WIPP area appears to be underlain by a brine reservoir (Earth Technology Corporation, 1987), this probability should be multiplied by 0.5. Finally, superimposed on this should be the probability that the drill hole will actually go as deep as the Castile brine reservoir—it could be being drilled for potash evaluation. Taking this probability arbitrarily as another 0.5, the net probability of Case II is

$$0.98 \times 0.5 \times 0.5 = .25.$$

(Not knowing what the other scenarios might be, this probability cannot be normalized by dividing it by ΣP_k.)

If Case IIC(rev) should be one of the 100 or so sets of input parameters with which Case II is analyzed, its probability would be the overall Case II probability divided by 100, or 2.5×10^{-3}. However, this is not likely, as Case IIC(rev) analyzes the consequences of an extreme case in which all the input parameters (except the initial pressure in the brine reservoir) are taken at the extremes of their ranges.

The calculation of integrated release for Case IIC(rev) in Subsection 5.4.2.8 of this SEIS would therefore appear as one of the last steps in the lower right hand corner of Figure I.1.2. Thus this calculation alone, although with an integrated release of 3.2, does not *per se* indicate noncompliance with the regulations.

Figure I.1.2

Hypothetical example of a complementary cumulative distribution function (CCDF)

I.2 **DATA**

I.2.1 <u>**FINAL WASTE POROSITY**</u>

If it is assumed, for purposes of calculation, that structural changes do not take place after mechanical compaction, the void volume remaining within a room after waste compaction determines the maximum amount of brine that may eventually enter the room. This value is difficult to estimate, however, because the mechanical and physical properties of the waste are highly variable and poorly characterized.

The compressive stress exerted by the surrounding salt is not sufficient to completely eliminate all voids in the waste. As the waste is compacted, its resistance to additional densification increases, and it becomes rigid enough to prevent further void reduction. A near-term limiting void volume within the repository, associated with purely mechanical densification and expected to be attained in 60 to 200 years, is used for this analysis and is assumed to represent a "steady-state."

Even after this time, the state of the repository will continue to change, as biological decomposition and chemical corrosion alter the chemical and structural nature of the waste. This longer-term evolution of the physical state of the repository is expected to be complex, to occur over a long period of time, and to include interactions between compaction processes and possible repository expansion as a result of gas generation. Its quantitative characterization may never be possible. At least for metal wastes, densification may continue beyond that produced by early room closure, and consequently the near-term limiting void volume is considered the greatest void volume that will exist within the waste. The final room porosity enters the calculations in this report in three ways. First, the estimated porosity is used to estimate the final permeability of the repository. This value is used in the Case I calculations, but does not enter directly into the Case II calculations. Permeability is used there to determine whether Castile brine-reservoir fluids effectively mix with the waste in the repository. Second, the final porosity estimate is used to estimate the volumes available within the repository for gas storage or saturation with brine. Finally, the porosity estimate is used to determine the volume of brine available to dissolve radionuclides. Dissolution is limited either by the mass required to reach the solubility limit of individual radionuclides or by the total mass of the radionuclides present, whichever is less.

The final void volume used here is based on the distribution of waste types in storage (Table I.2.1) (DOE, 1988a). A total of 6,804 drums are assumed to be stored in seven-pack configurations within a disposal room, each with an internal volume of 0.21 m^3. In assigning final porosities to each component, combustible waste (low-strength plastics, paper, and rags) is assumed to have such low strength that the near-term interconnected void porosity will be 0.1 or less after compaction to lithostatic pressure (approximately 14 MPa). Because combustible waste will collapse to a dense, interlocking structure, its hydraulic response is considered to be similar to that of silt, with a hydraulic conductivity of 10^{-8} m/s. (The porosity is $n = V_v/V$, where V is the boguspage.

TABLE I.2.1   Final void volumes in waste

	Waste Form				
	Combustible	Sludge	Metal/Glass	Other	Total
		Emplaced			
Percent by weight of total waste in storage	30	17	33	20	100
Initial volume in disposal room (m^3)	429	243	472	286	1430
Percent of solids per drum	24.8	66.5	21.9	---	
		Final			
Solids volume (m^3)	106	162	103	93	464
Void volume (m^3)	12	18	68	25	123
Waste volume (m^3)					587

Cf. Lappin et al., 1989, Table 4-5.

Sources. DOE, 1988a; Clements and Kudera, 1985.

compacted volume, and V_v is the void volume. $V = V_v + V_s$, where V_s is the solid volume that the waste would occupy if no voids were present. Later, the void ratio, e, will be used, which is defined as $e = V_v/V_s$, or $e = n/1 - n$.)

The mechanical properties of sludge are not well defined, but this category of waste represents only 17 percent of the total waste inventory. Sludge is much more difficult to compact than combustible waste, and therefore its total void content after compaction is likely to be greater. The same interconnected porosity, 0.1, is assumed for it in the compacted state, however, because many sludges may have a high cement content and are expected to form hydration products that decrease void interconnectivity. In the absence of any data about the hydraulic conductivity of sludge, a value two orders of magnitude greater than for grout has been assumed. The hydraulic conductivity of grout is 1×10^{-11} m/s (Coons et al., 1987), implying a final-state conductivity of 1×10^{-9} m/s for sludges.

The strengths of metallic and glass wastes make them much less compactible than combustible and sludge wastes. Most of the waste is metallic in content. The final porosity assumed for metal and glass waste is 0.4, based on powder-metallurgy literature (Hausner and Kumar, 1982) and on data on supercompaction, which suggests that compaction of metal waste to much greater than 0.6 of theoretical solid density is not likely. A lower final porosity for the metal waste can be expected, however, if the crushed-salt and bentonite backfill intrudes into the open spaces between the pieces of metal, a process that could reduce porosity by as much as 50 percent. Thus, a lower bound to metal-waste porosity is taken to be 0.20.

The properties of the waste category referred to as "other" remain undefined. In the absence of further information about the composition of this waste, its compacted porosity is assumed to be the average porosity of the combined combustible, sludge, and metal and glass waste categories, weighted according to the portion of the inventory that each represents.

Final void volumes for combustible, sludge, and metal and glass waste categories are given in Table I.2.1. The volume of solid waste per drum is computed using the average initial void fraction of each waste category (Clements and Kudera, 1985). Adding in the void volume of the unspecified "other" category of waste (20 percent of the inventory), the total void volume per room is 123 m^3, corresponding to a solids volume of 464 m^3. This 123 m^3 volume, divided by the total volume, 587 m^3, yields a porosity of 123/587 = 0.210. If the void volume in the metal waste is assumed to be reduced 50 percent by salt intrusion, the net void volume per room is approximately 123-34 = 89 m^3. This corresponds to a porosity of 89/587 = 0.152. The "expected" final void volume for the consolidated waste is the average of the estimated void volumes, or 106 m^3 per room, corresponding to an interconnected void porosity of 106/587 = 0.182. To be conservative, the release scenarios in Subsection 5.4 use a saturated void volume of 123 m^3.

The estimates above apply only to the waste and do not include any final-state porosity of backfill in the room, because the compacted salt-bentonite backfill is expected to be relatively impermeable. The void volume calculations take no credit for the fact that the metal and glass waste may contain minor amounts of easily compacted materials such as combustibles or sorbents. The only study that has quantitatively inventoried the contents of TRU waste in detail (Clements and Kudera, 1985) showed that metals represent only about 80 percent by weight of the INEL metal waste. The remainder of the metal category contents is combustible material (12 percent) and cement (5 percent), which would reduce its compacted porosity. A major uncertainty in this analysis is introduced by the absence of any information about the compactibility of the various waste types, although tests to determine compactibility are in progress.

An estimate was made of how rapidly the limiting void volume within a disposal room is approached (Figure 5.3). The calculated rate of closure of an empty disposal room (Munson et al., 1989) was used to determine the void volume at a given time. The void volume was obtained by subtracting the volumes of the solids in the waste and backfill and the volume of brine flowing into the room as a function of time (Nowak et al., 1988) from its current volume. Figure 5.3 is not completely consistent with values listed in Table I.2.1 because, in the absence of experimental results, equal rates of consolidation of waste and backfill were assumed.

An assumption in using closure data for an empty room for this estimate is that any backstress by the room contents is insufficient to retard void reduction. This appears to be warranted for room porosity greater than approximately 0.3: finite-element calculations show that backstress is significant only during the latest stages of closure. The no-backstress assumption is also consistent with the current model for compaction of the waste, which assumes that the final void volume depends only on the stress applied to the waste, and not on the stress history; that is, the only effect of backstress is to prolong the time required to achieve the final compacted state. This assumption, however, is another source of uncertainty. Estimates using these assumptions show that the limiting void volume could be achieved in 40 to 60 years; 60 to 200 years is assumed in Subsection 5.4.2.4, Brine Inflow. The amount of brine flowing into the room during 60 years is estimated to be between 6 and 37 m^3, a factor of 4 less than would be required to saturate the 123 m^3 of void volume at final-state. In fact, all this brine can be sorbed by the bentonite in the backfill (Subsection 5.4.2.4). In addition, the pressure of decomposition gases within the room, even assuming none leak out, would not reach lithostatic pressure in 60 years (Lappin et al., 1989, Subsection 4.10.2).

I.2.2 **RADIONUCLIDE SORPTION**

The K_d values used in the SEIS analyses are summarized in Tables I.2.2 through I.2.5. Table I.2.2 contains K_d values that are used to calculate radionuclide retardation in the matrix of the Culebra dolomite. Tables I.2.3 and I.2.4 contain sorption ratios for the clays that line the fractures in the aquifer. Table I.2.5 contains K_d values for use in radionuclide transport in the tunnels, seals, and Marker Bed 139 at the repository level. If the volume of the clays within the fractures is known, then the K_ds in Table I.2.3 can be used to calculate retardation within the fractures using the following expression

$$R_f = 1 + \rho_c K_{dc}(\delta_c/\delta), \qquad \text{(I-38)}$$

where K_{dc} is the distribution coefficient for the clay given in Table I.2.3; ρ_c is the density of the clay (2.5 g/cm^3); δ_c is the thickness of the clay coating the fracture; and δ is the fracture width (Neretnieks and Rasmuson, 1984).

Surface area-based distribution coefficients K_a (ml/m^2) for the clay are listed in Table I.2.4. These were calculated from the K_ds assuming a surface area of 50 m^2/g. This is similar to the surface area of 32 m^2/g measured by Nowak (1980) on a reference montmorillonite used in europium sorption studies and within the range of 15 to 88 m^2/g measured by Soudek (1984) on montmorillonite used in ion exchange studies.

A retardation factor for use in a transport equation for fracture-dominated flow where sorption occurs on the surface of the fracture fill clay can be calculated as

$$R = 1 + aK_a/\varphi \qquad \text{(I-39)}$$

Table I.2.2 K_d values for radionuclide transport in the matrix of the Culebra dolomite (ml/g)

Case	Pu	Am	Cm	U	Np	Pb, Ra	Th
Case I	100	200	(200)	1	(1)	(1)	(100)
Case IIA, IIA (rev)	50	200	(200)	1	(1)	(0.1)	(50)
Cases IIB, IIC, IIC (Rev), & IID 25	100	(100)	1	(1)	(0.05)	(25)	

Values in parentheses are poorly known; estimated by assumption of behavior similar to a homolog element.

Source. Lappin et al., 1989, Table E-10.

TABLE I.2.3 K_d values for radionuclide transport in the fracture clays of the Culebra dolomite (ml/g)

Case	Pu	Am	Cm	U	Np	Pb, Ra	Th
Case I	300	500	(500)	10	(10)	(100)	300
Case IIA, IIA (rev)	200	(300)	(300)	10	(10)	(10)	(200)
Cases IIB, IIC, IIC (Rev), & IID (100)	(100)	(100)	(1)	(1)	(5)	(100)	

Values in parentheses are poorly known; estimated by assumption of behavior similar to a homolog element.

Source. Lappin et al., 1989, Table E-11.

TABLE I.2.4 K_a values for radionuclide transport in the fracture clays of the Culebra dolomite (ml/m^2)

Case	Pu	Am	Cm	U	Np	Pb, Ra	Th
Case I	6	10	(10)	0.2	(0.2)	2	6
Case IIA, IIA (rev)	6	(6)	(6)	0.2	(0.2)	(0.2)	(6)
Cases IIB, IIC, IIC (rev) & IID	(2)	(2)	(0.02)	(0.02)	(0.1)	(2)	

Values in parentheses are poorly known; estimated by assumption of behavior similar to a homolog element.

Source. Lappin et al., 1989, Table E-12.

TABLE I.2.5 K_d and K_a values for radionuclide transport in tunnels, seals, and MB 139

	Pu	Am	Cm	U	Np	Pb, Ra	Th
Clay in crushed Salado salt, K_d (ml/g)	100	100	100	1	(10)	(1)	(100)
Anhydrite in MB 139, K_d (ml/g)	100	25	25	(1)	(1)	(1)	(100)
Anhydrite in MB 139, K_a (ml/m^2)	3700	925	(925)	(37)	(37)	(37)	(3700)

Values in parentheses are poorly known; estimated by assumption of behavior similar to a homolog element.

Source. Lappin et al., 1989, Table D-5.

where K_a is the sorption ratio in Table I.2.4, a is the specific surface of the fracture (fracture surface area per unit volume of fracture), and φ is the porosity.

Similarly a retardation factor for use in transport through the porosity-controlled flow in the tunnels and seals can be calculated as

$$R = 1 + \rho K_d (1 - \varphi)/\varphi \qquad \text{(I-40)}$$

where K_d is the distribution coefficient given in Table I.2.5, ρ is the grain density, and φ is the porosity.

The following procedure was used to obtain the recommended K_d values listed above. First, initial ranges of values were obtained from studies carried out under chemical conditions that were similar in some way to those expected under a variety of mixing ratios in the various media. Second, K_d values obtained under conditions closest to those expected in the WIPP were extrapolated to reference conditions consistent with the descriptions of Cases I and II. Data from parametric studies or theoretical calculations for simple, well-constrained systems were used to estimate the magnitude of the change in the K_d that might be related to differences between the actual experimental conditions and the range of conditions possible for the cases. Finally, uncertainties in the future physicochemical conditions in the repository and along the flow path in the Culebra dolomite were considered. Possible deviations of K_d values from those estimated in the previous step were evaluated, and a set of conservative, realistic K_d values was selected.

In the waste panels, solution chemistry will be dominated by the composition of Salado brines, leachates from the waste, concrete and steel drums, and the products of microbial degradation. In the SEIS analyses, it was assumed that the important sorbing substrates will be iron oxide corrosion products and bentonite backfill. Radionuclide transport within the waste panel was not considered in the report. Available sorption data were used to estimate the partitioning of radionuclides between solution and suspended solids.

The amount of radionuclide sorbed to the particulates will be related to the solution composition and to the total number of sorption sites of the substrate. Consideration was first given to sorption capacity independent of the effects of solution composition. For an oxide or oxyhydroxide, the total sorption capacity is related to the number of surface hydroxyl groups. For a clay such as bentonite, the sorption capacity will be determined by both the number of exchangeable (fixed-charge) sites and the number of hydroxyl groups (Kent et al., 1988).

The total sorption capacities of iron oxyhydroxide and bentonite were estimated from experimental data obtained under conditions very different from those assumed for the waste panel. Under conditions of low total dissolved solids, low concentration of cations such a Mg^{+2} and Ca^{+2} and low organic concentration, sorption capacities of bentonite could range from 10 to 100 milliequivalents (meq) per 100 grams (Drever, 1982; Tsunashima et al., 1981). Under similar conditions, the sorption capacity of iron oxyhydroxides could range from 60 to 300 meq per 100 grams (estimated from data in Hayes et al. (1988, Table 1). The actual sorption capacities will depend on the crystallinity and stoichiometry of the clays and iron oxides present in the repository.

Moderate concentrations of carbonate, organic-sequestering agents, Mg^{+2}, and Ca^{+2}, however, will keep some of the sorption sites from being occupied by actinide ions. The effects of the solution composition on sorption are discussed in more detail in Lappin et al. (1989, Section 3.3.4.2).

I.2.2.1 **Rationale for Extrapolation of K_d Values**

Table I.2.6 summarizes a variety of data on experimental measurements of K_ds in brine. The values for K_d used in this SEIS are lower than those listed in Table I.2.6. Many of the K_ds for the actinides reported in the literature are in the range 10,000 to 100,000 ml/g. The K_ds were calculated solely from the loss of radioactivity from solution, and therefore small errors in the measurement of a trace amount of radionuclide remaining in solution could lead to large errors in the calculated K_d. Review of experimental procedures used to obtain the values suggests that the results could be compromised by unrecognized precipitation; this error would lead to high K_ds that would overestimate the extent of sorption. This kind of error could be especially important for data from WIPP Brine A and B. Saturation index calculations by Melfi (1985) indicated that Brine A is supersaturated with respect to calcite ($CaCO_3$) and that Brine B is supersaturated with respect to gypsum ($CaSO_4 \bullet 2H_2O$). The extent to which the actinides or fission products can be incorporated into the crystal structure of either of these minerals has not been determined.

The uncertainties in the course of the future chemical evolution of the repository and aquifer require consideration of large ranges of pH, Eh, organic content, and carbonate content of the groundwaters. These possible variations in solution chemistry could result in order of magnitude changes of the K_ds from the values obtained in the experimental studies listed in Table I.2.6. Evaluation of the magnitude of these changes requires several assumptions about the nature of sorption reactions occurring on the substrates.

For the purpose of the SEIS, it is assumed that only the clay, anhydrite, and salt components of the Salado will come into contact with the radionuclides during transport. It is assumed that none of the elements sorb onto halite (K_d = 0). The sorption of trace metals onto salt-like minerals such as anhydrite is poorly understood; the paucity of relevant data precludes extrapolation of sorption behavior to physicochemical conditions that differ from those specifically examined in the experimental studies listed in Table I.2.6. Some qualitative extrapolations are made; they are based solely on the predicted aqueous speciation of the radionuclides.

TABLE I.2.6 Sources of K_d data used to estimate values for repository (Case I) and Culebra (Case II) transport (saline water ± organic ligands). (See Lappin et al. [1989], Table 3-4 for compositions of WIPP brines A and B)

				Reported K_d range					
Reference	Water	Rock	Organics	Pu	Am	Cm	U	Np	Eu
Nowak (1980)	Brine A	Bentonite	None	$2.3x10^3$ to $3.4x10^3$					350
	Brine B	Bentonite	None	$2x10^4$ to $4x10^4$	$4.1x10^3$ to $1.4x10^4$				$1.4x10^3$ to $1.7x10^3$
Dosch & Lynch (1978)	Brine A	Clay	None						$>1x10^3$
	Brine B	Clay	None	$4x10^4$ to $7.2x10^4$	310 to 1100	$2.7x10^3$ to $1.9x10^4$			$>1x10^4$
	Dissolution	Halite	None	17 to 59	11 to 306	56 to 354			
	Brine	(as clay)	None	($1x10^4$ to $2x10^4$)	($3.8x10^3$ to $1.8x10^5$)	$2x10^4$ $2.1x10^5$			
	Brine A	Rustler Dolomite							$>5x10^3$
	Brine B	Rustler Dolomite	None	$2.1x10^3$ to $5.4x10^3$	$3.2x10^2$ to $2.6x10^3$	$1.3x10^3$ $1.2x10^4$			$>5x10^3$
	Brine B	Anhydrite	None	$6.7x10^3$	$2.9x10^2$	$4.2x10^3$			$>1x10^3$
Dosch (1981)	Brine A	Culebra Dolomite	None				0 to 2		
	Brine B	Culebra Dolomite					1.5 to 608		

Cf. Lappin et al., 1989, Table 3-14.

TABLE I.2.6 Continued

				Reported K_d range					
Reference	Water	Rock	Organics	Pu	Am	Cm	U	Np	Eu
Serne et al. (1977)	Brine A	Rustler Dolomite					29 to 52		
	Brine B	Culebra Dolomite	None	50 to 200	340 to 1160	0.0 to 1.2×10^4	0 to 7.1	10 to 28	22 to 40
Paine (1978); Dosch & Lynch (1978)	Brine B	Culebra Dolomite	EDTA, etc.	25 to 6,000	100 to 2.8×10^4				
	Brine B	Rustler Dolomite	Waste	560 to 1.8×10^4	5.7×10^4 to 1.7×10^5				70 to 660
	Brine B	Anhydrite	Waste						400
	Brine B	Clay	Waste						2.8×10^4
Tien et al. (1983)	Salt brine	Claystone	None	3×10^2 to 1×10^4 Ra = 3	90 to 1,000	3×10^2 to 1×10^4	50	5 to 2,000	
	(TDS: > 3×10^{-4})	Carbonate	None	50 to 6×10^3	3×10^2 to 2×10^4	3×10^2 to 1×10^4	0 to 3	15 to 30	
		Salts	None	20 to 1×10^4	3×10^2 to 2×10^5	3.5×10^2 to 2×10^5			

TABLE I.2.6 Concluded

References	Water	Rock	Pu	Am	Cm	U	Np	Eu
			Reported K_d range			Additional		
References	Water	Rock	Pu	Am	Cm	U	Np	Eu
Dosch and Lynch (1978)	Solution C	Rustler Dolomite	$2.4x10^3$ to $7.3x10^3$	$2.4x10^3$ to $2.2x10^4$	$4.2x10^4$ to $1.1x10^5$			$>1x10^4$
	Solution C	Clay	$4x10^4$ to $1.8x10^5$	$2.3x10^3$ to $3.5x10^3$	$1.6x10^5$ to $4.2x10^5$			$>1x10^4$
	Solution C	Anhydrite	$7.7x10^4$	$2.2x10^3$	$1.8x10^5$			$>1x10^3$
Dosch (1981)	Solution C	Rustler Dolomite					13 to 175	
	H2 (B,C)	Culebra Dolomite					0.6 to 7.4	
Lynch and Dosch (1980)	H2 (B,C)	Rustler Dolomite	83 to 1055	119 to 383				
Serne et al. (1977)	Solution C	Culebra Dolomite	42 to 2206	2,500 to 3,000	$1.6x10^3$ to $4.3x10^4$	0 to 16	5 to 35	
Tien et al. (1983)	Saline H_2O	Claystone	300 Ra = 30	200 to $1x10^4$	1,000	20	1 to 75	
	(TDS: $5x10^3$ to $3x10^4$	Carbonate	500 Th = $2.7x10^3$ to $1x10^4$	3,000	3,000	3	21	
	Fresh H_2O	Claystone	30 to $1x10^4$ Ra = 300	700 to $1x10^4$	700 to $1x10^5$	1.270	2 to 400	
	(TDS: < $5x10^3$	Carbonate	100-7,000 Th = $1x10^3$ to $1x10^5$	500 to $1x10^5$	$6x10^3$ to $4x10^4$	0 to 15	15 to 30	

I.2.2.2 **Rationale for Choices of Recommended K_d Values**

The data for the simple systems discussed above suggest that the amount of sorption of actinides onto either clays or sulfates present in the repository could be several orders of magnitude less than that suggested by the K_d data listed in Table I.2.6. Although it is possible that under severe conditions, the K_ds will be close to zero, there is evidence that some sorption will occur; therefore in the SEIS the K_ds are not assumed to be zero. The rationales behind the K_d values chosen for each element are given below.

Plutonium. K_d values are decreased by 2 to 3 orders of magnitude from the values in Table I.2.6 to account for the potential effect of carbonate complexation and competition for sorption sites by competing cations.

Americium. K_d values are decreased by factors of 3 to 1,000 from values listed in the table to account for the potential effects of organic complexation. For example, Swanson (1986) found that EDTA significantly decreased Am sorption onto kaolinite and montmorillonite. The magnitude of this effect was a function of the pH and concentrations of EDTA, Ca, Mg, and Fe in solution.

Curium. K_d values were decreased by factors of 3 to 100 from the values listed in Table I.2.6 based on the assumption of similar behavior to Am and Eu.

Uranium and Neptunium. Generally, low K_ds have been measured in waters relevant to the WIPP. The K_d of uranium is very dependent on the pH and the extent of complexation by carbonate and organic ligands. A low value ($K_d = 1$) has been assumed in the SEIS to account for the possible effects of complexation and competition. Theoretical calculations (Siegel et al., 1989) and arguments based on similarities in speciation, ionic radii, and valence (Chapman and Smellie, 1986) suggest that the behavior of neptunium will be similar to that of uranium.

Thorium. There are few data for thorium under conditions relevant to the WIPP. Thorium K_d values were estimated from data for plutonium, a reasonable homolog element (Krauskopf, 1986). Data describing sorption of Th onto kaolinite (Riese, 1982) suggest that a high concentration of Ca and Mg will prevent significant amounts of sorption onto clays in the repository. Stability constants for organo-thorium complexes suggest that organic complexation could be important in the repository and may inhibit sorption (Langmuir and Herman, 1980).

Radium and Lead. There are very few sorption data for radium under conditions relevant to the WIPP. K_d values in Table I.2.6 were estimated from assumption of homologous Ra-Pb behavior in Tien et al. (1983). Data from Riese (1982) suggest that Ra will sorb onto clays but that high concentrations of Ca and Mg will inhibit sorption. Langmuir and Riese (1985) present theoretical empirical arguments that suggest that Ra will coprecipitate in calcite and gypsum/anhydrite in solutions close to saturation with respect to these minerals.

I.2.3 **BRINE RESERVOIR PARAMETERS**

In order to model the hydraulic connection between the Culebra member and a brine reservoir in the underlying Castile Formation, it is necessary to realistically define the hydrologic parameters that govern the transient hydraulic response of the brine reservoir. These parameters include

- P_i = initial reservoir pressure
- b = reservoir thickness
- T = reservoir transmissivity
- ρ = reservoir fluid density
- r_b = effective distance to the reservoir boundary
- φ = reservoir porosity
- α= reservoir compressibility

Interpretation of flow and buildup data from the WIPP-12 brine reservoir indicates that the reservoir is heterogeneous (Popielak et al., 1983). As a result, the hypothetical WIPP brine reservoir is being modeled as consisting of three separate media. Base-case values for each of the parameters listed above are derived from the available data on WIPP-12 (D'Appolonia Consulting Engineers, 1982 and 1983; Popielak et al., 1983) and on the interpretation presented in Section 3.4.3 of Lappin et al. (1989). Uncertainty ranges about these base-case values are derived from WIPP-12 test interpretations and from the limited data base from 12 other wells that have penetrated brine reservoirs in the Castile Formation in the vicinity of the WIPP site. Because data on 11 of these brine occurrences are limited, the parameter range is derived in most cases from the WIPP-12 and ERDA-6 data. For well locations and distributions of Castile brine occurrences, see Figure I.2.1. The following subsections define the base case and ranges of the appropriate parameters. The selections are summarized in Table I.2.7.

I.2.3.1 **Initial Reservoir Pressure**

Two types of data were considered to be best suited for determining initial reservoir pressure. The first is the data on the earliest buildup recorded after encountering the brine reservoir, and the second is the data on the longest buildup recorded.

After the brine reservoir was encountered, WIPP-12 was shut in for 1.43 days. The buildup observed after shut-in was near instantaneous, because only a very limited volume of reservoir fluid had been produced. The maximum pressure observed during this buildup was 1.5 MPa at the surface, a good choice for static reservoir pressure. This pressure corresponds to a reservoir pressure of 12.7 MPa when extrapolated to the center of the brine reservoir at WIPP-12 (918.8 m below the ground surface (BGS)).

The longest buildup period followed Flow Test 3. The flow sequence was 7.0 days in length, followed by a buildup period lasting 278.4 days. For this test, the Horner method (Lee, 1982) was appropriate. By extrapolating to a Horner time of one, an undisturbed reservoir pressure of 1.4 MPa was obtained. This corresponds to an initial pressure of 12.6 MPa at the reservoir center depth of 918.8 m BGS.

Figure I.2.1

TABLE I.2.7 Base-case and range of values of parameters describing the brine reservoir

Parameter	Symbol	Base case	Range	Units
Initial pressure P_i	12.7	7.0 to 17.4	MPa	
Effective thickness	b	7.0	7.0 to 24.0	m
Transmissivity of inner zone	T_i	$7x10^{-4}$	$7x10^{-6}$ to $7x10^{-2}$	m^2/s
Distance to intermediate zone contact	r_2	300	100 to 900	m
Transmissivity of intermediate zone	T_o	$7x10^{-6}$	$7x10^{-8}$ to $7x10^{-4}$	m^2/s
Distance to outer zone contact	r_3	2,000	30 to 8,600	m
Transmissivity of outer zone	T_m	$1x10^{-11}$	Constant	m^2/s
Fluid density	ρ_f	1240	Constant	kg/m^3
Porosity	φ	0.005	0.001-0.01	
Compressibility α	$1x10^{-9}$	$1x10^{-10}$ to $1x10^{-8}$	1/Pa	

Cf. Lappin et al., 1989, Table E-4.

For this modeling study, the base-case reservoir pressure is taken from the highest pressure monitored during the testing of the brine reservoir at WIPP-12, which is equivalent to a pressure of 12.7 MPa at the reservoir center. Of the 13 wells in the northern Delaware Basin that have encountered brine reservoirs, only 4, including WIPP-12, have been tested adequately enough to estimate the formation pressure. These pressures range from 12.6 to 14.3 MPa at formation depth (Popielak et al., 1983). Minimum pressures for nine other wells have been estimated from the minimum pressure needed to allow flow at the surface. From these nine estimates, minimum formation pressures range from 7.0 to 17.4 MPa. The range of initial reservoir pressures for this study is therefore taken to be 7.0 to 17.4 MPa. The base-case value is representative of the WIPP-12 reservoir (for which the best data are available) and is 12.7 MPa at reservoir depth.

I.2.3.2 **Reservoir Thickness**

In most cases, the brine reservoirs encountered in the Castile Formation are in the lower portion of the uppermost anhydrite unit present at that location. The uppermost anhydrite unit at WIPP-12 is Anhydrite III, which is locally 96.6 m thick (Popielak et al., 1983). The WIPP-12 brine reservoir was at the base of Anhydrite III and appears to have been limited to a small fractured zone.

Anhydrite III at WIPP-12 was mapped by coring, caliper logs, acoustic televiewer logs, neutron logs, and spinner logs (Popielak et al., 1983). A review of these observations identified seven megascopic fractures in Anhydrite III. All these fractures were high-angle fractures with dips ranging from 70 degrees to vertical. Only two showed any evidence of brine production, as identified by the spinner log conducted by the USGS (D'Appolonia Consulting Engineers, 1982). The uppermost brine-producing fracture (fracture C) extended from 916.2 to 917.1 m; the lowermost (fracture D) extended from 919.0 to 921.1 m BGS. These depths were taken from the acoustic televiewer log. The spinner log defined the interval from which nearly 100 percent of the flow was coming as that between 916.2 to 921.4 m BGS, which correlates well with both the caliper log and the acoustic televiewer log (Popielak et al., 1983; D'Appolonia Consulting Engineers, 1982).

Because the reservoir is heterogeneous and composed of high-angle fractures, its thickness is difficult to define from borehole reconnaissance at a single location. The base-case effective thickness of the reservoir is estimated to be 7 m and to occur between 915.3 and 922.3 m BGS. The center of the reservoir is taken to be at a depth of 918.8 m BGS, which is the center of the interval that produced nearly all of the inflow at WIPP-12 (D'Appolonia Consulting Engineers, 1982). All downhole pressures are referenced from a depth of 918.8 m BGS. The base-case effective thickness of 7 m shown in Table I.2.7 can be considered a minimum thickness. From the center of the reservoir to the base of Anhydrite III is approximately 12.0 m. The maximum effective thickness will be considered 24 m centered at 918.8 m BGS. Because the product of hydraulic conductivity and thickness (transmissivity) cannot be determined in the reservoir characterization analyses, sensitivity calculations will be performed upon transmissivity. The variation in transmissivity caused by thickness uncertainty will be less than the variation in transmissivity caused by uncertainty in hydraulic conductivity. As a result, the total variation in formation transmissivity will be driven largely by hydraulic conductivity variation, as described in the following subsection.

I.2.3.3 **Reservoir Transmissivity**

For modeling, the WIPP-12 reservoir is conceptualized as being composed of two separate, concentric, fractured media with different transmissivities. Because this modeling study will allow very long flow periods in the brine reservoir, the far-field matrix is also expected to contribute to the reservoir response. This matrix is modeled by attaching an infinite low-transmissivity zone to the outside edge of the intermediate zone through the application of a Carter-Tracy boundary condition (Carter and Tracy, 1960; Reeves et al., 1986). This outermost zone represents the intact Castile Anhydrite III. Popielak et al. (1983) determined that the intact formation matrix had a permeability of less than 2×10^{-19} m^2. Assuming a thickness of 7 m, the transmissivity of the outer zone is equal to approximately 1×10^{-11} m^2/s. The transmissivity of the outer zone is at least six to eight orders of magnitude smaller than the base-case transmissivity of the inner and intermediate zones. For this modeling study, the outer zone represented by the Carter-Tracy boundary condition is assigned a constant transmissivity of 1×10^{-11} m^2/s.

From hydraulic interpretations, it was determined that the inner region of the reservoir can be modeled as a cylindrical zone having a transmissivity of 7×10^{-4} m^2/s and extending out from the well to an effective radius of 300 m. The remainder of the reservoir was interpreted as having a smaller mean transmissivity. This intermediate zone is assigned a lower transmissivity equal to 7×10^{-6} m^2/s out to a radius of 2,000 m. These values are interpreted from WIPP-12 testing and are considered base-case values listed in Table I.2.7 for the hypothetical brine reservoir. The transmissivities of these two zones are estimated to range, somewhat arbitrarily, by two orders of magnitude from the base-case values. The only Castile brine reservoir transmissivity data available for comparison to these base-case values and ranges are presented by Popielak et al. (1983), who determined transmissivities from as low as 1.6×10^{-9} m^2/s at ERDA-6 to as high as 8×10^{-4} m^2/s at WIPP-12.

I.2.3.4 **Reservoir Fluid Density**

The brine from the WIPP-12 brine reservoir has an average level of total dissolved solids of 328,000 mg/L, as determined from laboratory analyses of 13 water samples (Popielak et al., 1983). The average specific gravity, based on 59 field analyses, is 1.215. In addition to these traditional analyses, four borehole-pressure-gradient surveys were performed in 1982 and 1983 at WIPP-12 as part of the hydraulic testing program. These surveys showed pressure gradients ranging from 0.0121 to 0.0123 MPa/m, with an average of 0.0122 MPa/m. This average gradient corresponds to an average fluid density of 1240.6 kg/m^3. For this study, the base-case brine-reservoir fluid density is taken to be 1241 kg/m^3. This parameter will not be varied, and a representative range is not defined.

I.2.3.5 **Reservoir Boundary**

Because of the isolated distribution of brine reservoir encounters in the Castile Formation, the reservoirs must be considered limited, with some outer boundary beyond which hydraulic communication is minimal. Methods used to infer the limits of brine reservoirs in the Castile Formation are varied. One method is to look at a map of wells penetrating the Castile Formation and identify which wells did, and which did not, encounter a brine reservoir. When a well that encountered a brine reservoir is surrounded by wells that did not, the distance of the latter wells from the brine reservoir well represents a maximum radius for the boundary of that reservoir. For example, WIPP-12 is surrounded by four nearby wells that did not encounter brine in the Castile Formation. These four wells range in distance from 2 to 3 km from WIPP-12. Therefore, if it is conservatively assumed that WIPP-12 is located at the center of the reservoir and the reservoir is circular, the WIPP-12 reservoir has at most a 2,000 m radius. Most brine reservoirs in the Castile in the northern Delaware Basin are found to have radii varying from approximately 800 to 3,200 m. Other shapes than circular are possible, but they have not been included in the analysis.

A recent investigation of a different kind that may be used to delineate the extent of the WIPP-12 brine reservoir is a time-domain electromagnetic survey (TDEM) performed at land surface over the waste emplacement panels (Earth Technology Corporation, 1988). This study suggests that there is a low-resistivity body, interpreted as a brine reservoir, within the Castile Formation under portions of the waste emplacement panels. If one assumes that this brine is connected to the WIPP-12 brine reservoir, then one reservoir boundary is at least 1,600 to 2,000 m from WIPP-12.

Another method of inferring the reservoir extent at WIPP-12 is to estimate the total bulk volume of the reservoir using the concept of the storage coefficient of an elastic aquifer. The storage coefficient is defined as the volume of water removed from a vertical column of aquifer of height m and unit basal area when the head declines by one unit (Domenico, 1972). The equation for the storage coefficient can be written as

$$S = b \rho g (\alpha + \varphi \beta) \qquad \text{(I-41)}$$

where b is the aquifer thickness, ρ is fluid density, g is the acceleration due to gravity, α is the compressibility of solids, φ is the porosity, and β is the compressibility of the fluid. Domenico (1972) showed that the amount of water released from storage for a given head decline over an area A is equal to

$$\Delta V = S\ A\ \Delta\ h \qquad \text{(I-42)}$$

where h is the given head decline. Equation (I-42) can be expanded to

$$\Delta V = \rho\ g\ (\alpha + \varphi\ \beta)\ (bA)\ \Delta h \qquad \text{(I-43)}$$

where the product (bA) is equal to the bulk volume of the aquifer (V_b) over which the unit decline in head has occurred and from which water has subsequently been released. Knowing that pressure decline is equal to the product ($\rho g \Delta h$), and solving for the bulk volume of the aquifer, Equation (I-43) can be expressed as

$$V_b = V\ /\ (\alpha + \varphi\ \beta)\ \Delta\ P \qquad \text{(I-44)}$$

Therefore, if the total compressibility of an aquifer and the total pressure change ΔP that has occurred as a result of known fluid volume release (ΔV) are known, an estimate of the total bulk volume of the aquifer can be made. This calculation assumes that the pressure change has been uniform over the total bulk volume and that no mass has been transferred across the aquifer boundaries. Assuming a total thickness and a reservoir geometry, one can estimate the distance to the reservoir boundary.

From the time of initial penetration of the brine reservoir at WIPP-12 to the end of Flow Test 3, a volume of 36,935 m^3 was produced from the reservoir. The residual pressure drop measured at the wellbore at the end of a 278.4 day shut in was 0.23 MPa. For the range of total compressibilities, adopted Equation (I-44) gives a range in aquifer bulk volume of 1.66 x 10^7 to 1.66 x 10^9 m^3. Then, if the brine reservoir is a right-circular cylinder with a range in effective thickness from 7 to 24 m, the estimated reservoir boundary radius is between 460 and 8,600 m. Popielak et al. (1983) reported that at the ERDA-6 reservoir the production of 262.3 m^3 of reservoir fluid resulted in a change in pressure in the aquifer of 0.36 MPa. Using the same ranges of compressibility and effective thickness as above, the range in aquifer bulk volume at ERDA-6 is estimated to be 7.32 x 10^4 to 7.32 x 10^6 m^3. This corresponds to a range in estimated reservoir radius from 30 to 560 m.

The final method considered here for identifying boundaries or large-scale heterogeneities is hydraulic-test interpretation. The hydraulic data do not provide evidence to accurately define the outer boundary location for the WIPP-12 brine reservoir. However, the fact that the reservoir did not recover to static pressure during 278 days of buildup following Flow Test 3 suggests that a boundary may have been encountered in the volume of rock stressed during the testing activities at WIPP-12.

The potential range of reservoir radii based on the minimum and maximum estimates calculated using the various methods is from 30 to 8,600 m. The minimum and maximum estimates, and therefore the range, for reservoir radii come from the calculation based upon estimating the total reservoir bulk volume. The large variation in these estimates comes from the two order of magnitude range in the uncertainty of total aquifer compressibility. Although it is not probable that any of these brine reservoirs have radii as great as 8,600 m, this value will be used to represent a maximum case (i.e., greatest volume). The base-case reservoir radius, taken from the hydraulic interpretations presented in an earlier subsection, is 2,000 m. Because for long flow periods the hydraulic response of the reservoir is a product of the coupled matrix-fracture diffusivities, a Carter-Tracy boundary condition is attached to the peripheral edge of the modeled region. This boundary condition represents the low-permeability far-field anhydrite matrix, which is considered homogeneous and infinite.

I.2.3.6 **Reservoir Porosity**

Porosity determinations were made on the reservoir anhydrite through geophysical logging and laboratory tests. A neutron-porosity log, a gamma-density log, and an acoustic log were used to estimate total porosity within Anhydrite III of the Castile Formation. Estimates of porosity from these logs ranged from 0 to 5 percent (Popielak et al., 1983). Laboratory porosity determinations were also performed on two intact pieces of core from Anhydrite III. The first piece of core (from 858 m BGS) had a porosity of 0.8 percent, and the second piece (from 916.5 m BGS) had a porosity of 0.2 percent (Popielak et al., 1983).

The brine occurrence appears to be associated with a zone of secondary porosity resulting from the deformation fracturing of the brittle anhydrites. A medium like this is often characterized by very low secondary porosities and high transmissivities. Because we have no accurate means of estimating secondary porosity for this medium, this modeling will adopt the same range of secondary porosity of 0.1 to 1.0 percent that was used by Popielak et al. (1983). The base-case value of porosity is chosen to be 0.5 percent.

I.2.3.7 **Reservoir Compressibility**

In this study, the specific storage is calculated in the classical hydrogeologic representation where the medium compressibility is normalized with respect to the bulk volume (Narasimhan and Kanehiro, 1980). The medium compressibility (α) for a triaxial system can be defined as the inverse of the bulk modulus (B) of the rock

$$\alpha = (1/B) \tag{I-45}$$

The bulk compressibility of Anhydrite III was estimated from values of the bulk modulus determined from acoustic logs that were run in WIPP-12. Popielak et al. (1983) reported that the values of bulk modulus for Anhydrite III range from 3.45×10^{10} to 6.89×10^{9} Pa. This represents a compressibility range from 2.9×10^{-11} to 1.45×10^{-10} Pa^{-1}. These values are considered representative of the intact anhydrite. The compressibility of a fractured or jointed rock is generally an order of magnitude higher than that of a non-fractured rock (Freeze and Cherry, 1979). Freeze and Cherry give a range for medium compressibility for a fractured or jointed rock from 1×10^{-8} to 1×10^{-10} Pa^{-1}. This study adopts this range of compressibility and takes 1×10^{-9} Pa^{-1} to be the base-case value.

I.2.4 **BOREHOLE PARAMETERS**

In Case II, a borehole is assumed to be drilled through the repository and to encounter a brine reservoir within the Castile Formation. In this subsection, the borehole location and properties used in the simulations are discussed. The respective parameters are summarized in Table I.2.8.

The borehole is assumed to pass through the center of the southwestern waste panel. This is conservative in that travel times in the Culebra aquifer calculated using the groundwater flow model (LaVenue et al., 1988) are shortest between the southwestern corner of the waste disposal area and off-site locations such as that of the hypothetical stock well.

Elevations of the ground surface and the Rustler units are based on the H-3 hydropad because of its nearness. The elevation for the center of the facility is taken from Bechtel (1985). Interpolation between WIPP-12 and Cabin Baby-1 was used to determine the elevation of the Salado-Castile contact at the breach-borehole location. The elevation for the center of the brine reservoir is based on interpolation between the Anhydrite III elevation at WIPP-12 and Cabin Baby-1 and the relative position of the brine reservoir in Anhydrite III at WIPP-12. A schematic representation of the borehole showing elevations and thicknesses of the various units of interest and the locations of the 60-m long borehole plays are shown in Figure I.2.2.

TABLE I.2.8 Specifications for intrusion borehole for Case II simulations

Parameter	Value	Units
Borehole UTM location at center of southwestern waste panel (Case I)	613324 3581146	m E m N
Revised location (Case II)	613331 3581141	m E m N
Elevations		
Ground surface	1033	m
Center of Culebra	825	m
Rustler-Salado contact	783	m
Center of waste panel	381	m
Salado-Castile contact	181	m
Center of brine reservoir	109	m
Drilled diameter		
In Rustler (oil well)	0.413	m
In Rustler (gas well)	0.457	m
In Salado and Castile (oil well)	0.311	m
In Salado and Castile (gas well)	0.356	m
Hole diameters used in numerical analysis		
Cased inside diameter of average hole in Rustler	0.326	m
Diameter of average borehole in Salado and Castile	0.334	m
Borehole plugs		
Lengths	60	m
Locations (above brine reservoir, below potash zone, and below Rustler)		
Effective borehole permeability		
Open borehole period	infinite	
Plug in Castile	10^{-15}	m^2
Plugs in Salado	10^{-18}	m^2
For times greater than 150 years		
Case IIA	10^{-12}	m^2
Cases IIB and IIC	10^{-11}	m^2

Cf. Lappin et al., 1989, Table E-2.

Figure I.2.2

Borehole diameters for oil and gas wells are 0.413 m (16-1/4 inch) and 0.457 m (18 inch), respectively, from ground surface to the top of the Salado Formation and 0.311 m (12-1/4 inch) and 0.356 m (14 inch), respectively, below the top of the Salado Formation.

In the SWIFT II simulations, the hydraulic conductance of the borehole between the brine reservoir and the Culebra is used as input for modeling the hydraulic coupling of the two water-bearing horizons. The hydraulic conductance is defined in terms of a transmissibility T_w by the relation

$$T_w = kA/L \tag{I-46}$$

where k is the effective borehole permeability, A is the borehole cross-sectional area, and L is the length of the plugs or "rubbleized" borehole zones. Since two potential borehole diameters are possible in exploratory drilling of oil and gas wells, an average cross-sectional area for these two well types was used. An effective borehole hydraulic conductance was calculated as a harmonic average (i.e., $K_{ave} = [(1/K_1 + 1/K_2)/2]^{-1}$) using the appropriate borehole or plug lengths with specific permeabilities and cross-sectional areas. For modeling purposes, it is assumed that the plugs remain intact for the first 75 years after emplacement, and then their transmissibilities increase linearly until 150 years, when the effective borehole permeability is 10^{-12} m^2 for Cases IIA and IIA (rev) and 10^{-11} m^2 for Cases IIB, IIC, IIC (rev), and IID. This time-varying transmissibility is implemented stepwise with equal increments to the transmissibility at 75, 100, 125, and 150 years.

I.2.5 REPOSITORY SOURCE-TERM PARAMETERS

The parameters necessary for quantifying the source term to the Culebra aquifer for the Case II simulations using SWIFT II are summarized in Tables I.2.9, I.2.10, and I.2.11.

Waste Species and Mass Inventory. Calculations were performed for four radioactive decay chains (^{240}Pu, ^{239}Pu, ^{238}Pu, and ^{241}Am) and stable lead for a time period of 10,000 years. The initial total waste inventories for the decay-chain members of interest and stable lead in the repository are presented in Tables I.2.10 and I.2.11.

Calculations of brine inflow in Cases IIA, IIB, and IIC indicate an average value of 1.3 m^3/yr for one panel of seven rooms plus accessways. In cases IIA (rev) and IIC (rev), this value was adjusted to 1.4m^3/yr. In the Case II simulations, this flux is added to the flux from the brine reservoir to the Culebra. As a consequence of the different specified hydrologic properties of the rooms for Cases IIA and IIB, the mass loading to the borehole is different for the two situations (see Table 5.57). For Cases IIA, IIA(rev), IIC, and IIC(rev), all fluid flowing from the Castile brine reservoir is assumed to have access to the waste mass in one panel, whereas in Cases IIB and IID, the fluid from the Castile brine reservoir does not mix with the waste. Therefore, in Cases IIB and IID the only fluids reaching the Culebra are uncontaminated brine from the Castile and contaminated brine from the Salado. In the first three cases, 1.3 m^3/yr of Salado brine inflow (1.4 m^3/yr for the revised cases) that has contacted the waste mass is specified to enter the borehole and flow to the Culebra aquifer. In Case IID, 0.1m^3/yr of Salado brine, from one room only, passes through the waste mass to the borehole.

TABLE I.2.9 Specifications for repository parameters used in Case II simulations

Parameter	Symbol	Base Case	Units
Soluble radionuclide concentration for each decay-chain member in			
Cases IIA, IIA (rev), and IID	C_s	1×10^{-6}	molar
	C_s	2.4×10^{-7}	kg/kg
Cases IIB, IIC, and IIC (rev)	C_s	1×10^{-4}	molar
	C_s	2.4×10^{-5}	kg/kg
Soluble stable-Pb concentration			
in repository	C_s	1.16×10^{2}	mg/L
	C_s	1.16×10^{-4}	kg/kg
in Culebra	C_s	5.4×10^{1}	mg/L
	C_s	5.4×10^{-5}	kg/kg
Mass in initial waste inventory	M_i	Reported in Table I.2.10	g
Mass in initial waste inventory (Case II revised)	M_i	Reported in Table I.2.11	g
Mass of waste in contact with circulating fluids after borehole is plugged	---	$M_i/8$	g
Mass of waste in southwestern waste panel in contact with circulating fluids after borehole is plugged (Case II revised)	---	$4.6M_i/43.5$	g
Pore volume in southwestern waste panel (Case II revised)	---	1,330	m^3
Fluid loading from repository to the borehole (q)			
Cases IIA, II, and IIC	---	1.3	m^3/yr
Case IID	---	0.1	m^3/yr
Cases IIA (rev) & IIC (rev)	---	1.4	m^3/yr

Cf. Lappin et al., 1989, Table E-1.

Note. Based on the specified radionuclide solubilities expressed as molarity, solubility values expressed as kg/kg have about a 6 percent range. Because of the large uncertainty in molarity values, a single solubility value for all radionuclides was used in numerical simulations.

TABLE I.2.10 Specification of mass inventory of waste radionuclide species and stable lead in the repository for the Case II simulations

Decay chain or waste species	Nuclide	Half-life (years)	Initial waste (Ci)	Inventory (g)
$^{240}Pu \rightarrow {}^{236}U$	^{240}Pu	6.54×10^3	1.05×10^5	$4.76 x 10^5$
	^{236}U	2.34×10^7	0	0
^{239}Pu	^{239}Pu	2.41×10^4	4.25×10^5	6.93×10^6
$^{238}Pu \rightarrow {}^{234}U \rightarrow {}^{230}Th \rightarrow {}^{226}Ra \rightarrow {}^{210}Pb$	^{238}Pu	87.7	3.90×10^5	2.31×10^5
	^{234}U	2.44×10^5	0	0
	^{230}Th	7.7×10^4	0	0
	^{226}Ra	1.6×10^3	0	0
	^{210}Pb	22.3	0	0
$^{241}Am \rightarrow {}^{237}Np \rightarrow {}^{233}U \rightarrow {}^{229}Th$	^{241}Am	4.32×10^2	7.75×10^5	2.26×10^5
	^{237}Np	2.14×10^6	8.02	1.14×10^4
	^{233}U	1.59×10^5	7.72×10^3	8.15×10^5
	^{229}Th	7.43×10^3	0	0
Stable Pb	n.a.	n.a.	n.a.	1.33×10^9

Cf. Lappin et al., 1989, Table E-5.

Note. n.a. means not applicable.

TABLE I.2.11 Specification of mass inventory of waste radionuclide species and stable lead in the repository for the Case II(rev) simulations

Decay chain or waste species	Radionuclide	Half-life (years)	Ci/g	Initial inventory* (g)	Inventory at 175 Years[a] (g)
$^{240}Pu \rightarrow ^{236}U$	^{240}Pu	6.54×10^{3}	2.28×10^{1}	5.27×10^{5}	5.17×10^{5}
	^{236}U	2.34×10^{7}	6.47×10^{-5}	0	9.52×10^{3}
^{239}Pu	^{239}Pu	2.41×10^{4}	6.21×10^{-2}	7.88×10^{6}	7.84×10^{6}
$^{238}Pu \rightarrow ^{234}U \rightarrow ^{230}Th$	^{238}Pu	8.77×10^{1}	1.71×10^{1}	3.06×10^{5}	0[b]
$\rightarrow ^{226}Ra \rightarrow ^{210}Pb$	^{234}U	2.44×10^{5}	6.26×10^{-3}	0	3.01×10^{5}
	^{230}Th	7.70×10^{4}	2.02×10^{-2}	0	0[c]
	^{226}Ra	1.60×10^{3}	9.89×10^{-1}	0	0[d]
	^{210}Pb	2.23×10^{1}	7.64×10^{1}	0	0[d]
$^{241}Pu \rightarrow ^{241}Am \rightarrow ^{237}Np$	^{241}Pu	1.44×10^{1}	1.03×10^{2}	4.56×10^{4}	0[b]
$\rightarrow ^{233}U \rightarrow ^{229}Th$	^{241}Am	4.32×10^{2}	3.43×10^{0}	2.25×10^{5}	2.06×10^{5}
	^{237}Np	2.14×10^{6}	7.05×10^{-4}	1.53×10^{4}	7.93×10^{4}
	^{233}U	1.59×10^{5}	9.65×10^{-3}	9.82×10^{5}	9.81×10^{5}
	^{229}Th	7.43×10^{3}	2.10×10^{-1}	0	0[c]
Stable Pb	n.a.	n.a.	n.a.	1.33×10^{9}	1.33×10^{9}

Cf. Lappin et al., 1989, Table E-5.

Note. n.a. means not applicable.

* Initial inventory in Ci is presented in Table B.2.13.

[a] The transport calculations start 175 years after the beginning of institutional control.

[b] Because ^{238}Pu and ^{241}Pu have short half-lives and large retardation factors, their migration from the source is minimal. Therefore, the conservative approach converts all ^{238}Pu and ^{241}Pu to daughter products at simulation beginning.

[c] Because of large retardation factors relative to their parents, ^{230}Th and ^{229}Th migration is controlled by their parents. Because of this fact and the fact that both nuclides have very little mass in place at 175 years, they are not considered initially present at 175 years.

[d] These nuclides are not present in quantities large enough at 175 years to warrant source inclusion.

Source Term in the Repository. The concentration of the waste species in these fluids is constrained by their solubilities. For the radionuclides, the solubilities were set equal to 10^{-6} molar for Cases IIA, IIA (rev), and IID and to 10^{-4} molar for Cases IIB, IIC, and IIC (rev). The solubility for stable Pb was set at 116 mg/L in the repository fluids. All fluids entering the borehole from the waste panel had concentrations at these values except as modified by radioactive decay and the total mass available in one panel. The solubility of stable lead in the Culebra groundwaters was specified at 54 mg/L.

I.2.6 CULEBRA PARAMETERS

A fractured, porous medium is assumed to exist along the travel path between the breach borehole and the stock well. The definition of the flow path, the stock-well location, and the solute-transport properties within the Culebra are discussed below. Additional discussion on fracturing in the Culebra and its effect on hydraulic and tracer tests is presented in Reeves et al. (1987). The base case and range of values for the Culebra parameters are summarized in Table I.2.12. The range of values is presented for discussion purposes only. They are not used in the Case IIA and IIA (rev) simulations. For Cases IIB, IIC, IIC (rev), and IID, lower or higher end values of the range were selected, whichever would result in more rapid or longer distance solute transport.

A double-porosity flow is assumed along the travel path. The double-porosity data base is limited; base case and ranges of parameter values are documented using available data, but must be considered as uncertain.

Regional Flow Field. A review of the hydrologic modeling for the Culebra in the vicinity of the WIPP site is discussed in Lappin et al. (1989, Section 3.3.5). The Culebra groundwater flow model by LaVenue et al. (1988) was used in Cases IIA, IIB, IIC, and IID for estimating the Darcy velocity distribution in the regional flow field and for determining the travel path from the borehole to the stock well. Calibration of the model included hydrologic data available up to about October 1987. The model was calibrated to undisturbed head conditions only and did not include data from the large-scale multipad pumping tests that have been performed at the WIPP site. For Cases IIA (rev) and IIC (rev), this flow field description was updated to include all data collected through June 16, 1989. (See Subsection 4.3.3.2.)

As discussed above in Subsection I.2.4, the borehole is assumed to be drilled through the center of the southwestern waste panel. A particle-tracking code was used to determine the flow path for transport from this release location to a hypothetical stock well. The location of the stock well was based on two constraints: the well is assumed to lie on a flow path from the breach borehole, and the well must be located in an area where the water is potentially fresh enough to support stock.

TABLE I.2.12Parameter base-case and range values selected for the Culebra dolomite

Parameter	Symbol	Base Case	Range	Units
Free-water diffusivity	D_	$5x10^{-6}$	$5x10^{-7}$ to $9x10^{-5}$	cm^2/s
Radionuclides: Case IIA	D_	$1x10^{-6}$	n.a.	cm^2/s
Cases IIB, IIC, IID	D_	$5x10^{-7}$	n.a.	cm^2/s
Cases IIA (rev) and IIC (rev)		See Table I.2.13		
Stable Pb: Case IIA	D_	$4x10^{-6}$	n.a.	cm^2/s
Cases IIB, IIC, IID	D_	$1x10^{-6}$	n.a.	cm^2/s
Matrix tortuosity		0.15	0.03-0.5	
Case IIA, IIA (rev)		0.15	n.a.	
Cases IIB, IIC, IIC (rev), IID		0.03	n.a.	
Fracture spacing	2L_	2.0	0.25-7.0	m
Cases IIA, IIA (rev)	2L_	2.0	n.a.	m
Cases IIB, IIC, IIC (rev), IID	2L_	7.0	n.a.	m
Porosity	φ_	0.16	0.07-0.30	
Cases IIA, IIA (rev)	φ_	0.16	n.a.	
Cases IIB, IIC, IIC (rev), IID	φ_	0.07	n.a.	
Fracture porosity	φ_	$1.5x10^{-3}$	$1.5x10^{-4}$ to $1.5x10^{-2}$	
Longitudinal dispersivity	∝	100	50 to 300	m
Matrix distribution coefficient				
Case IIA: Plutonium	K_d	50	-	ml/g
Americium	K_d	200	-	ml/g
Uranium	K_d	1	-	ml/g
Neptunium	K_d	1	-	ml/g
Thorium	K_d	50	-	ml/g
Radium	K_d	0.1	-	ml/g
Lead	K_d	0.1	-	ml/g
Cases IIB, IIC, IID				
Plutonium	K_d	25	-	ml/g
Americium	K_d	100	-	ml/g
Uranium	K_d	1	-	ml/g
Neptunium	K_d	1	-	ml/g
Thorium	K_d	25	-	ml/g
Radium	K_d	0.05	-	ml/g
Lead	K_d	0.05	-	ml/g

Cf. Lappin et al., 1989, Table E-6.

Note: The Culebra groundwater flow model presented in LaVenue et al. (1988) was used for calculating fluxes and determining flow paths. The transient fracture flux along the flow path from the release point in the Culebra aquifer to the off-site stock well is calculated through hydraulic coupling of the brine reservoir, borehole region, and Culebra aquifer.

Free-Water Diffusivity. Base-case and range values for free-water diffusion coefficients for the radionuclides of interest and stable lead are presented in Lappin et al. (1989; Subsection E.2.4.2). For the calculations reported in the draft SEIS, a single value was necessary for all members of a decay chain because of the numerical formulation of the SWIFT II model. For Case IIA, values of $1x10^{-6}$ cm^2/s and $4x10^{-6}$ cm^2/s are selected for the radionuclides and stable lead, respectively. Values a factor of two smaller were used for the Case IIB, IIC, and IID simulations. For Cases IIA (rev) and IIC (rev), improvements in the SWIFT II codes permitted species-specific diffusion coefficients (Table I.2.13).

The base-case and range of values selected for this study (Tables I.2.12 and I.2.13) are substantially lower than those in Reeves et al. (1987) for two reasons: 1) the previous study did not specifically address the radioactive decay-chain members identified in the present study, and 2) the much higher salinities that are a result of flow from the Salado and Castile can cause a reduction in the free-water diffusivity by as much as a factor of two.

Matrix Porosity. Porosities have been measured in the laboratory for 82 core samples of Culebra dolomite from 15 borehole or hydropad locations at and surrounding the WIPP site. The results are summarized in Table I.2.14. Porosities were measured by the Boyle's Law technique using helium or air on all samples and by the water-resaturation technique on 30 samples. An excellent correlation was obtained between porosity values from the two techniques. From the 82 samples with porosity measurements using the Boyle's Law technique, an average porosity of 15.2 percent was obtained with a range from about 3 to 30 percent. For comparison, core samples from the H-3 and H-11 hydropads, which are the two hydropads closest to the off-site pathway, had average porosities of 19.8 percent (6 samples) and 16.2 percent (10 samples), respectively. Porosities ranged from about 17 to 24 percent for the H-3 hydropad and about 10 to 30 percent for the H-11 hydropad.

Matrix porosities of Culebra dolomite measured by Sandia National Laboratories using the ^{22}Na diffusion technique range from 1.1 to 8.7 percent. Corresponding tortuosities range from 0.03 to 0.09. The porosities calculated from the diffusion experiments are termed diffusion-porosity values and are lower than those measured by Boyle's Law or mercury-porosimetry techniques. These values lie at the lower end of the range of values shown in Table I.2.14. Possible explanations for the differences between values measured by these different techniques include sample heterogeneity, incomplete resaturation of previously dried samples, and deviations of actual pore geometry from the idealized model assumed in simple versions of Fick's First Law of Diffusion for solute migration in a porous rock. In general, the samples used in the diffusion measurements are fine-grain dolomites free from large cracks and are chosen for mineral homogeneity and structural competence. No claim has been made that these samples are representative of the Culebra dolomite in general or that these results are transferable to field-scale transport.

For transport along the off-site pathway in the Culebra, a base-case matrix porosity of 16 percent is chosen for the Cases IIA and IIA (rev) simulations. For Cases IIB, IIC, IIC (rev), and IID, a matrix porosity of 7 percent is selected as a lower end value. Boguspagexxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxx xxxxxxxxxxxxxxxxxxxxxxxxxxxxxx

Table I.2.13 Free-water diffusion coefficients (cm^2/s) for radionuclides and stable lead for the Case II simulations

Element	Case IIA (rev)	Case IIC (rev)	Range of Values in Literature[a]
Pu	1.7×10^{-6}	8.5×10^{-7}	4.8×10^{-7} - (3×10^{-6})
Am	1.8×10^{-6}	9.0×10^{-7}	5.3×10^{-7} - (3×10^{-6})
U	2.7×10^{-6}	1.4×10^{-6}	1.1×10^{-6} - $\underline{4.3 \times 10^{-6}}$
Np	1.8×10^{-6}	9.0×10^{-7}	5.2×10^{-7} - (3×10^{-6})
Ra	3.8×10^{-6}	1.9×10^{-6}	$\underline{7.5 \times 10^{-6}}$
Pb	4.0×10^{-6}	2.0×10^{-6}	$\underline{8 \times 10^{-6}}$
Th	1.0×10^{-6}	5.0×10^{-7}	5×10^{-7} - $\underline{1.53 \times 10^{-6}}$

[a] Data from values compiled by Brush (1988) (indicated by parentheses); values calculated from the Nernst expression by Li and Gregory (1974) (underlined); and measurements by Torstenfelt et al. (1982) (all others). Temperature dependence has not been considered for the recommended values. Literature values are further discussed in Lappin et al. (1989), Section E.2.4.2.

Cf. Lappin et al., 1990.

Table I.2.14Summary of porosities measured in Culebra core samples

Well	Sample Identification Number	Porosity Determination (%) Helium or Air	Water Resaturation
H-2a	-1	11.6	11.3
H-2a	-2	12.2	
H-2b	1-1	14.1	
H-2b	2-1, 3-1	15.4	
H-2b	1-2	11.8	
H-2b	2-2, 3-2	10.3	
H-2b1	-1F	10.5	
H-2b1	-1	8.2	8.8
H-2b1	-2	14.2	
H-2b1	-3	15.3	15.8
H-3b2	1-3	18.8	
h-3b2	1-4	16.8	
H-3b3	2-3, 3-3	18.0	
H-3b3	2-4, 3-4V	20.2	
H-3b3	1-6	24.4	
H-3b3	2-5, 3-5	20.5	
H-4b	1-9	29.7	
H-4b	2-6, 3-6V	20.8	
H-5b	-1	12.5	
H-5b1	-1A	13.0	
H-5b1	-IB	15.6	
H-5b1	-2	22.8	23.7
H-5b1	-2F	24.8	
H-5b1	-3	13.3	12.8
H-6b	2-7	10.8	
H-6b	2-8	11.6	
H-6b	1-7	10.7	
H-6b	1-8	25.5	
H-7b1	-1	17.7	18.1
H-7b1	-1F	14.9	
H-7b1	-2A	20.6	
H-7b1	-2B	27.8	
H-7b2	-1	15.9	14.8
H-7b2	-2	11.8	12.9
H-7c	-1A	12.5	12.9
H-7c	-1B	16.5	
H-7c	-1C	13.4	
H-7c	-1F	13.8	
H-10b	-1	8.9	
H-10b	-2	11.5	11.7
H-10b	-2F	6.6	
H-10b	-3	11.2	10.6

TABLE I.2.14   Concluded

Porosity Determination (%)

Sample

Well	Identification Number	Helium or Air	Water Resaturation
H-11	-1	15.5	15.3
H-11	-2	10.5	11.3
H-11	-2F	10.4	
H-11b3	-1	30.3	27.5
H-11b3	-1F	22.3	
H-11b3	-2	9.9	10.3
H-11b3	-2F	12.3	
H-11b3	-3	13.0	12.6
H-11b3	-4	15.2	
H-11b3	-4F	22.4	
W-12	-1A	2.8	
W-12	-1B	11.4	
W-12	-2	11.6	11.9
W-12	-2B	12.6	
W-12	-2F	13.5	
W-12	-3	13.4	13.0
W-13	-1	14.3	15.2
W-13	-2	21.9	22.6
W-13	-2F	26.0	
W-13	-3A	17.9	
W-13	-3B	9.7	
W-25	-1	11.5	12.0
W-26	-1	12.4	12.2
W-26	-1F	11.2	
W-26	-2	12.6	12.6
W-26	-3	12.7	
W-28	-1A	14.2	
W-28	-1B	13.0	
W-28	-2	18.7	18.8
W-28	-3	17.0	16.9
W-28	-3F	17.9	
W-30	-1	12.8	12.4
W-30	-2	15.0	15.2
W-30	-3A	17.6	
W-30	-3B	14.9	
W-30	-3F	14.9	
W-30	-4	23.9	
AEC-8	-1	7.9	8.6
AEC-8	-1F	12.2	
AEC-8	-2	10.9	10.6

Number of Samples = 82
Average = 15.2%
Standard Deviation = 5.3%
Range = 2.8 to 30.3%

Cf. Lappin et al., 1989, Table E-8.

Although lower values have been measured or derived, an average lower value of 7 percent along the flow path is considered most representative.

Matrix Tortuosity. Tortuosity values for dolomite are not available, although a review of the literature does permit an estimation of a potential range. Bear (1972), in his review of unconsolidated media, presents values ranging from 0.3 to 0.7. De Marsily (1986) reports tortuosities varying from 0.1 for clay to 0.7 for sand. Barker and Foster (1981) report diffusion coefficients for Cl^- in chalk samples that indicate tortuosities of 0.02 to 0.17. Katsube et al. (1986) calculate tortuosity values from 0.02 to 0.19 from diffusion experiments on crystalline-rock samples. As noted earlier, diffusion experiments performed by Sandia National Laboratories on a limited number of core samples have yielded tortuosities in the range of 0.03 to 0.09.

Matrix tortuosity estimates for the Culebra were calculated based on formation-factor and matrix-porosity determinations on 15 core samples. The values, ranging from 0.03 to 0.33 with an average value of 0.14, are summarized in Table I.2.15.

For the Case IIA and IIA (rev) simulations, a base-case matrix tortuosity of 0.15 was selected as representative. This value is the same as that used in the regional-scale transport simulations presented in Reeves et al. (1987). A lower-end estimate of 0.03 for matrix tortuosity was selected for the Case IIB, IIC, IIC (rev), and IID simulations.

Rock Density. Rock-density determinations were performed on 73 Culebra core samples from 15 borehole or hydropad locations. The values range from 2.78 to 2.84 g/cm^3 with an average and standard deviation of 2.82 and 0.02, respectively. A value of 2.82 g/cm^3 was chosen as the base-case value for all simulations.

Fracture Porosity. Estimates of the fracture porosity can be obtained by interpreting tracer tests conducted at sites exhibiting double-porosity transport behavior. Tracer tests have been performed at five locations (H-2, H-3, H-4, H-6, and H-11 hydropads) at the WIPP site. Of these, the tests conducted at the H-3, H-6, and H-11 hydropads appear to demonstrate fracture-transport behavior as evidenced by the very rapid tracer breakthrough between wells on at least one flow path at each hydropad site. Detailed test interpretations have been reported for only the H-3 hydropad (Kelley and Pickens, 1986).

A first estimate of the fracture porosity can be calculated from the convergent-flow tracer tests by the relation

$$\varphi_f = Q\, t_p / \pi\, r_t^2\, b \qquad \text{(I-47)}$$

where φ_f is the fracture porosity, Q is the discharge rate at the pumping well, t_p is the time to reach the peak concentration, r_t is the distance between the tracer-addition and pumping wells, and b is the aquifer thickness.

The time to reach the peak concentration is used in this estimation procedure because it is assumed that this time is representative of the average transport rate between the tracer-addition and pumping wells. Although the time to reach the peak concentration is also dependent on longitudinal dispersivity and diffusive losses to the matrix, this approach provides a first estimate of fracture porosity for calibration of the tracer-breakthrough curves.

TABLE I.2.15 Summary of formation factors and calculated tortuosities from Culebra core samples

Well	Identification Number	Helium Porosity (%)	Formation Factor	Calculated Tortuosity
H-2b1	-1F	10.5	326.77	0.03
H-5b1	-2F	24.8	12.20	0.33
H-7b1	-1F	14.9	73.49	0.09
H-7c	-1F	13.8	79.61	0.09
H-10b	-2F	6.6	406.78	0.04
H-11	-2F	10.4	94.82	0.10
H-11b3	-1F	22.3	36.35	0.12
H-11b3	-2F	12.3	101.93	0.08
H-11b3	-4F	22.4	32.74	0.14
W-12	-2F	13.5	47.30	0.16
W-13	-2F	26.0	13.26	0.29
W-26	-1F	11.2	68.77	0.13
W-28	-3F	17.9	26.30	0.21
W-30	-3F	14.9	31.49	0.21
AEC-8	-1F	12.2	90.09	0.09
-Average		15.6%	96.13	0.14
range		6.6 to 26.0%		0.03 to 0.33

Cf. Lappin et al., 1989, Table E-9.

The calculated fracture porosities for the flow paths exhibiting the strongest fracture control are 2×10^{-3} and 1×10^{-3} for the H-3 and H-11 hydropads, respectively. Since these two hydropads are closest to the off-site transport pathway, an average value of 1.5×10^{-3} was selected as the base-case fracture porosity.

Matrix-Block Length. The fractured Culebra dolomite is conceptualized in this study as consisting of three orthogonal fracture sets that define rectilinear matrix units. Both horizonal and vertical (or near vertical) fracture sets have been observed in core samples, shaft excavations, and in outcrop areas (Kelley and Pickens, 1986). The matrix-block sizes are expected to vary spatially across the WIPP site. However, since the matrix-block-size data base is so limited at the present time, the effects of this variability cannot be assessed. Therefore, this study analyzes double-porosity effects in terms of an "equivalent" block size assumed to be applicable over the entire length of the travel path.

Block sizes have been interpreted for the H-3 hydropad in the range of 0.5 to 2.4 m for the two travel paths at the hydropad (Kelley and Pickens, 1986). While detailed interpretations have not been completed for the H-11 hydropad tracer test, a preliminary evaluation of the breakthrough curves suggests matrix-block sizes in the range of 0.8 to 3 m. A base-case value of 2 m is selected for matrix-block length, with a range of values of 0.25 to 7 m. There is no physical significance to the value of 7 m chosen as the upper limit for fracture spacing. It simply corresponds to a representative measured thickness for the Culebra dolomite. Base case matrix-block size of 2 m was selected for the Case IIA and IIA (rev) simulations and the upper end matrix-block size of 7 m was selected for the Case IIB, IIC, IIC (rev), and IID simulations.

Longitudinal Dispersivity. A review of the literature for various tracer-test scales and contaminant-plume sizes (e.g., Lalleman-Barres and Peaudecerf, 1978; Pickens and Grisak, 1981) suggests that, up to moderate travel distances of 500 to 1,000 m, longitudinal dispersivity can be expressed as a function of the mean travel distance of the solute. Longitudinal dispersivity, as indicated by these authors, ranges from several to 10 percent of the travel-path length. Although it is assumed that longitudinal dispersivity is directly related to the mean travel distance of the solute, one would not expect the longitudinal dispersivity to increase beyond some maximum or asymptotic value. This study adopts a range of 50 m to 300 m, i.e., approximately 1.5 to 9 percent of the average path length (3,280 m), with a base-case value equal to 100 m.

Matrix Distribution Coefficients. Estimates of the distribution coefficients (K_d) for the radionuclides and stable Pb, describing their interaction with the Culebra under Case II conditions, are presented and discussed in Lappin et al. (1989, Appendix E). There is a considerable uncertainty in defining representative K_d values for the waste species of interest; however, estimates were based on the limited data available. The values used in the Case IIA through Case IID simulations are summarized in Table I.2.12.

I.2.7 **LOCATION OF THE STOCK WELL**

For the Case II calculations, the specified release point to the biosphere from the Culebra is a hypothetical stock well. The location of this well is constrained by two factors. First, the well must lie on one of the principal flow paths leaving the WIPP site. Second, the well must be located in an area where the water is sufficiently fresh (i.e., TDS < 10,000 mg/L) to support stock.

At the WIPP site itself, the water in the Culebra carries too great a burden of total dissolved solids (TDS) to be usable, even for stock; these levels range from 16,000 to nearly 150,000 mg/L. Water quality improves to the south. At a distance of 14 km, the TDS levels are down to about 3,000 mg/L. Unfortunately, there is a 9-km gap between the few test wells near the south edge of the site (wells P-17, H-17, and H-12) and the next wells to the south (wells H-9 and Cabin Baby). The closest possible position at which a livestock well might yield Culebra water with an acceptable TDS level must be somewhere in this gap. This closest possible position was estimated by using the maximum water-quality gradient in the immediate site area, where there are enough data to determine these gradients reliably.
The hypothetical stock well used in the SEIS calculations is 5 km south of the site.

Probably the actual nearest location to the south where acceptable water can really be found is somewhat more distant than this. The present-day solute distribution in the Culebra is not static; solutes will redistribute as time passes as the result of groundwater flow. Given the presence of relatively dense, high-TDS water north of the selected stock well discharge point, it is expected that the long-term water quality changes at the hypothetical well location will be in the form of a very slow increase in TDS. This suggests that the length of the travel path required to reach potable water to the south will increase with time, making the stock well location selected for this SEIS conservative with respect to long-term salinity changes, i.e., exposures to lead and radionuclides reported here will be over-estimated.

REFERENCES FOR APPENDIX I

Abitz et al. (R. J. Abitz, J. Myers, and P. Drez), 1989. "Composition of Salado Formation Brines from the WIPP Repository Horizon," DOE-WIPP Publication, in review.

ATSDR (Agency for Toxic Substances and Disease Registry), 1988. "Toxicological Profile for Lead," Oak Ridge National Laboratory, Oak Ridge, Tennessee, draft.

Baes et al. (C. F. Baes, III, R. D. Sharp, A. L. Sjoreen, and R. W. Shor), 1984. A Review and Analysis of Parameters for Assessing Transport of Environmentally Released Radionuclides Through Agriculture, ORNL-5786, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Barker, J.A., and S.S.D. Foster, 1981. "A Diffusion Exchange Model for Solute Movement in Fissured Porous Rock," Quarterly Journal of Engineering Geology, London, Vol.14, p.17-26.

Bear, J., 1972. Dynamics of Fluids in Porous Media, Dover Publications Inc., New York, New York, p. 764.

Bechtel National Inc., 1985. Quarterly Geotechnical Data Report, DOE/WIPP-221, U.S. Department of Energy, Carlsbad, New Mexico.

Botts, R. P., 1977. The Short-Term Effects of Lead on Domestic and Wild Animals, EPA-600/3-77-009, Corvallis Environmental Research Laboratory, Corvallis, Oregon.

Brookins, D. G., 1988. Eh-pH Diagrams for Geochemistry, Springer-Verlag, New York, New York.

Brush, L. H., 1988. Feasibility of Disposal of High-Level Radioactive Wastes into the Seabed: Review of Laboratory Investigations of Radionuclide Migration Through Deep-Sea Sediments, SAND87-2438, Sandia National Laboratories, Albuquerque, New Mexico.

Campbell et al. (J. E. Campbell, D. E. Longsine, and M. Reeves), 1986. Risk Methodology for Geologic Disposal of Radioactive Waste: The Distributed Velocity Method of Solving the Convective-Dispersion Equation, SAND80-0717, NUREG/CR-1376, Sandia National Laboratories, Albuquerque, New Mexico.

Campbell et al. (J. E. Campbell, D. E. Longsine, and R. M. Cranwell), 1981. Risk Methodology for Geologic Disposal of Radioactive Waste: The NWFT/DVM Computer Code User's Manual, SAND81-0886, NUREG/CR-2081, Sandia National Laboratories, Albuquerque, New Mexico.

Campbell et al. (J. E. Campbell, P. C. Kaestner, B. S. Langkopf, and R. B. Lantz), 1980. Risk Methodology for Geologic Disposal of Radioactive Waste: The Network Flow and Transport (NWFT) Model, SAND79-1920, NUREG/CR-1190, Sandia National Laboratories, Albuquerque, New Mexico.

Carslaw, H. S. and J. C. Jaeger, 1959. Conduction of Heat in Solids, second edition, Oxford University Press, New York, New York, pp. 30-32.

Carter, R. D. and G. W. Tracy, 1960. "An Improved Method for Calculating Water Influx," Petroleum Transactions, AIME, Society of Petroleum Engineers of AIME, Vol. 219, pp. 415-417, American Institute of Mining Engineers.

Chapman, N. A., and J. A. T. Smellie, 1986. "Natural Analogues to the Conditions around a Final Repository for High-Level Radioactive Waste," Chemical Geology, Vol. 55, No. 3/4.

Clements, T. L. Jr., and D. E. Kudera, 1985, September. TRU Waste Sampling Program: Volume 1 -- Waste Characterization, EGG-WM-6503, Vol. 1, EG&G, Idaho Falls, Idaho.

Coons et al. (W. Coons, A. Bertgstrom, P. Gnirk, M. Gray, B. Knecht, R. Pusch, J. Steadman, B. Stillborg, M. Tokonami, and M. Vaajasaari), 1987. State-of-the-Art Report on Potentially Useful Materials for Sealing Nuclear Waste Repositories, Stripa Project Report 87-12, Swedish Nuclear Fuel and Waste Management Co., Stockholm, Sweden.

D'Appolonia Consulting Engineers, Inc., 1982. Data File Report: ERDA-6 and WIPP-12 Testing, prepared for Westinghouse Electric Corporation and the U.S. Department of Energy, Albuquerque, New Mexico, 7 vols.

D'Appolonia Consulting Engineers, Inc., 1983. Data File Report: ERDA-6 and WIPP-12 Testing, Addendum 2, Vol. VI, Project No. NM 78-648-600B, prepared for Westinghouse Electric Corporation and the U.S. Department of Energy, Albuquerque, New Mexico.

De Marsily, G., 1986. Quantitative Hydrogeology, Academic Press Inc., New York, New York, p. 440.

DOE (U.S. Department of Energy), 1988a. Integrated Data Base for 1988: Spent Fuel and Radioactive Waste Inventories, Projections, and Characteristics. DOE/RW-0006, Rev. 4.

DOE (U.S. Department of Energy), 1988b. Internal Dose Conversion Factors for Calculation of Dose to the Public, DOE/EH-0071, Washington, D.C.

Domenico, P. A., 1972. Concepts and Models in Groundwater Hydrology, McGraw-Hill Book Company, New York, New York.

Dosch, R. G., 1981. Solubility and Sorption Characteristics of Uranium (VI) Associated with Rock Samples and Brines/Groundwaters from WIPP and NTS, SAND80-1595, Sandia National Laboratories, Albuquerque, New Mexico.

Dosch, R. G., and A. W. Lynch, 1978. Interaction of Radonuclides with Geomedia Associated with the Waste Isolation Pilot Plant (WIPP) in New Mexico, SAND78-0297, Sandia National Laboratories, Albuquerque, New Mexico.

Drever, J. I., 1982. The Geochemistry of Natural Waters, Prentice Hall, Englewood Cliffs, New Jersey, pp. 78-79.

Earth Technology Corporation, 1987. Final Report for Time Domain Electromagnetic (TDEM) Surveys at the WIPP Site, SAND87-7144, Sandia National Laboratories, Albuquerque, New Mexico, for the U.S. Department of Energy.

EPA (U.S. Environmental Protection Agency), 1986. Superfund Public Health Evaluation Manual, Office of Solid Waste and Emergency Response, OSWER Directive 9285.4-1, EPA 540/1-86/060, Washington, D.C.

Envirosphere Company, 1987. EPUS-Environmental Pathways Uncertainty and Screening Model User Guide, Prepared for California Energy Commission.

Freeze, R. A. and J. A. Cherry, 1979. Groundwater, Prentice-Hall, Englewood Cliffs, New Jersey.

Frick, T. C., and R. W. Taylor, eds., 1962. Petroleum Production Handbook, Vol. II, Society of Petroleum Engineers of AIME, Dallas, Texas.

Hausner, H. H., and M. Kumar Mal, 1982. Handbook of Powder Metallurgy, Chemical Publishing Co. Inc., p. 123.

Hayes et al. (K. F. Hayes, C. Papelis, and J. O. Leckie), 1988, October. "Modeling Ionic Strength Effects on Anion Adsorption at Hydrous Oxide/Solution Interfaces," Journal of Colloid and Interface Science, Vol. 125, No. 2, pp. 717-726.

Helgeson, H. C., 1969. "Thermodynamics of Hydrothermal Systems at Elevated Temperatures and Pressures," American Journal of Science, Vol. 267, pp. 729-804.

Hunter et al. (R. L. Hunter, R. M. Cranwell, and M. S. Y. Chu), 1986. Assessing Compliance with the EPA High-Level Waste Standard: An Overview, SAND86-0121, NUREG/CR-4510, Sandia National Laboratories, Albuquerque, New Mexico.

ICRP (International Commission on Radiological Protection), 1959. Report of Committee II on Permissible Dose for Internal Radiation, International Commission on Radiological Protection, ICRP Publication 2, Pergamon Press, New York, New York.

ICRP (International Commission on Radiological Protection), 1975. Report of the Task Group on Reference Man: Anatomical, Physiological, and Metabolic Characteristics, ICRP Publication 23, Pergamon Press, New York, New York.

ICRP (International Commission on Radiological Protection), 1977. Recommendations of the International Commission on Radiological Protection, ICRP Publication 26, Pergamon Press, New York, New York.

ICRP (International Commission on Radiological Protection), 1979. Annals of the ICRP, ICRP Publication 30, "Limits for Intake of Radionuclides by Workers," Pergamon Press, New York, New York.

ICRP (International Commission on Radiological Protection), 1983. "Radionuclide

Transformations: Energy and Intensity of Emissions," Annals of the ICRP, ICRP Publication 38, Pergamon Press, New York, New York.

Jackson, K. J., 1988, Verification and Validation Studies of the Addition of Pitzers' Equations to the EQ3/6 Brine Model, UCRL-53841, Lawrence Livermore National Laboratory, Livermore, California.

Katsube et al. (T. J. Katsube, T. W. Melnyk, and J. P. Hume), 1986. Pore Structure from Diffusion in Granitic Rocks, Atomic Energy of Canada Ltd., Technical Record 381, p.28.

Kelley, V. A, and J. F. Pickens, 1986. Interpretation of the Convergent-Flow Tracer Tests Conducted in the Culebra Dolomite at the H-3 and H-4 Hydropads at the Waste Isolation Pilot Plant (WIPP) Site, SAND86-7161, Sandia National Laboratories, Albuquerque, New Mexico.

Kent et al. (D. B. Kent, V. S. Tripathi, N. B. Ball, J. O. Leckie, and M. D. Siegel), 1988. Surface-Complexation Modeling of Radionuclide Adsorption in Subsurface Environments, NUREG/CR-4807, SAND86-7175, Sandia National Laboratories, Albuquerque, New Mexico.

Krauskopf, K. B., 1986. "Thorium and Rare-Earth Metals as Analogs for Actinide Elements," Chemical Geology, Vol. 55, pp. 323-335.

Lallemand-Barres, A., and P. Peaudecerf, 1978. "Recherche des Relations Entre la Valeur de la Dispersivité Macroscropique d'un Milieu Aquifère, ses Autres Caractéristiques et les Conditions de Mesure," Bulletin du Bureau de Recherches Geologiques et Minieres, Ser. 2, Sec.III, 4-1978, pp. 277-284.

Langmuir, D., and A. C. Riese, 1985. "The Thermodynamic Properties of Radium," Geochimica et Cosmochimica Acta, Vol. 49, pp. 1593-1601.

Langmuir, D., and J. S. Herman, 1980. "The Mobility of Thorium in Natural Waters at Low Temperatures," Geochimica et Cosmochimica Acta, Vol. 44, pp. 1753-1766.

Lappin, A. R., 1988. Summary of the Site-Selection Studies Conducted from 1983 Through 1987 at the Waste Isolation Pilot Plant (WIPP) Site, Southeastern New Mexico, SAND88-0157, Sandia National Laboratories, Albuquerque, New Mexico,

Lappin et al. (A. R. Lappin, R. L. Hunter, D. P. Garber, and P. B. Davies, eds.), 1989. Systems Analysis, Long-Term Radionuclide Transport, and Dose Assessment, Waste Isolation Pilot Plant (WIPP), Southeastern New Mexico; March 1989, SAND89-0462, Sandia National Laboratories, Albuquerque, New Mexico.

Lappin et al. (A. R. Lappin, R. L. Hunter, D. P. Garber, and P. B. Davies, eds.), 1990. Systems Analysis, Long-Term Radionuclide Transport, and Dose Assessment, Waste Isolation Pilot Plant (WIPP), Southeastern New Mexico; September 1989, SAND89-1996, Sandia National Laboratories, Albuquerque, New Mexico (to be published).

LaVenue et al. (A. M. LaVenue, A. Haug, and V. A. Kelley), 1988. Numerical Simulation of Ground-Water Flow in the Culebra Dolomite at the Waste Isolation Pilot Plant (WIPP) Site: Second Interim Report, SAND88-7002, Sandia National Laboratories.

Lee, W. J., 1982. Well Testing, Vol. I, SPE Textbook Series, Society of Petroleum Engineers of AIME, Dallas, Texas.

Li, Y., and S. Gregory, 1974. "Diffusion of Ions in Seawater and in Deep-Sea Sediments," Geochimica et Cosmochimica Acta, Vol. 38, pp. 703-714.

Longsine et al. (D. E. Longsine, J. Bonano, and C. P. Harlan), 1987. User's Manual for the NEFTRAN Computer Code, NUREG/CR-4766, SAND86-2405, Sandia National Laboratories, Albuquerque, New Mexico.

Lynch, A. W., and R. G. Dosch, 1980. Sorption Coefficients for Radionuclides on Samples from the Water-Bearing Magenta and Culebra Members of the Rustler Formation, SAND80-1064, Sandia National Laboratories, Albuquerque, New Mexico.

Martin, W. E., and S. G. Bloom, 1980. "Nevada Applied Ecology Group Model for Estimating Plutonium Transport and Dose to Man," in W. C. Hanson, ed., Transuranic Elements in the Environment, DOE/TIC-22800, U.S. Department of Energy, Oak Ridge, Tennessee.

Melfi, J. R., 1985. A Geochemical Model for Radionuclide Speciation in High Ionic Strength Solutions, Master's Thesis, University of New Mexico, Albuquerque, New Mexico.

Merck & Co., Inc., 1979. The Merck Veterinary Manual, ed. O. H. Siegmund, Rahway, New Jersey.

Munson et al. (D. E. Munson, A. F. Fossum, and P. E. Senseny), 1989. "Approach to First Principles Model Prediction of Measured WIPP In Situ Room Closure in Salt," Proceedings of the 30th U.S. Rock Mechanics Symposium, University of West Virginia, June 19-22, 1988, A. A. Balkema, Rotterdam, The Netherlands, to be published.

Narasimhan, T. N. and B. Y. Kanehiro, 1980. "A Note on the Meaning of Storage Coefficient," Water Resources Research, Vol.16, No.2, pp. 423-429.

NCRP (National Council on Radiation Protection and Measurements), 1984. Radiological Assessment: Predicting the Transport, Bioaccumulation, and Uptake by man of Radionuclides Released to the Environment, NCRP Report 76, Bethesda, Maryland.

Neill, R. H., 1989. Review of the Draft Supplement Environmental Impact Statement, DOE Waste Isolation Pilot Plant, April 1989, EEG-41, Environmental Evaluation Group, Albuquerque, New Mexico.

Neretnieks, I., and A. Rasmuson, 1984. "An Approach to Modeling Radionuclide Migration in a Medium with Strongly Varying Velocity and Block Sizes Along the Flow Path," Water Resources Research, Vol. 20, No. 12, pp. 1823-1836.

Nowak, E. J., 1980. Radonuclide Sorption and Migration Studies of Getters for Backfill Barriers, SAND79-1110, Sandia National Laboratories, Albuquerque, New Mexico.

Nowak et al. (E. J. Nowak, D. F. McTigue, and R. Beraun), 1988. Brine Inflow to WIPP Disposal Rooms: Data, Modeling, and Assessment, SAND88-0112, Sandia National Laboratories, Albuquerque, New Mexico.

NRC (U.S. Nuclear Regulatory Commission), 1976. "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Regulatory Guide 1.10911, NRC, Washington, D.C.

Paine, R. T., 1978. Chemistry Related to the WIPP Site, Draft Final Report, Sandia Contract GTK/07-1488, Chemistry Department, University of New Mexico, Albuquerque, New Mexico.

Pickens, J. F., and G. E. Grisak, 1981. "Scale-Dependent Dispersion in a Stratified Granular Aquifer," Water Resources Research, Vol. 17, No.4, pp. 1191-1211.

Pickens et al. (J. F. Pickens, G. E. Grisak, J. D. Avis, D. W. Belanger, and M. Thury), 1987. "Analysis and Interpretation of Borehole Hydraulic Tests in Deep Boreholes: Principles, Model Development, and Applications," Water Resources Research, Vol. 23, No. 7, pp. 1341-1375.

Popielak et al. (R. S. Popielak, R. L. Beauheim, S. R. Black, W. E. Coons, C. T. Ellingson, and R. L. Olsen), 1983. Brine Reservoirs in the Castile Formation Waste Isolation Pilot Plant (WIPP) Project Southeastern New Mexico, prepared by D'Appolonia Consulting Engineers, TME 3153, Inc., for the U.S. Department of Energy.

Poston, J. W., 1985. Internal Dose Assessment, Technical Management Services, New Hartford, Connecticut.

Randall et al. (W. S. Randall, M. E. Crawley, and M. L. Lyon), 1988. 1988 Annual Water Quality Data Report for the Waste Isolation Pilot Plant. DOE-WIPP 88-006, U.S. Department of Energy.

Reeves et al. (M. Reeves, D. S. Ward, N. D. Johns, and R. M. Cranwell), 1986. Theory and Implementation for SWIFT II: The Sandia Waste-Isolation Flow and Transport Model for Fractured Media Release 4.84, NUREG/CR-3328, SAND83-1159, Sandia National Laboratories, Albuquerque, New Mexico, for the U.S. Nuclear Regulatory Commission.

Reeves et al. (M. Reeves, V. A. Kelley, and J. F. Pickens), 1987. Regional Double-Porosity Solute Transport in the Culebra Dolomite: An Analysis of Parameter Sensitivity and Importance at the Waste Isolation Pilot Plant (WIPP) Site, SAND87-7105, Sandia National Laboratories, Albuquerque, New Mexico.

Riese, A. C., 1982. Adsorption of Radium and Thorium onto Quartz and Kaolinite: A Comparison of Solution/Surface Equilibria Models, Ph.D. Thesis, Colorado School of Mines, Golden, Colorado.

Serne et al. (R. J. Serne, D. Rai, M. J. Mason, and M. A. Molecke), 1977. Batch K_d Measurements of Nuclides to Estimate Migration Potential at the Proposed Waste Isolation Pilot Plant in New Mexico, PNL-2448, Pacific Northwest Laboratories, Richland, Washington.

Siegel et al. (M. D. Siegel, S. J. Lambert, and K. L. Robinson, eds.), 1989. Hydrogeochemical Studies of the Rustler Formation and Related Rocks in the WIPP Area, Southeastern

New Mexico, SAND88-0196, Sandia National Laboratories, Albuquerque, New Mexico.

Soudek, A., 1984. Binary and Ternary Ion Exchange on Sodium Montmorillonite for Application to Backfill in Nuclear Water Disposal," Master's Thesis, Chemical Engineering, University of California, Berkeley, California.

Swanson, J. L., 1986. Organic Complexant-Enhanced Mobility of Toxic Elements in Low-Level Wastes: Final Report, NUREG/CR-4660, PNL-4965-10, Pacific Northwest Laboratory, Richland, Washington.

Tien et al. (P. L. Tien, F. B. Nimick, A. B. Muller, P. A. Davis, R. V. Guzowski, L. E. Duda, and R. L. Hunter), 1983. Repository Site Data and Information in Bedded Salt: Palo Duro Basin, Texas, NUREG/CR-3129, SAND82-2223, Sandia National Laboratories, Albuquerque, New Mexico.

Till, J. E. and H. R. Meyer, 1983. Radiological Assessment: A Textbook on Environmental Dose Analysis, NUREG/CR-3332, ORNL-5968, U.S. Nuclear Regulatory Commission, Washington, D.C.

Torstenfelt et al. (B. Torstenfelt, H. Kipatsi, K. Anderson, B. Allard, and University Olofsson), 1982. "Transport of Actinides Through a Bentonite Backfill," in W. Lutze, ed., Scientific Basis for Nuclear Waste Management V, Elsevier Science Publishing Company, New York, pp. 659-668.

Tsunashima et al. (A. Tsunashima, G. W. Brindley, and M. Bastovanov), 1981. "Adsorption of Uranium from Solutions by Montmorillonite; Compositions and Properties of Uranyl Monmorillonites," Clays and Clay Minerals, Vol. 29, No. 1.

USPHS (U.S. Public Health Service), 1970. Radiological Health Handbook, U.S. Department of Health, Education, and Welfare, Rockville, Maryland.

Van Everdingen, A. F. and W. Hurst, 1949. "Application of the Laplace Transformation to Flow Problems in Reservoirs," Transactions of the Society of Petroleum Engineers, American Institute of Mining Engineers, Dallas, Texas, Vol. 186, pp. 305-324.

WEC (Westinghouse Electric Corporation), 1989. Radioactive Mixed Waste Compliance Manual, Appendix 6.4.1, WP-02-07, Rev. 0, Waste Isolation Pilot Plant, Carlsbad, New Mexico.

Whelan et al. (G. Whelan, D. L. Strenge, J. G. Droppo, Jr., B. L. Steelman, and J. W. Burk), 1987. The Remedial Action Priority System (RAPS): Mathematical Formulations, DOE/RL/87-09, PNL-6200, Pacific Northwest Laboratory, Richland, Washington.

Wolery, T. J., 1983. EQ3NR–A Computer Program for Geochemical Aqueous Speciation-Solubility Calculations: User's Guide and Documentation, UCRL-53414, Lawrence Livermore National Laboratory, Livermore, California.

APPENDIX J

BIBLIOGRAPHY

TABLE OF CONTENTS

NOTE TO THE READER:

This appendix contains the most current bibliography related to the WIPP. The list contains all writings about the WIPP, not merely those referenced in this document. The citations are organized into seven subject areas:

1. Design Development
2. Environmental
3. Geochemistry
4. Geology
5. Hydrology
6. Repository
7. Resources

Appendix J has not been reprinted in this final SEIS. The reader is referred to the draft SEIS for the complete Appendix.

J.2 **DESIGN DEVELOPMENT**

Bechtel National, Inc., 1979. Geology Laboratory Results of Rock Testing, Document No. 22-4-510-04, Rev. 2, San Francisco, California.

Bechtel National, Inc., 1979 & 1980. WIPP Title I Design Report and supplements.

Bechtel National, Inc., 1980. Waste Isolation Pilot Plant, Design Basis, Exploratory Shaft, D-31-R-03, Rev. 2.

Bechtel National, Inc., 1980. Waste Isolation Pilot Plant, Design Basis, Exploratory Shaft Furnishings and Station, D-37-S-02, Rev. 2.

Bechtel National, Inc., 1980. Waste Isolation Pilot Plant, Design Basis, Exploratory Shaft Geomechanical Instrumentation, D-32-X-01, Rev. 2.

Bechtel National, Inc., 1980. Waste Isolation Pilot Plant, Design Basis, SPDV Underground Excavations, D-51-W-02.

Bechtel National, Inc., 1981. Stability Analysis of Underground Openings, RMC-II, Study Report No. 51-R-510-04, Rev. 0.

Bechtel National, Inc., 1982. Waste Isolation Pilot Plant, Design Basis, Underground Experimental Areas, D-52-K-04, Rev 2.

Bechtel National, Inc., 1983. Primary Creep of Rock Salt, 76-D-540AD-01, San Francisco, California.

Bechtel National, Inc., 1983. Waste Isolation Pilot Plant, Design Basis, Exhaust Shaft, D-35-R-01, Rev. 1.

Bechtel National, Inc., 1983. Waste Isolation Pilot Plant, Design Basis, Design Classification, D-76-K-03, Rev. 9.

Bechtel National, Inc., 1983. Waste Isolation Pilot Plant, Design Basis, Geomechanical Instrumentation, D-76-X-01, Rev. 6.

Bechtel National, Inc., 1983. Waste Isolation Pilot Plant, Design Basis, Underground Excavations, D-51-W-01, Rev. 6.

Bechtel National, Inc., 1983. Waste Isolation Pilot Plant, Design Basis, Waste Shaft, D-31-R-01, Rev. 8.

Bechtel National, Inc., 1983. Waste Isolation Pilot Plant, Design Basis, WIPP-RMC IIA (General), D-76-K-01, Rev 7.

Bechtel National, Inc., 1983. Waste Isolation Pilot Plant Preliminary Design Validation Report, San Francisco, California.

Bechtel National Inc., 1984. Waste Isolation Pilot Plant, Design Basis, Underground Shops and Facilities, D-54-E-01, Rev. 6.

Branstetter, L. J., 1985. Second Reference Calculations for the WIPP, SAND83-2461, Sandia National Laboratories, Albuquerque, New Mexico.

Branstetter, L. J., R. D. Krieg, and C. M. Stone, 1982. The Effect of a Random Variation of Rock Salt Creep on Calculations of Storage Room Closure for the WIPP Project, SAND82-0024, Sandia National Laboratories, Albuquerque, New Mexico.

DOE (U.S. Department of Energy), 1980. Final Environmental Impact Statement: Waste Isolation Pilot Plant, DOE/EIS-0026, Washington, D.C.

DOE (U.S. Department of Energy), 1980. Waste Isolation Pilot Plant, Safety Analysis Report, Carlsbad, New Mexico.

DOE (U.S. Department of Energy), 1983. Waste Isolation Pilot Plant, Design Validation Plan, Document No. 76B510AB-000, Rev. 6, Carlsbad, New Mexico.

DOE (U.S. Department of Energy), 1986. Waste Isolation Pilot Plant, Design Validation Strategy, Carlsbad, New Mexico.

DOE (U.S. Department of Energy), 1988. "General Design Criteria Manual," DOE Order 6430.1A, U.S. Department of Energy.

Dunning, D. E., Jr., et al., "A Combined Methodology for Estimating Dose Rates and Health Effects from Exposure to Radioactive Effluents," ORNL/TM-7105, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Fischer, N. T., ed., 1985. "Ecological Monitoring at the Waste Isolation Pilot Plant, Second-Semi-Annual Report," DOE-WIPP-85-002, Carlsbad, New Mexico.

Fischer, N. T., ed., 1987. "Ecological Monitoring Program at the Waste Isolation Pilot Plant, Annual Report for FY 1986," DOE-WIPP-87-003, Carlsbad, New Mexico.

Fischer, N. T., ed., 1987. "Ecological Monitoring Program, Annual Report, CY 1986," DOE/WIPP 87-003, Waste Isolation Pilot Plant, Carlsbad, New Mexico.

Fischer, N. T., ed., 1988. "Ecological Monitoring Program, Annual Report, CY 1987," DOE/WIPP 88-008, Waste Isolation Pilot Plant, Carlsbad, New Mexico.

Fischer, N. T., J. S. Cockman, S. B. Jones, E. T. Louderbough, and A. L. Rodriguez, 1987. Ecological Monitoring Program at the Waste Isolation Pilot Plant, Annual Report for FY 1986, DOE/WIPP-87-003, Carlsbad, New Mexico.

Flynn, D. T., N. T. Fischer, J. P. Harvill, W. S. Randall, and A. L. Rodriguez, 1988. Annual Site Environmental Monitoring Report for the Waste Isolation Pilot Plant, CY 1987, DOE/WIPP-88-009, Carlsbad, New Mexico.

Flynn, D. T., J. S. Cockman, N. T. Fischer, J. P. Harvill, E. T. Londerbough, M. L. Lyon, A. L. Rodriquez, B. A. Sladek, 1989. Annual Site Environmental Report for the Waste Isolation Pilot Plant, CY 1988, DOE/WIPP-89-005, Carlsbad, New Mexico.

Holt, R., and D. Powers. Geotechnical Activities in the Exhaust Shaft, DOE-WIPP-86-008, Westinghouse Electric Corp., Carlsbad, New Mexico.

Matalucci, R. V., C. L. Christensen, T. O. Hunter, M. A. Molecke, and D. E. Munson, 1982. WIPP Research and Development Program: In Situ Testing Plan, SAND81-2628, Sandia National Laboratories, Albuquerque, New Mexico.

McKinney, R. F., and R. S. Newton, 1983. Site Validation Field Program Plan, Rev. 1.2, Westinghouse Electric Corp., Carlsbad, New Mexico.

Miller, J. D., C. M. Stone, L. J. Branstetter, 1982. Reference Calculations for Underground Rooms of the WIPP, SAND82-1176, Sandia National Laboratories, Albuquerque, New Mexico.

Morgan, H. S., R. D. Grieg, and R. V. Matalucci, 1981. Comparative Analysis of None Structural Codes Used in the Second WIPP Benchmark Problem, SAND81-1389, Sandia National Laboratories, Albuquerque, New Mexico.

National Research Council on the Waste Isolation Pilot Plant (WIPP), 1984. Review of the Scientific and Technical Criteria for the Waste Isolation Pilot Plant (WIPP), DOE/DP/48015-1, National Academy Press, Washington, D.C.

Powers, D. W., S. J. Lambert, S. E. Shaffer, L. R. Hill, and W. D. Weart, eds., 1978. Geological Characterization Report, Waste Isolation Pilot Plant (WIPP) Site, Southeastern New Mexico, SAND78-1596, Sandia National Laboratories, Albuquerque, New Mexico.

Sandia National Laboratories, 1977. WIPP Conceptual Design Report, SAND77-0274, Albuquerque, New Mexico.

Serata, S., 1979. Empirical Design Basis of Underground Salt Openings in the WIPP Project, DR-51-R-1.

Spiegler, 1983. Evaluation of the Suitability of the WIPP Site, EEG-23, New Mexico Environmental Evaluation Group, Santa Fe, New Mexico.

Stormont, J.C., 1984. Plugging and Sealing Program for the Waste Isolation Pilot Plant (WIPP), SAND84-1057, Sandia National Laboratories, Albuquerque, New Mexico.

Weart, W. D., 1983. Summary Evaluation of the Waste Isolation Pilot Plant (WIPP Site Suitability, SAND83-0450, Sandia National Laboratories, Albuquerque, New Mexico.

WEC (Westinghouse Electric Corporation), 1981. Design Criteria, Waste Isolation Pilot Plant (WIPP), Site and Preliminary Design Validation (SPDV), WIPP-DOE-72, Rev. 1, Carlsbad, New Mexico.

WEC (Westinghouse Electric Corporation), 1982. Site Validation Program, WIPP-DOE-116, Rev. 1, Carlsbad, New Mexico.

WEC (Westinghouse Electric Corporation), 1983. Results of Site Validation Experiments, Waste Isolation Pilot Plant (WIPP) Project, Southeastern New Mexico, Vols. 1-2, TME 3177.

WEC (Westinghouse Electric Corporation), 1984. Design Criteria, Waste Isolation Pilot Plant (WIPP), Revised Mission Concept-IIA (RMC-IIa), WIPP-DOE-71, Rev. 4.

WEC (Westinghouse Electric Corporation), 1984. Geotechnical Activities in the Waste Handling Shaft, WTSD-TME-038, Carlsbad, New Mexico.

WEC (Westinghouse Electric Corporation), 1988. WIPP Operational Safety Requirements Administration Plan, WP-04-7, Rev. 4, Waste Isolation Pilot Plant, Carlsbad, New Mexico.

WEC (Westinghouse Electric Corporation), 1989. Radioactive Mixed Waste Compliance Manual, Appendix 6.4.1, WP-02-07, Rev. 0, Waste Isolation Pilot Plant, Carlsbad, New Mexico.

WEC (Westinghouse Electric Corporation), 1989. WIPP RCRA Pare A Permit Application.

J.3 **ENVIRONMENTAL**

ACGIH (American Conference of Governmental Industrial Hygienists), 1984. Documentation for the Threshold Limit Values, fourth edition (1980) and updates (1981, 1982, 1983, 1984), Cincinnati, Ohio.

ACGIH (American Conference of Governmental Industrial Hygienists), 1986. Documentation of the Threshold Limit Values and Biological Exposure Indices, fifth edition, Cincinnati, Ohio.

AEC (Atomic Energy Commission), 1972. Environmental Survey of Transportation of Radioactive Materials to and from Nuclear Power Plants, WASH-1238, Directorate of Regulatory Standards, Washington, D.C.

Alexander, H. C., W. M. McCarty, and E. A. Bartlett, 1978. "Toxicity of Perchlorethylene, Trichloroethylene, 1,1,1-Trichloroethane, and Methylene Chloride to Fathead Minnows," Bulletin of Environmental Contamination & Toxicology, Vol. 20, pp. 344-352.

American Cancer Society, 1988. Cancer Facts and Figures, American Cancer Society, New York, New York.

ATSDR (Agency for Toxic Substances and Disease Registry), 1987. Toxicological Profile for Methylene Chloride, draft, prepared by Life Systems, Inc.

ATSDR (Agency for Toxic Substances and Disease Registry), 1988. Toxicological Profile for Lead, draft, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

ATSDR (Agency for Toxic Substances and Disease Registry), 1988. Toxicological Profile for Trichloroethylene, draft, prepared by Technical Resources, Inc. under contract, No. 68-03-23618.

Aviado, D. M., 1975. Toxicology, Vol. 3, p. 321.

Baes, C. F., and T. H. Orton, 1979. "Productivity of Agricultural Crops and Forage," in A Statistical Analysis of Selected Parameters for Predicting Food Chain Transportation and Internal Dose of Radionuclides, eds. F. O. Hoffman and C. F. Baes, ORNL/NUREG/TM-282, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Baes, C. F., R. D. Sharp, A. L. Sjoreen, and R. W. Shor, 1984. A Review and Analysis of Parameters for Assessing Transport of Environmentally Released Radionuclides Through Agriculture, ORNL-5786, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Banz, I., P. Bradshaw, J. S. Cockman, N. T. Fischer, J. K. Prince, A. Rodriguez, and D. W. Uhland, 1987. Annual Site Environmental Monitoring Report for the Waste Isolation Pilot Plant, CY 1986, DOE/WIPP-87-002, Waste Isolation Pilot Plant, Carlsbad, New

Mexico.

Barnes, H., 1974. "Geologic and Hydrologic Background for Selecting Site of Pilot-Plant Repository for Radioactive Waste," Bulletin of the Association of Engineering Geologists, Vol. 11, No. 1, pp. 83-92.

Barrio-Lage, G., F. S. Parsons, and R. S. Nassar, 1987. "Kinetics of the Depletion of Trichloroethylene," Environmental Science and Technology, Vol. 21, pp. 366-370.

Bednarz, J. C., 1987. The Los Medanos Cooperative Raptor Research and Management Program, 1986 Annual Report, Contract No. 59-WRK-90469-5D, U.S. Department of Energy, Carlsbad, New Mexico.

Bednarz, J. C., and T. J. Hayden, 1988. The Los Medanos Cooperative Raptor Research and Management Program Final Report, 1985-1987, Contract No. 59-WRK-90469-SD, U.S. Department of Energy, Carlsbad, New Mexico.

BEIR (National Academy of Sciences, Committee on the Biological Effects of Ionizing Radiation), 1980. "The Effects on Populations of Exposure to Low Levels of Ionizing Radiation," Washington, D.C.

BEIR (National Academy of Sciences, Committee on the Biological Effects of Ionizing Radiation), 1988. "Health Risks of Radon and Other Internally Deposited Alpha Emitters."

Bell, Z. G., K. H. Oson, and T. J. Benya, 1978. "Final Report of Audit Findings of the Manufacturing Chemists Association," administered trichloroethylene chronic inhalation study at Industrial Biotest Laboratories, Inc., Decatur, Illinois, unpublished.

Bertram-Howery, S. G., and R. L. Hunter, 1989. Plans for Evaluation of the Waste Isolation Pilot Plant's Compliance with EPA Standards for Radioactive Waste Management and Disposal, SAND88-2871, Sandia National Laboratories, Albuquerque, New Mexico, prepared for the U.S. Department of Energy.

Best, T. L., and K. S. Neuhauser, eds., 1979. Report of Biological Investigations at the Los Medanos Waste Isolation Pilot Plant (WIPP) Area of New Mexico During FY 1978, SAND79-0368, Sandia National Laboratories, Albuquerque, New Mexico.

Bingham, F. W., and G. E. Barr, 1979. Scenarios for Long-Term Release of Radionuclides from a Nuclear-Waste Repository in the Los Medanos Region of New Mexico, SAND78-1730, Sandia National Laboratories, Albuquerque, New Mexico.

Black, J. A., W. J. Birge, W. E. McDonnell, A. G. Westerman, B. A. Ramey and D. M. Bruser, 1982. "The Aquatic Toxicity of Organic Compounds to Embryo-Larval Stages of Fist and Amphibians," Res. Report No. 133, Water Resource Institute, University of Kentucky.

Blaney, H. F., and E. G. Hanson, 1965. Consumptive Use and Water Requirements in New Mexico, Technical Report 32, New Mexico State Engineer's Office.

Botts, R. P., 1977. The Short Term Effects of Lead on Domestic and Wild Animals, EPA-600/3-77-009, Corvallis Environmental Research Laboratory, Corvallis, Oregon.

Bradshaw, P. L., and E. T. Louderbough, 1987. Compilation of Historical Radiological Data Collected in the Vicinity of the WIPP Site, DOE/WIPP-87-004, Waste Isolation Pilot Plant, Carlsbad, New Mexico.

Braithwaite, J. W., and M. A. Molecke, 1980. "Nuclear Waste Canister Corrosion Studies Pertinent to Geologic Isolation," Nuclear and Chemical Waste, Vol. 1, (preprint: SAND79-1935J), Sandia National Laboratories, Albuquerque, New Mexico.

Brewer, L. W., and J. H. Metcalf, 1977. Environmental Monitoring at the WIPP Site, CY 1976, SAND77-1201, Sandia National Laboratories, Albuquerque, New Mexico.

Briggs, G. A., 1969. "Plume Rise," AEC Critical Review Series, TID-25075.

Burek, J. D., K. D. Nitschke, T. Bell, D. L. Wackerle, A. C. Childs, J. Beyer, D. A. Dittenber, L. W. Rampy, and M. J. McKenna, 1980. "Methylene Chloride: A Two Year Inhalation Toxicity and Oncogenicity Study in Rats and Hamsters," Toxicology Research Laboratory, Health and Environmental Sciences, Dow Chemical Company, Midland, Michigan.

Burek, J. D., K. D. Nitschke, T. Bell, D. L. Wackerle, A. C. Childs, J. Beyer, D. A. Dittenbar, L. W. Rampy, and M. J. McKenna, 1984. "Methylene Chloride: A Two Year Inhalation Toxicity and Oncogenicity Study in Rats and Hamsters", Fund. Appl., Toxicology, Vol. 4, No. 1, pp. 30-47.

CAPCOA (California Air Pollution Control Officer Association), 1987. Air Toxics Assessment Manual, August.

Carlson, G. P., 1975. Toxicol., Vol. 5, p. 69.

Channell, J. K., 1982. Calculated Radiation Doses from Radionuclides Brought to The Surface if Future Drilling Intercepts The WIPP Repository and Pressurized Brine, EEG No. 11.

Cherry, N., H. Venable, and H. Woldron, 1983. "The Acute Behavioral Effects of Solvent Exposure," J. Soc. Occup. Med., Vol. 33, pp. 13-18.

Church, H. W., 1978. Excerpts from WIPP Meterological, Air Quality, and Radiological Measurements Program Plan, SAND78-1144, Sandia National Laboratories, Albuquerque, New Mexico.

Clayton, G. D., and F. E. Clayton, eds., 1981. Patty's Industrial Hygiene and Toxicology, third edition, John Wiley and Sons, New York.

Cornish, H. H., 1973. Am. Ind. Hyg. Assoc. J., Vol. 34, p. 487.

Council on Environmental Quality, 1975. Fluorocarbons and the Environment.

Davies, 1966. Aerosol Science, Academic Press, New York.

DiFerrante, E., 1979. Trace Metals: Exposure and Health Effects, Pergamon Press, New York.

Dilling, W. L., N. B. Tefertiller and G. J. Kallos, 1975. "Evaporation Rates of Methylene Chloride, Chloroform, 1,1,1-Trichloroethane, Trichloroethylene, Tetrachloroethylene, and Other Chlorinated Compounds in Dilute Aqueous Solutions," Environ. Sci. Technol., Vol. 9(a), pp. 833-838.

DOE (U.S. Department of Energy), 1979. Draft Environmental Impact Statement: Management of Commercially Generated Radioactive Waste, DOE/EIS-0046-D, Vols. 1-2, Washington, D.C.

DOE (U. S. Department of Energy), 1979. Environmental and Other Evaluations of Alternatives for Long-Term Management of Buried INEL Transuranic Waste, IDO-10085.

DOE (U.S. Department of Energy), 1979. Environmental Aspects of Commercial Radioactive Waste Management, Vols. 1-3, DOE/ET-0029, Office of Nuclear Waste Management.

DOE (U.S. Department of Energy), 1980. Final Environmental Impact Statement, Waste Isolation Pilot Plant, Vols. 1-2, DOE/EIS-0026, U.S. Department of Energy, Washington, D.C.

DOE (U.S. Department of Energy), 1980. Safety Analysis Report, Waste Isolation Pilot Plant, Albuquerque, New Mexico.

DOE (U.S. Department of Energy) and State of New Mexico, 1981. Agreement for Consultation and Cooperation Between [the U.S.] Department of Energy and the State of New Mexico on the Waste Isolation Pilot Plant, as amended 1985.

DOE (U.S. Department of Energy), 1981. Waste Isolation Pilot Plant (WIPP): Record of Decision, Vol. 46, Federal Register, No. 18, p. 9162, January 28, 46 FR 9162.

DOE (U.S. Department of Energy), 1984. A Guide to Reducing Radiation Exposure to As Low As Reasonably Achievable (ALARA), DOE/EV/1830-T5.

DOE (U.S. Department of Energy), 1986. Safety Analysis for the Radioactive Waste Management Complex at the Idaho National Engineering Laboratory, WM-PD-86-011.

DOE (U.S. Department of Energy), 1986. Safety Analysis Report for the Waste Isolation Pilot Plant, Amendment 9, Carlsbad, New Mexico.

DOE (U.S. Department of Energy), 1987. Environmental Protection, Safety, and Health Protection Information Reporting Requirements, DOE Order 5484.1, Change 3, Washington, D.C.

DOE (U. S. Department of Energy), 1987. "Final Environmental Impact Statement, Disposal of Hanford Defense High-Level, Transuranic and Tank Wastes, Hanford Site, Richland, Washington, Volumes 1-5, DOE/EIS-0113.

DOE (U.S. Department of Energy), 1987. 1986 Environmental Monitoring Program Report for the Idaho National Engineering Laboratory Site, DOE/ID-12082 (86), Radiological and Environmental Sciences Laboratory, Idaho Falls, Idaho.

DOE (U.S. Department of Energy), 1988. Annual Site Environmental Monitoring Report for the Waste Isolation Pilot Plant, CY 1987, DOE/WIPP-88-009, Carlsbad, New Mexico.

DOE (U.S. Department of Energy), 1988. Environmental Assessment, Management Activities for Retrieved and Newly-generated Transuranic Waste, Savannah River Plant, DOE/EA-0315.

DOE (U.S. Department of Energy), 1988. Final Environmental Impact Statement: Special Isotope Separation Project - Idaho National Engineering Laboratory, DOE/EIS-0136, Vol. 1, Idaho Falls, Idaho.

DOE (U.S. Department of Energy), 1988. Internal Dose Conversion Factors for Calculation of Dose to the Public, DOE/EH-0071, Washington, D.C.

DOE (U.S. Department of Energy), 1988. "Radiation Protection for Occupational Workers," DOE Order 5480.11, Washington, D.C.

DOE (U.S. Department of Energy), 1988. "Radiation Protection of the Public and the Environment," draft, DOE Order 5400.3, Washington, D.C.

DOE (U.S. Department of Energy), 1988. Radiological Effluent Monitoring and Environmental Surveillance for U.S. DOE Operations, draft, DOE Order 5400.XY.

DOE (U.S. Department of Energy), 1989. Environmental Assessment for Radioactive and Hazardous Waste Treatment at the Process Experimental Pilot Plant, draft, Idaho National Engineering Laboratory, Idaho Falls, Idaho.

Doull, J., C. D. Klaasen and M. O. Amour, 1979. Casarett and Doull's Toxicology: The Basic Science of Poisons, second edition, MacMillan Publishing Co., Inc., New York.

Dowhower, S. L., 1981. "Effects of Trampling and Salt Additions on Vegetation of Sandy Soils of the Los Medanos Area, New Mexico," unpublished thesis, New Mexico State University, Las Cruces, New Mexico.

Dunning, D. E., 1986. Estimates of Internal Dose Equivalent From Inhalation and Ingestion of Selected Radionuclides, WIPP-DOE-176, Rev. 1, Westinghouse Electric Corporation - Advanced Energy Systems Division, Albuquerque, New Mexico.

EEG (Environmental Evaluation Group), 1979. Radiological Health Review of the Draft Environmental Impact Statement (DOE/EIS-0026-D) Waste Isolation Pilot Plant, U.S. Department of Energy, EEG-3, eds. R.H. Neill, J. K. Channell, C. Wofsy, and M.A. Greenfield, Health and Environment Department, State of New Mexico, Santa Fe, New Mexico.

EEG (Environmental Evaluation Group), 1980. Calculated Radiation Doses From Deposition of Material Released in Hypothetical Transportation Accidents Involving WIPP-Related Radioactive Wastes, J. K. Channell, Health and Environment Department, State of

New Mexico, Santa Fe, New Mexico.

EEG (Environmental Evaluation Group), 1981. An Approach to Calculating Upper Bounds on Maximum Individual Doses From the Use of Contaminated Well Water Following a WIPP Repository Breach, EEG-9, P. Spiegler, Health and Environment Department, State of New Mexico, Santa Fe, New Mexico.

EEG (Environmental Evaluation Group), 1981. Radiological Health Review of the Final Environmental Impact Statement (DOE/EIS-0026) Waste Isolation Pilot Plant, U.S. Department of Energy, EEG-10, Health and Environment Department, State of New Mexico, Santa Fe, New Mexico.

EEG (Environmental Evaluation Group), 1982. Calculated Radiation Doses from Radionuclides Brought to the Surface If Future Drilling Intercepts the WIPP Repository and Pressurized Brine, EEG-11, J. K. Channell, Health and Environment Department, State of New Mexico, Santa Fe, New Mexico.

EEG (Environmental Evaluation Group), 1982. Estimated Radiation Doses Resulting if an Exploratory Borehole Penetrates a Pressurized Brine Reservoir Assumed to Exist Below the WIPP Repository Horizon - A Single Hole Scenario, EEG-15, S. T. Bard, Health and Environment Department, State of New Mexico, Santa Fe, New Mexico.

EEG (Environmental Evaluation Group), 1982. Radionuclide Release, Transport and Consequence Modeling for WIPP. A Report of a Workshop Held on September 16-17, 1981, EEG-16, Health and Environment Department, State of New Mexico, Santa Fe, New Mexico.

EEG (Environmental Evaluation Group), 1982. Review Comments on Environmental Analysis Cost Reduction Proposals (WIPP-DOE-136) July 1982, EEG-19, J. K. Channell, Health and Environment Department, State of New Mexico, Santa Fe, New Mexico.

EEG (Environmental Evaluation Group), 1983. An Evaluation of the Non-radiological Environmental Problems Relating to the WIPP, EEG-20, T. E. Baca, Health and Environment Department, State of New Mexico, Santa Fe, New Mexico.

EEG (Environmental Evaluation Group), 1984. Proposed Preoperational Environmental Monitoring Program For WIPP, EEG-26, P. Spiegler, Health and Environment Department, State of New Mexico, Santa Fe, New Mexico.

Emlen, J. T., 1971. "Population Densities of Birds Derived from Transect Counts," Auk Vol. 88, pp. 323-342.

Envirosphere Company, 1987. EPUS-Environmental Pathways Uncertainty and Screening Model User Guide, Prepared for California Energy Commission.

EPA (U.S. Environmental Protection Agency), 1975. Manual of Protective Action Guides and Protective Action is for Nuclear Incidents, EPA-520/1-75-001.

EPA (U.S. Environmental Protection Agency), 1979. Water-Related Environmental Fate of 129 Priority Pollutants, Vol. I-II, 440/4-79-029, Office of Planning and Water Standards, Washington, D.C.

EPA (U.S. Environmental Protection Agency), 1980. Ambient Water Quality Criteria for Carbon Tetrachloride, EPA 440/5-80-026, Office of Water Regulations and Standards, Criteria and Standards Division, Washington, D. C.

EPA (U.S. Environmental Protection Agency), 1980. A Method for Determining the Compatibility of Hazardous Wastes, EPA-600/2-80-076, Cincinnati, Ohio.

EPA (U.S. Environmental Protection Agency), 1982. Federal Register, Vol. 47, No. 250, p. 58196.

EPA (U.S. Environmental Protection Agency), 1984. Health Effects Assessment for 1,1,1-Trichloroethane, ECAO-CIN-HOO5, Environmental Criteria and Assessment Office, Cincinnati, Ohio.

EPA (U.S. Environmental Protection Agency), 1985. Criteria Document on Dichloromethane, final draft, Office of Drinking Water.

EPA (U.S. Environmental Protection Agency), 1985. "Development of Statistical Distribution or Ranges of Standard Factors Used in Exposure Assessments," Office of Research and Development, Washington, D.C.

EPA (U.S. Environmental Protection Agency), 1985. "Health Assessment Document for Dichloromethane (Methylene Chloride)," final report, Environmental Protection Agency, Research Triangle Park, North Carolina.

EPA (U.S. Environmental Protection Agency), 1986. Superfund Public Health Evaluation Manual, Office of Emergency and Remedial Response, Office of Solid Waste and Emergency Response, Washington, D.C., Directive 9285.4-1, EPA 540/1-86/060.

EPA (U.S. Environmental Protection Agency), 1987. Health Advisories for 25 Organics, Office of Drinking Water, PB87-235578.

EPA (U.S. Environmental Protection Agency), 1987. "Technical Analysis of New Methods and Data Regarding Dichloromethane Hazard Assessments," draft.

EPA (U.S. Environmental Protection Agency), 1987. "Update to Health Assessment Document and Addendum for Dichloromethane (Methylene Chloride): Pharmacokinetics, Mechanism of Action, and Epidemiology," draft report, EPA 600/8-87/1030A.

EPA (U.S. Environmental Protection Agency), 1988. "Guidance for Conducting Remedial Investigations and Feasibility Studies under CERCLA, Office of Emergency and Remedial Response and Office of Solid Waste and Emergency Response," draft.

EPA (U.S. Environmental Protection Agency), 1988. "National Emission Standards for Hazardous Air Pollutants: Proposed Rule and Notice of Public Hearing," CFR, Title 40, Part 61, July 28, 53 FR 28496.

EPA (U.S. Environmental Protection Agency), 1988. No Migration Variances to the Hazardous Waste Land Disposal Prohibitions: A Guidance Manual for Petitioners, draft, Contract No. 68-01-7310, Washington, D.C.

EPAEPA (U.S. Environmental Protection Agency), 1988. "Solid Waste Disposal Facility Criteria," 53 Federal Register, 33421-2.

EPA (U.S. Environmental Protection Agency), 1988. "Update to Health Assessment Document and Addendum for Dichloromethane (Methylene Chloride): Pharmaco-kinetics, Mechanism of Actions, and Epidemiology," draft report, EPA 600/8-87/1030A.

EPA (U.S. Environmental Protection Agency), 1988. Users Network for Applied Modeling of Air Pollution (UNAMAP) 6 Version: Industrial Source Complex Model, dated 88167.

Ewing, R. I., 1982. Environmental Gamma Ray Spectra from Selected Sites in Southeastern New Mexico, SAND82-0738, Sandia National Laboratories, Albuquerque, New Mexico.

Fischer, N. T., E. T. Louderbough, C. C. Reith, A. L. Rodriguez, and D. Uhland, 1985. Ecological Monitoring Program at the Waste Isolation Pilot Plant, Second Semi-Annual Report, DOE/WIPP 85-002, Carlsbad, New Mexico.

Fischer, N. T., ed., 1987. Ecological Monitoring Program at the Waste Isolation Pilot Plant Annual Report for FY 1986, DOE/WIPP-87-003, Carlsbad, New Mexico.

Fischer, N. T., E. T. Louderbough, A. L. Rodriguez, S. D. Prill, J. S. Cockman, and W. D. Byers, 1988. Ecological Monitoring Program at the Waste Isolation Pilot Plant Annual Report for Calendar Year 1987, DOE/WIPP-88-008, U.S. Department of Energy Report.

Flynn, D. T., 1988. Annual Site Environmental Monitoring Report for the Waste Isolation Pilot Plant Calendar Year 1987, DOE/WIPP-88-009, U.S. Department of Energy Report, Carlsbad, New Mexico.

Fujita, H., A. Koizumi, M. Yamamoto, M. Kumai, T. Gadamoto, and M. Skeda, 1984. "Inhibition of Delta-Azminoleuvliante Dehydratase in Trichloroethylene-Exposed Rats, and the Effects on Heme Regulation," Biochem. Biophysi. Acta, Volume 800, pp. 1-10.

Fukuda, K., K. Takemota, and H. Tsuruta, 1983. "Inhalation Carcinogenicity of Trichloroethylene in Mice and Rats," Ind. Health, Volume 21, pp. 243-54.

Gifford, F. A., 1961. "Use of Routine Meteorological Observations for Estimating Atmospheric Dispersion," Nuclear Safety, Vol. 2, no. 4, pp. 47-57.

Gosselin, R. E., 1984. Clinical Toxicology of Commercial Products, fifth edition, Williams and Wilson, Baltimore, Maryland.

Hansen, L. J., 1970. Engineering Evaluation of the Super Tiger Overpack Designed for the Shipment of Large Quantities of Hazardous Material, C2378, Mechanics Research, Inc., Tacoma, Washington.

Hart, J. S., 1981. Ecosystems Studies at the Los Medanos Site, Eddy County, New Mexico, Volume 1 of 2, TME-3106.

Hart, J. S., 1982. Ecosystems Studies at the Los Medanos Site, Eddy County, New Mexico, Volume 2 of 3, TME-3141.

Haun, C. C., E. H. Vernot, K. I. Darmer, and S. S. Diamond, 1972. "Continuous Animal Exposure to Low Levels of Dichloromethane," AMRL-TR-72-130, Paper No. 12, in Proceedings of the Third Annual Conference on Environmental Toxicology, Wright-Patterson Air Force Base, Ohio, Aerospace Medical Research Laboratory, pp. 199-208.

Hausner, H. H., and M. Kumar Mal, 1982. Handbook of Powder Metallurgy, Chemical Publishing Co.

Healy, T. E., T. R. Poole, and A. Hooper, 1982. "Rat Fetal Development and Maternal Exposure to Trichloroethylene at 100 ppm," Dr. J. Anaesth, Vol. 54, pp. 337-341.

Holzworth, G. C., 1972. "Mixing Heights, Wind Speeds, and Potential for Urban Air Pollution Throughout The Contiguous United States," EPA.

HSDB (Hazardous Substances Data Bank), 1987. September 1987, on line.

Hunter, R. L., R. M. Cranwell, and M. S. Y. Chu, 1986. Assessing Compliance with the EPA High-Level Waste Standard: An Overview, NUREG/CR-4510 or SAND86-0121, Sandia National Laboratories for the Nuclear Regulatory Commission.

IARC (International Agency for Research on Cancer), 1979. "Some Halogenated Hydrocarbons," IARC Monographs on the Evaluation of Carcinogenic Risk of Chemicals to Humans, World Health Organization, Lyon, France, Vol. 20, pp. 371-399.

IARC (International Agency for Research on Cancer), 1987. "Overall Evaluations of Carcinogenicity: An Updating of IARC Monographs," IARC Monographs on the Evaluation of Carcinogenesis Risks to Humans," Volumes 1-42. (Supplement 7) International Agency for Research on Cancer.

ICRP (International Commission on Radiological Protection), 1959. Report of the ICRP Committee II on Permissible Dose for Internal Radiation, International Commission on Radiological Protection, ICRP Publication 2, Pergamon Press, New York.

ICRP (International Commission on Radiological Protection), 1974. Report of the Task Group on Reference Man, ICRP Publication 23 (ICRP 74), Pergammon Press, Elmsford, New York.

ICRP (International Commission on Radiological Protection), 1975. Report of the Task Group on Reference Man: Anatomical, Physiological, and Metabolic Characteristics, ICRP Publication 23, Pergamon Press, New York.

ICRP (International Commission on Radiological Protection), 1977. Recommendations of the International Commission on Radiological Protection, ICRP Publication 26 (ICRP 77), Pergammon Press, Elmsford, New York.

ICRP (International Commission on Radiological Protection), 1979. Limits for Intakes of Radionuclides by Workers, ICRP Publication 30 (ICRP 79a) Pergammon Press, Elmsford, New York.

ICRP (International Commission on Radiological Protection), 1983. "Radionuclide Transformations: Energy and Intensity of Emissions," Annals of the ICRP, ICRP Publication 38, Pergamon Press, New York.

IRIS (Integrated Risk Information System), 1989. U.S. Environmental Protection Agency, January.

Kazantzis, G. and R. R. Bomford, 1960. Lancet, Vol. 1, p. 360.

Kent, D. B., V. S. Tripathi, N. B. Ball, and J. O. Leckie, 1988, March. Surface-Complexion Modeling of Radionuclide Adsorption in Subsurface Environments, NUREG/CR-4807, SAND86-7175, Sandia National Laboratories, Albuquerque, New Mexico.

Kimmerle, G., and A. Eban, 1973. "Metabolism, Excretion: 1. Experimental Exposures on Rats," Arch. Toxicol. 30:1215-126.

Kjellstrand, P., B. Holmquist, A. Alma, M. Kanje, S. Romare, I. Jonsson, L. Mannson, and M. Bjerkamo, 1983. "Trichloroethylene: Further Studies of the Effect on Body and Organ Weights and Plasma Butyryl Cholinesterase Activity in Mice," Acta. Pharmacl. Toxicology, Vol. 53, pp. 375-384.

Kjellstrand, P., M. Kanje, L. Mansson, M. Bjerkamo, S. Mortenson, J. Lanke, and B. Holmquist, 1983. "Trichloroethylene: Effects on Body and Organ Weights in Mice, Rats, and Gerbils," Toxicology, Vol. 21, pp. 105-115.

Kniskern, L. A. and W. E. Pittsman, 1952. Industrial Test Lab., No. 2335, Progress Report NS-041001, Philadelphia Naval Shipyard, Pennsylvania.

Lappin, A. R., R. L. Hunter, and D. Garber eds., 1989. Systems Analysis, Long-Term Radionuclide Transport, and Dose Assessments, Waste Isolation Pilot Plant (WIPP), Southeastern New Mexico, SAND89-0462, Sandia National Laboratories, Albuquerque, New Mexico.

Lappin, A. R., R. L. Hunter, D. P. Garber, and P. B. Davies, eds., 1990. Systems Analysis, Long-Term Radionuclide Transport, and Dose Assessment, Waste Isolation Pilot Plant (WIPP), Southeastern New Mexico; September 1989, SAND89-1996, Sandia National Laboratories, Albuquerque, New Mexico (to be published).

Lee, D. H. K. ed., 1972. Metallic Contaminants and Human Health, Academic Press, New York.

Lehmann, J., and D. Paech, 1972. "Effects of Some Vaporized Lipophilic Solvents on Carbon Dioxide Fixation," Experentia, Vol. 28, pp. 1415-1416.

Lewis, R. J. ed., 1979. Registry of Toxic Effects of Chemical Substances, ninth edition, U.S. Department of Health and Human Services, National Institute for Occupational Safety and Health, Cincinnati, Ohio.

Lord, K. J., and W. E. Reynolds eds., 1985. Archaeological Investigations of Three Sites within the WIPP Core Area, Eddy County, New Mexico, Chambers Consultants and Planners, Albuquerque, New Mexico, prepared for the U.S. Army Corps of Engineers, Albuquerque District, New Mexico.

Lorenz, R. S., et al., 1980. Fission Product Release From Highly Irradiated LWR Fuel, NUREG/CR-0772, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Louderbough, E. T., 1986. Operational Environmental Permit Compliance Plan for the Waste Isolation Pilot Plant, DOE/WIPP 2-4.

MacLennan, R. B., and S. C. Schermer, 1979. "An Archaeological Survey for the Waste Isolation Pilot Plant: Access Roads and Railroad Right-of-Way," Agency for Conservation Archaeology, Eastern New Mexico University, Portales, New Mexico.

Magnani, N. J., and J. W. Braithwaite, 1979. Corrosion Resistant Metallic Canisters: An Important Element of a Multibarrier Disposal Program, SAND79-1429C, Sandia National Laboratories, Albuquerque, New Mexico, presented at the Materials Research Society Symposium, November 1979, Boston, Massachussets.

Maling, H. H., et al., 1975. Tox. Appl. Pharm., Vol. 33, p. 291.

Maltoni, C., G. Lefemine, and G. Cotti, 1986. "Experimental Research on Trichloroethylene Carcinogenesis," Archir. Res. Industrial Carcinogenesis Series, eds. C. Maltoni and M. A. Mehlman, Vol. V, Princeton Scientific Publishing Co., Inc., Princeton, New Jersey.

Manual of Protective Action Guides and Protective Action is for Nuclear Incidents, 1975 (September). EPA-520/1-75-001, U.S. Environmental Protection Agency.

Martin, W. E., and S. G. Bloom, 1980. "Nevada Applied Ecology Group Model for Estimating Plutonium Transport and Dose to Man," in Transuranic Elements in the Environment, ed. W. C. Hanson, DOE/TIC-22800, U.S. Department of Energy, Oak Ridge, Tennessee.

Matejka, D. O., 1977. Meteorological Data Quarterly Report, WIPP Site, Eddy County Summer Quarter, May-June 1976, New Mexico, SAND77-1146, Sandia National Laboratories, Albuquerque, New Mexico.

Mazza, V., and A. Brancaccio, 1967. "Characteristics of the Formed Elements of the Blood and Bone Marrow in Experimental Trichloroethylene Intoxication," Folia Med., 501381-324.

Mercer, D. D., P. L. Baker, J. S. Cockman, N. T. Fischer, D. T. Flynn, J. P. Harvill, K. L. Knudtsen, E. T. Louderbough, 1988. Operational Environmental Monitoring Plan for the Waste Isolation Pilot Plant, DOE/WIPP 88-025, prepared for the U. S. Department of Energy, Carlsbad, New Mexico.

Merck, (The Merck Index), 1983. Tenth Edition, Merck and Company, Rahway, New Jersey.

Merck & Co., Inc., 1979. The Merck Veterinary Manual, ed. O. H. Siegmund, Rahway, New Jersey.

Metcalf, J. H., and L. W. Brewer, 1977. Environmental Monitoring at the WIPP Site, CY 1976, SAND77-1021, Sandia National Laboratories, Albuquerque, New Mexico.

Miller, C. W., 1979. "The Interception Fraction," in A Statistical Analysis of Selected Parameters for Predicting Food Chain Transport and Internal Dose of Radionuclides, Final Report, eds. F. O. Hoffman and C. F. Baes, ORNL/NUREG/TM-282.

Minnema, D. M., and L. W. Brewer, 1983. Background Radiation Measurements at the Waste Isolation Pilot Plant (WIPP) Site, Carlsbad, New Mexico, SAND83-1296, Sandia National Laboratories, Albuquerque, New Mexico.

Molecke, M. A., 1978. Waste Isolation Pilot Plant Transuranic Wastes Experimental Characterization Program, Executive Summary, SAND78-1356, Sandia National Laboratories, Albuquerque, New Mexico.

Molecke, M. A., 1979. Gas Generation from Transuranic Waste Degradation: Data Summary Interpretation, SAND79-1245, Sandia National Laboratories, Albuquerque, New Mexico.

Molecke, M. A., 1980. WIPP HLW Experimental Program: Preliminary Description of In Situ HLW Tests, SAND79-0510, Sandia National Laboratories, Albuquerque, New Mexico.

Moore, R. E., 1977. The AIRDOS-II Computer Code for Estimating Radiation Dose to Man From Airborne Radionuclides in Areas Surrounding Nuclear Facilities, ORNL-5245, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Moore, R. E., C. F. Baes III, L. M. McDowell-Boyer, A. P. Watson, F. O. Hoffman, J. C. Pleasant, and C. W. Miller, 1979. AIRDOS-EPA: A Computerized Methodology for Estimating Environmental Concentrations and Dose to Man for Airborne Release of Radionuclides, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Morris, J. B., F. A. Smith, and R. H. Garman, 1979. "Studies on Methylene Chloride-Induced Fatty Liver," Exp. Molec. Path., Vol. 30, pp. 386-393.

NAS (National Academy of Sciences, National Research Council), 1980. The Effects on Populations of Exposure to Low Levels of Ionizing Radiation., National Academy Press, Washington, D.C.

National Research Council, 1976. Halocarbons, Environmental Effects of Chlorofluoromethane Release.

National Science Foundation, 1975. The Possible Impact of Fluorocarbons and Halocarbons on Ozone.

NCRP (National Council on Radiation Protection and Measurements), 1975. Natural Background Radiation in the United States, NCRP Report No. 45, Washington, D.C.

NCRP (National Council on Radiation Protection and Measurements), 1984. Exposures from the Uranium Series with Emphasis on Radon and its Daughters, Report No. 77.

NCRP (National Council on Radiation Protection and Measurements), 1984. Radiological Assessment: Predicting the Transport, Bioaccumulation, and Uptake by Man of Radionuclides Released to the Environment, Report No. 76.

NCRPM (National Council on Radiation Protection and Measurements), 1975. Natural Background Radiation in the United States, NCRP Report 45.

NCRPM (National Council on Radiation Protection and Measurements), 1976. Environmental Radiation Measurements, NCRP Report 50.

Nielsen, J., 1976. An Archaeological Reconnaissance of Sandia National Laboratories' Los Medanos Nuclear Waste Disposal Facility, Eddy County, New Mexico, SAND77-7024, Sandia National Laboratories, Albuquerque, New Mexico.

Nielsen, J., 1977. An Archaeological Reconnaissance of a Proposed Site for the Sandia National Laboratories, Albuquerque, New Mexico, SAND77-7024, Sandia National Laboratories, Albuquerque, New Mexico.

NIOSH (National Institute for Occupational Safety and Health), 1985. NIOSH Pocket Guide to Chemical Hazards, U.S. Department of Health and Human Services, Public Health Service Centers for Disease Control.

Nitschke, K. D., J. D. Burek, T. Bell, L. W. Rampy, and M. G. McKenna, 1982. "Methylene Chloride: A Two Year Inhalation Toxicity and Oncogenicity Study," Toxicology Research Laboratory, Health and Environmental Sciences, Dow Chemical Company, Final Report.

NOAA (National Oceanic and Atmospheric Administration), 1972-1976. Climatological Data, National Climatic Center, Asheville, North Carolina.

NOAA (National Oceanic and Atmospheric Administration), 1974. Climates of the United States, Vol. 2--Western States, Roswell, New Mexico, Water Information Center, Inc., Port Washington, New York.

Nomiyama, E., and H. Nomiyama, 1977. "Dose-Responses Relationship for Trichloroethylene in Man," Int. Arch. Occup. Environ. Health, Vol. 30, pp. 237-248.

Nomiyama, E., H. Nomiyama, and H. Arai, 1986. "Re-evaluation of Subchronic Toxicity of Trichloroethylene," Toxicol. Lett. 31 (Suppl.), 225 (Abstract).

Norpoth, K., V. Witting, M. Springorans, and C. Wittig, 1974. "Induction of Microsomal Enzymes in the Rate Liver by Inhalation of Hydrocarbon Solvents," Int. Arch. Aberted, Vol. 33, pp. 315-321.

NRC (U.S. Nuclear Regulatory Commission), 1976. "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Regulatory Guide 1.10911, Washington, D.C.

NRC (Nuclear Regulatory Commission), 1977. Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes, NUREG-0170, Vol. 1, Office of Standards Development, Washington, D.C.

NRC (Nuclear Regulatory Commission), 1977. "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I (Revision I)," Regulatory Guide 1.109, Office of Standards Development, Washington, D.C.

NRC (Nuclear Regulatory Commission), 1980. Accident Generated Particulate Materials and Their Characteristics - A Review of Background Information, NUREG/CR-2651, PNL-4154, Pacific Northwest Laboratory.

NTP (National Toxicology Program), 1982. "Carcinogenesis Bioassay of Trichloroethylene in F344 Rats and B6C3F1 Mice," CAS No. 70-01-06, NTP 81-94, NIH Publication No. 82-1799.

NTP (National Toxicology Program), 1986. "Toxicology and Carcinogenesis Studies of Dichloromethane (Methylene Chloride) (CAS No. 75-09-2) in F344/N Rats and B6C3F1 Mice (Inhalation Studies)," Tech. Rep. Ser. 306.

NTP (National Toxicology Program), 1986. "Toxicology and Carcinogenesis Studies of Trichloroethylene in F344/N Rats and B6C3F1/Mice," NTP TR 243, U. S. Department of Health and Human Services, National Institutes of Health, Bethesda, Maryland.

NUS Corporation, 1979. Environmental Characterization of Bedded Salt Formation and Overlying Areas of the Permian Basin, ONWI-27, prepared by NUS Corporation for the Office of Nuclear Waste Isolation, Battelle Memorial Institute, Columbus, Ohio.

Oehme, F. W., 1979. Toxicity of Heavy Metals in the Environment, Part 2, Macel Dekker, Inc.

Pasquill, F., 1961. "The Estimation of the Dispersion of Windborne Material," Meteorological Mag., Vol. 90, p. 33.

Pocalujka, L. P., et al., 1979. Meteorological and Air Quality Data Quarterly Report, WIPP Site: Eddy County, New Mexico, Winter Quarter, December 1976 through February 1977, Sandia National Laboratories, Albuquerque, New Mexico.

Pocalujka, L. P., et al., 1979. Meteorological Data Quarterly Report, WIPP Site: Eddy County, New Mexico, Autumn Quarter, September through November 1976, SAND79-7041, Sandia National Laboratories, Albuquerque, New Mexico.

Pocalujka, L. P., et al., 1980. Meteorological and Air Quality Data Quarterly Report, WIPP Site: Eddy County, New Mexico, Spring Quarter, March 1977 through May 1977, SAND79-7109, Sandia National Laboratories, Albuquerque, New Mexico.

Pocalujka, L. P., et al., 1980. Meteorological and Air Quality Data Quarterly Report, WIPP Site: Eddy County, New Mexico, Summer Quarter, June 1977 through August 1977, SAND80-7107, Sandia National Laboratories, Albuquerque, New Mexico.

Pocalujka, L. P., et al., 1980. Meteorological and Air Quality Data Quarterly Report, WIPP Site

Eddy County, New Mexico, Autumn Quarter, September 1977 through November 1977, SAND80-7121, Sandia National Laboratories, Albuquerque, New Mexico.

Pocalujka, L. P., et al., 1981. Meteorological and Air Quality Data Quarterly Report, WIPP Site: Eddy County, New Mexico, Winter Quarter, December 1977 through February 1978.

Pocalujka, L. P., et al., 1981. Final Meteorological and Air Quality Data Report, WIPP Site: Eddy County, New Mexico, Spring Quarter March 1978 through February 1980, SAND81-7052, Sandia National Laboratories, Albuquerque, New Mexico.

Poston, J. W., 1985. Internal Dose Assessment, Technical Management Services, New Hartford, Connecticut.

Price, P. J., C. M. Hassett, and J. J. Mansfield, 1978. "Transforming Activities of Trichloroethylene and Proposed Industrial Alternatives," In Vitro, Vol. 14, pp. 290-293.

Prill, S. D., and G. R. Buckle, 1988. Guidance Manual, Surface Water and Sediment Sampling for the Environmental Monitoring Programs, Waste Isolation Pilot Plant, DOE/WIPP 88-007, U.S. Department of Energy, Carlsbad, New Mexico.

Ratcliffe, J. M., 1981. Lead in Man and the Environment, Ellis Horwood Ltd.

Reinhardt, C. F., L. S. Mullen, and M. E. Maxfield, 1973. "Epinephrine-Induced Cardiac Arrhythmia Potential of Some Common Industrial Solvents," J. Occup. Med., Vol. 15, p. 953.

Reith, C. C., 1985. Environmental Data Acquisition at the Waste Isolation Pilot Plant: A Chronological Description and Justification Matrix, Contract No. DE-AC04-78AL05346, IT Corporation, Carlsbad, New Mexico.

Reith, C. C., and G. Daer, 1985. Radiological Baseline Program for the Waste Isolation Pilot Plant: Program Plan, WTSD-TME-057, Waste Isolation Pilot Plant, Carlsbad, New Mexico.

Reith, C. C., R. J. Eastmond, and R. F. Kehrman, 1986. "Preoperational Environmental Programs for the Waste Isolation Pilot Plant," Waste Isolation in the U.S., Technical Programs and Public Transportation, Vol. 1, pp. 81-87.

Reith, C. C., and E. T. Louderbough, 1986. "The Dispersal and Impact of Salt from Surface Storage Piles at WIPP," Proceedings of the Symposium of Waste Management '86, Tucson, Arizona.

Reith, C. C., E. T. Louderbough, R. J. Eastmond, and A. L. Rodriguez, 1985. Ecological Monitoring Program for the Waste Isolation Pilot Plant, SemiAnnual Report: July-December 1984, WTSD-TME-058, Waste Isolation Pilot Plant, Carlsbad, New Mexico.

Reith, C. C., J. K. Prince, N. T. Fischer, A. Rodriguez, D. Uhland, and D. J. Winstanley, 1985. Annual Site Environmental Monitoring Report for the Waste Isolation Pilot Plant, Carlsbad, New Mexico, DOE/WIPP 86-002, Waste Isolation Pilot Plant, Carlsbad, New Mexico.

SSandia National Laboratories, 1977. Preliminary Review Copy--Draft Environmental Impact Statement for Waste Isolation Pilot Plant, Eddy County, New Mexico, SAND77-0650, Albuquerque, New Mexico.

Sandia National Laboratories, 1979. Summary of Research and Development Activities in Support of Waste Acceptance Criteria for WIPP, SAND79-1305, Sandia National Laboratories, Albuquerque, New Mexico.

Santos, H., 1987. RADCOMP V1.5 User Reference Manual, Waste Isolation Pilot Plant, Carlsbad, New Mexico.

Sawyer, S., 1988. "Applications Analysis Report Hazcon Solidification," U.S. Environmental Protection Agency SITE Report, Enviresponse, Inc., Edison, New Jersey.

Sax, N. I., 1984. Dangerous Properties of Industrial Materials, sixth edition, New Van Nostrand Reinhold Company.

Sax, N. I., 1989. Dangerous Properties of Industrial Materials, seventh edition, New Van Nostrand Reinhold Company.

Shefelbine, H. C., 1978. Preliminary Evaluation of the Characteristics of Defense Transuranic Wastes, SAND78-1850, Sandia National Laboratories, Albuquerque, New Mexico.

Sittig, M., 1979. Hazardous and Toxic Effects of Industrial Chemicals, Moyes Data Corp.

Sittig, M., 1985. Handbook of Toxic and Hazardous Chemicals and Carcinogens, second edition, Noyes Publications, New Jersey.

Soil Conservation Service, 1971. Soil Survey of Eddy Area, New Mexico, Government Printing Office, Washington, D.C.

Stewart, R. D., T. N. Fischer, M. J. Hosko, J. L. Peterson, E. D. Baretta, and H. C. Dodd, 1972. "Experimental Human Exposure to Methylene Chloride," Arch. Environ. Health, Vol. 25, pp. 342-348.

Stinebaugh, R. E., 1979. Retrieval of Canistered Experimental Waste at the Waste Isolation Pilot Plant, SAND79-1239, Sandia National Laboratories, Albuquerque, New Mexico.

Sutter, S. L., 1982. "Accident Generated Particulate Materials and their Characteristics: A Review of Background Information," May 1982, PNL-4154, Pacific Northwest Laboratories, Richland, Washington.

Swanson, J. L., 1986. Organic Complexant-Enhanced Mobility of Toxic Elements in Low-Level Wastes: Final Report, NUREG/CR-4660, PNL-4965-10, Pacific Northwest Laboratory, Richland, Washington.

Thompson, D. J., 1988. Environmental Thermoluminescent Dosimetry Measurements at the WIPP Site, 1976-1985, SAND87-0843, Sandia National Laboratories, Albuquerque, New Mexico.

Till, J. E., and H. R. Meyer, 1983. Radiological Assessment: A Textbook on Environmental Dose Analysis, NUREG/CR-3332, U.S. Nuclear Regulatory Commission, Washington, D.C.

Torkelson, T. R., F. Oyen, D. D. McCollister, and V. K. Rowe, 1958. "Toxicity of 1,1,1-Trichloroethane as Determined on Laboratory Animals and Human Subjects," Am. Ind. Hyg. Assoc. J., Vol. 19, pp. 353-362.

Torres, B., and R. Balestri, 1978. Consequence Assessment of Radionuclide Release to the Environment from the WIPP Repository, BDM/TAC-78-497-TR, The BDM Corporation, Albuquerque, New Mexico.

University of New Mexico, 1983. A Handbook of Rare and Endemic Plants of New Mexico, New Mexico Native Plants Protection Advisory Committee, University of New Mexico Press, Albuquerque, New Mexico.

UNSCEAR (United Nations Scientific Committee on the Biological Effects of Atomic Radiation), 1972. "Ionizing Radiation: Levels and Effects, A Report to the General Assembly, with Annexes."

UNSCEAR (United Nations Scientific Committee on the Biological Effects of Atomic Radiation), 1988. "Sources, Effects and Risks of Ionizing Radiation, A Report to the General Assembly, with Annexes."

USDHHS (U.S. Department of Health and Human Services), 1978. Occupational Health Guidelines, Cincinnati, Ohio.

USDHHS (U.S. Department of Health and Human Services), 1985. Fourth Annual Report on Carcinogens, NTP 85-002, National Toxicology Program, Research Triangle Park, North Carolina.

USPHS (U.S. Public Health Service), 1970. Radiological Health Handbook, Rockville, Maryland.

Verschueren, K., 1983. Handbook of Environmental Data on Organic Chemicals, second edition, Van Nostrand Reinhold Co., New York.

Vogel, T. M., and P. L. McCarthy, 1985. "Biotransformation of Tetrachloroethylene to Trichloroethylene, Dichloroethylene, Vinyl Chloride, and Carbon Dioxide Under Methanogenic Conditions," Appl. Environ. Microbiol., Vol. 49, pp. 1030-1083.

Weinstein, R. S., and S. S. Diamond, 1972. "Hepatotoxicity of Dichloromethane (Methylene Chloride) with Continuous Inhalation Exposure at a Low Dose Level," in Proceedings of the Third Annual Conference on Environmental Toxicology, Wright-Patterson Air Force Base, Ohio, Aerospace Medical Research Laboratory, pp. 209-220.

Weisburger, E. K., 1977. "Carcinogenicity Studies in Halogenated Hydrocarbons," Environ. Health Perspec., Vol. 21, pp. 7-16.

Welch, L., 1987. "Reports of Clinical Disease Secondary to Methylene Chloride Exposure--A Collection of 141 Cases," ONTS/EPA 3/31.

WEC (Westinghouse Electric Corporation), 1980. Environmental Procedure and Instruction Manual, DOE/WIPP-02-3, Waste Isolation Pilot Plant, Carlsbad, New Mexico.

White, J. F., and G. P. Carlson, 1979. "Influence of Alternations in Drug Metabolism in Spontaneous and Epinephrine-Induced Cardiac Arrhythmias in Animals Exposed to Trichloroethylene," Toxico. Appl. Pharmacol., Vol. 47, pp. 515-527.

White, J. F., and G. P. Carlson, 1981. "Epinephrine-Induced Cardiac Arrhythmias in Rabbits Exposed to Trichloroethylene Potentiation by Ethanol," Toxico. Appl. Pharmacol., Vol. 60, pp. 466-471.

Wolfe, H. G., et al., eds., 1977. An Environmental Baseline Study of the Los Medanos Waste Isolation Pilot Plant (WIPP) Project Area of New Mexico: A Progress Report, SAND77-7017, Sandia National Laboratories, Albuquerque, New Mexico.

Wolfe, H. G., et al., eds., 1977. An Environmental Baseline Study of the Los Medanos Waste Isolation Pilot Plant (WIPP) Project Area of New Mexico: A Progress Report (Addendum), SAND77-7018, Sandia National Laboratories, Albuquerque, New Mexico.

J.4 **GEOCHEMISTRY**

Adams, S. S., 1967. "Bromine in the Salado Formation, Carlsbad Potash District, New Mexico," unpublished dissertation, Harvard University, Cambridge, Massachusetts, 202 pp.

Adams, S. S., 1969. "Bromine in the Salado Formation, Carlsbad Potash District, New Mexico," New Mexico Bureau of Mines and Mineral Resources Bulletin 93, 112 pp.

Barr, G. E., and B. F. Jones, 1988. "Normative Analyses of Groundwater from the Rustler Formation, Associated with the Waste Isolation Pilot Plant (WIPP) in Southeastern New Mexico," U.S. Geological Survey, unpublished data.

Barr, G. E., S. J. Lambert, and J. A. Carter, 1979. "Uranium-Isotope Disequilibrium in Groundwaters of Southeastern New Mexico and Implications Regarding Age-Dating of Waters," STI/PUB/403, IAEA-SM-228/32, Proceedings of the International Symposium on Isotope Hydrology, 1978, International Atomic Energy Agency, Vienna, Austria.

Barr, G. E., S. J. Lambert, and J. A. Carter, 1979. Uranium-Isotope Disequilibrium in Groundwaters of Southeastern New Mexico and Implications Regarding Age-Dating of Waters. Also published as SAND77-1799, Sandia National Laboratories, Albuquerque, New Mexico.

Bodine, M. W., 1978. "Clay-Mineral Assemblages from Drill Core of Ochoan Evaporites, Eddy County," Geology and Mineral Deposits of Ochoan Rocks in the Delaware Basin and Adjacent Areas, New Mexico Bureau of Mines and Mineral Resources Circular 159, Socorro, New Mexico.

Bodine, M. W., Jr., and B. F. Jones, 1986. "The Salt Norm: A Quantitative Chemical-Mineralogical Characterization of Natural Waters," U.S. Geological Survey Water-Resources Investigation Report 86-4086.

Bodine, M. W., Jr., B. F. Jones, and S. J. Lambert, 1988. "Normative Analysis of Groundwaters from the Rustler Formation," in Hydrogeochemical Studies of the Rustler Formation and Related Rocks in the WIPP Area, Southeastern New Mexico, eds. M. D. Siegel, S. J. Lambert, and K. L. Robinson, SAND88-0196, Sandia National Laboratories, Albuquerque, New Mexico.

Bodine, M., and C. MacMillan, 1978. Investigation of Petrology, Mineralogy and Geochemistry of a Permian Basin Cone, interim report prepared for Sandia National Laboratories, Albuquerque, New Mexico.

Brookins, D. G., 1981. "Geochronologic Studies Near the WIPP Site, Southeastern New Mexico," New Mexico Geological Society Special Publication No. 10, pp. 147-152.

Brookins, D. G., 1988. Eh-pH Diagrams for Geochemistry, Springer-Verlag, Berlin-Heidelberg-New York.

Brookins, D. G., and S. J. Lambert, 1987. "Radiometric Dating of Ochoan (Permian) Evaporites, WIPP Site, Delaware Basin, New Mexico, USA," Materials Research Society Symposium Proceedings, Vol. 84, pp. 771-780.

Brookins, D. G., and J. K. Register, 1981. Potassium-Argon and Neutron Activation Analysis Studies of Polyhalites Near the WIPP Site, SAND81-7071, Sandia National Laboratories, Albuquerque, New Mexico.

Brookins, D. G., J. K. Register, M. E. Register, and S. J. Lambert, 1980. "Long-Term Stability of Evaporite Minerals, Geochronological Evidence," SAND79-1283, in Proceedings of International Symposium on Scientific Basis for Nuclear Waste Management, Materials Research Society, Vol. 2, pp. 479-486.

Brush, L. H., 1989. Test Plan for Laboratory and Modeling Studies of Repository and Radionuclide Chemistry, draft, Sandia National Laboratories, Albuquerque, New Mexico.

Chapman, J. B., 1986. Stable Isotopes in Southeastern New Mexico Groundwater: Implications for Dating Recharge in the WIPP Area, EEG-35, State of New Mexico Environmental Evaluation Group, Santa Fe, New Mexico.

Chapman, J. B., 1988. Chemical and Radiochemical Characteristics of Groundwater in the Culebra Dolomite, Southeastern New Mexico, EEF-35. New Mexico Environmental Evaluation Group, Santa Fe, New Mexico.

Chapman, N. A., and J. A. T. Smellie, 1986. "Natural Analogues to the Conditions around a Final Repository for High-Level Radioactive Waste," Chemical Geology, June, Vol. 55, No. 3/4.

CHEMTOX Database, 1988. Resource Consultants, Brentwood, Tennessee.

Clark, D. G. and D. J. Tinston, 1973. Br. F. Pharm., Vol. 49, p. 355.

Colton, I. D., and J. G. Morse, 1985. Water Quality Sampling Plan, WIPP/DOE-215, Waste Isolation Pilot Plant, Carlsbad, New Mexico.

Craig, H., 1961. "Isotopic Variations in Meteoric Waters," Science, Vol. 133, pp. 1702-1703.

Dosch, R. G., and G. W. Lynch, 1978. Interaction of Radionuclides with Geomedia Associated with the Waste Isolation Pilot Plant (WIPP) Site in New Mexico, SAND78-0297, Sandia National Laboratories, Albuquerque, New Mexico.

Drever, J. I., 1982. The Geochemistry of Natural Waters, Prentice Hall, Englewood Cliffs, New Jersey, pp. 78-79.

EEG (Environmental Evaluation Group), 1983. The Geochemistry of Two Pressurized Brines From the Castile Formation in the Vicinity of the Waste Isolation Pilot Plant (WIPP) Site, EEG-21, S. Faith, P. Spiegler and K. R. Rehfeldt, Health and Environment Department, State of New Mexico, Santa Fe, New Mexico.

EEG (Environmental Evaluation Group), 1984. Occurrence of Gases in the Salado Formation, EEG-25, L. Chaturvedi, Health and Environment Department, State of New Mexico, Santa Fe, New Mexico.

EEG (Environmental Evaluation Group), 1985. Chemistry of Rustler Fluids, EEG-31, D. S. Ramey, Health and Environment Department, State of New Mexico, Santa Fe, New Mexico.

Epstein, S., R. P. Sharp, and A. J. Gow, 1965. "Six-Year Record of Oxygen and Hydrogen Isotope Variations in South Pole Firn," Jour. Geoph. Res., Vol. 70, pp. 1809-1814.

Epstein, S., R. P. Sharp, and A. J. Gow, 1970. "Antarctic Ice Sheet: Stable-Isotope Analyses of Byrd Station Cores and Interhemispheric Climatic Implications," Science, Vol. 168, pp. 1570-1572.

Fogel, M. M., A. R. Taddio, and S. Fogel, 1986. "Biodegradation of Chlorinated Ethenes by a Methane-Utilizing Mixed Culture," App. Environ. Micro., Vol. 51, pp. 720-4.

Gonzalez, D. D., 1983. Hydrogeochemical Parameters of Fluid-Bearing Zones in the Rustler and Bell Canyon Formations: Waste Isolation Pilot Plant (WIPP), Southeastern New Mexico, SAND83-0210, Sandia National Laboratories, Albuquerque, New Mexico.

Gross, G. W., R. N. Hoy, and C. J. Duffy, 1976. Application of Environmental Tritium in the Measurement of Recharge and Aquifer Parameters in a Semi-Arid Limestone Terrain, Report No. 60, New Mexico Water Resources Research Institute.

Hallen, R. T., J. W. Pyne, Jr., and P. M. Malton, 1986. "Transformation of Chlorinated Ethenes and Ethanes by Anaerobic Microorganisms," in ACS Div. Environ. Chem., 192nd National Mt., Vol. 25, pp. 344-6.

Hansch, C. and A. J. Leo, 1979. Substituent Constraints for Correlation Analysis in Chemistry and Biology, John Wiley and Sons, New York.

Hayes, K. F., C. Papelis, and J. O. Leckie, 1988. "Modeling Ionic Strength Effects on Anion Adsorption at Hydrous Oxide/Solution Interfaces," Journal of Colloid and Interface Science, October, Vol. 125, No. 2.

Hazline (Hazard Line), January 1989. On line.

Hoy, R. N., and G. W. Gross, 1982. A Baseline Study of Oxygen 18 and Deuterium in the Roswell, New Mexico, Groundwater Basin, Report No. 144, New Mexico Water Resources Research Institute.

Kirk-Othmer, ed., 1985. Concise Encyclopedia of Chemical Technology, Wiley-Interscience.

Knauth, L. P., and M. A. Beeunas, 1986. "Isotope Geochemistry of Fluid Inclusions in Permian Halite, with Implications for the Isotopic History of Ocean Water and the Origin of Saline Formation Waters," Geochimica at Cosmochimica Acta, Vol. 50, pp. 419-434.

Kopp, O. D., and D. W. Combs, 1975. "Mineral Sources of Water in Evaporite Sequences Report," ORNL Subcontract No. 3670, Department of Geological Science, University of

Tennessee.

Lambert, S. J., 1977. The Geochemistry of Delaware Basin Groundwaters, SAND77-0420, Sandia National Laboratories, Albuquerque, New Mexico.

Lambert, S. J., 1978. "The Geochemistry of Delaware Basin Groundwaters," Geology and Mineral Deposits of Ochoan Rocks in Delaware Basin and Adjacent Areas, New Mexico Bureau of Mines and Mineral Resources Circular 159, pp. 32-38.

Lambert, S. J., 1986. K-Ar and Rb-Sr Age Determinations from Clay Minerals and Related Minerals from the WIPP, Southeastern New Mexico, SAND86-0223J, Sandia National Laboratories, Albuquerque, New Mexico.

Lambert, S. J., 1986. "Radiometric Dating of Ochoan (Permian) Evaporites, WIPP Site, Delaware Basin, New Mexico," SAND86-2192C, Sandia National Laboratories, Albuquerque, New Mexico, published in the Proceedings of Materials Research Society Symposium, Boston, Massachusetts.

Lambert, S. J., 1987. Feasibility Study: Applicability of Geochronologic Methods Involving Radiocarbon and Other Nuclides to the Groundwater Hydrology of the Rustler Formation, SAND86-1054, Sandia National Laboratories, Albuquerque, New Mexico.

Lambert, S. J., 1987. "Stable-Isotope Studies of Groundwaters in Southeastern New Mexico," SAND85-1978, Sandia National Laboratories, Albuquerque, New Mexico, in The Rustler Formation at the WIPP Site, ed. L. Chaturvedi, EEG-34, EEG-21, State of New Mexico Environmental Evaluation Group, Santa Fe, New Mexico.

Lambert, S. J., 1988. "Isotopic Constraints on the Rustler and Dewey Lake Groundwater Systems," in Hydrogeochemical Studies of the Rustler Formation and Related Rocks in the WIPP Area, Southeastern New Mexico, eds. M. D. Siegel, S. J. Lambert, and K. L. Robinson, SAND88-0196, Sandia National Laboratories, Albuquerque, New Mexico.

Lambert, S. J., and J. A. Carter, 1987. Uranium-Isotope Systematics in Groundwaters of the Rustler Formation, Northern Delaware Basin, Southeastern New Mexico, SAND87-0388, Sandia National Laboratories, Albuquerque, New Mexico.

Lambert, S. J., and J. A. Carter, (in press). "Uranium-Isotope Disequilibrium in Brine Reservoirs of the Castile Formation, Northern Delaware Basin, Southeastern New Mexico," SAND83-0144, Sandia National Laboratories, Albuquerque, New Mexico.

Lambert, S. J., and D. M. Harvey, 1987. Stable-Isotope Geochemistry of Groundwaters in the Delaware Basin of Southeastern New Mexico, SAND87-0138, Sandia National Laboratories, Albuquerque, New Mexico.

Lambert, S. J., and K. L. Robinson, 1984. Field Geochemical Studies of Groundwaters in Nash Draw, Southeastern New Mexico, SAND83-1122, Sandia National Laboratories, Albuquerque, New Mexico.

Langmuir, D., and A. C. Riese, 1985. "The Thermodynamic Properties of Radium," Geochimica et Cosmochimica Acta, Vol. 49, pp. 1593-1601.

Li, Y., and S. Gregory, 1974. "Diffusion of Ions in Seawater and in Deep-Sea Sediments," Geochimica et Cosmochimica Acta, Vol. 38, pp. 703-714.

Melfi, J. R., 1985. A Geochemical Model for Radionuclide Speciation in High Ionic Strength Solutions, Abstract of Thesis, University of New Mexico, Albuquerque, New Mexico.

Myers, J., P. James, and P. Drez, 1988. "The Redox State of the Culebra Member of the Rustler Formation," in Hydrogeochemical Studies of the Rustler Formation and Related Rocks in the WIPP Area, Southeastern New Mexico, eds. M. D. Siegel, S. J. Lambert, and K. L. Robinson, SAND88-0196, Sandia National Laboratories, Albuquerque, New Mexico.

Niou, S., and D. E. Deal, 1989. Migration of Brine and Nitrogen in Creeping Salt, Proc. Waste Management 1989, University of Arizona, Tucson.

O'Neil, J. R., C. M. Johnson, L. D. White, and E. Roedder, 1986. "The Origin of Fluids in the Salt Beds of the Delaware Basin, New Mexico and Texas," Journal of Applied Geochemistry, Vol. 1.

Paine, R. T., 1977. Chemistry Related to the WIPP Site, Draft Final Report, Sandia Contract GTK/07-1488, Chemistry Department, University of New Mexico, Albuquerque, New Mexico.

Plummer, N. L., D. L. Parkhurst, G. W. Fleming, and S. A. Dunkle, 1988. "PHRWPITZ: A Computer Program for Geochemical Calculations in Brines," U.S. Geological Survey, unpublished data.

Potter, R. W., II, 1977. "Pressure Corrections for Fluid-Inclusion Homogenization Temperatures Based on the Volumetric Properties of the System NaCl-H2O," Journal of Research U.S. Geological Survey, Vol. 5, No. 5, pp. 603-607.

Rai, D., and M. J. Mason, 1977. Kd Measurements of Nuclides to Estimate Migration Potential at the Proposed Waste Isolation Pilot Plant in New Mexico, PNL-2448, Pacific Northwest Laboratory, Richland, Washington.

Randall, W. S., M. E. Crawley, and M. L. Lyon, 1988. Annual Water Quality Data Report for the Waste Isolation Pilot Plant, DOE/WIPP-88-006, U.S. Department of Energy.

Register, J. K. Rubedium-Strontium and Related Studies of the Salado Formation, Southeastern New Mexico, SAND81-7072, Sandia National Laboratories, Albuquerque, New Mexico.

Register, J. K., D. G. Brookins, M. E. Register, and S. J. Lambert, 1980. "Clay Mineral-Brine Interaction During Evaporite Diagenesis: Lanthanide Distribution in WIPP Samples," SAND79-1284C, Sandia National Laboratories, Albuquerque, New Mexico, presented in Proceedings of International Symposium on Scientific Basis for Nuclear Waste Management, Materials Research Society, Vol. 2, pp. 445-452.

Riese, A. C., 1982. Adsorption of Radium and Thorium onto Quartz and Kaolinite: A Comparison of Solution/Surface Equilibria Models, Ph.D. Thesis, Colorado School of Mines, Golden, Colorado.

Robinson, K. L., 1988. Analysis of Solutes in Groundwaters from the Rustler Formation at and Near the WIPP Site, SAND86-0917, Sandia National Laboratories, Albuquerque, New Mexico.

Roedder, E., and H. E. Belkin, 1980. "Thermal Gradient Migration of Fluid Inclusions in Single Crystals of Salt from the Waste Isolation Pilot Plant," Proceedings of International Symposium on Scientific Basis for Nuclear Waste Management, Materials Research Society, Vol. 2, pp. 453-464.

Sewards, T., M. L. Williams, C. L. Stein, and K. Keil, 1988. "Mineralogy of the Culebra Dolomite," in Hydrogeochemical Studies of the Rustler Formation and Related Rocks in the WIPP Area, Southeastern New Mexico, eds. M. D. Siegel, S. J. Lambert, and K. L. Robinson, SAND88-0196, Sandia National Laboratories, Albuquerque, New Mexico.

Siegel, M. D., S. J. Lambert, and K. L. Robinson, eds., 1989. Hydrogeochemical Studies of the Rustler Formation and Related Rocks in the WIPP Area, Southeastern New Mexico, SAND88-0196, Sandia National Laboratories, Albuquerque, New Mexico.

Siegel, M. D., K. L. Robinson, and J. Myers, ed., 1988. "Solute Relationships in Groundwaters From the Culebra Member of the Rustler Formation Near the WIPP Site, Southeastern New Mexico," in Hydrogeochemical Studies of the Rustler Formation and Related Rocks in the WIPP Area, Southeastern New Mexico, eds. M. D. Siegel, S. J. Lambert, and K. L. Robinson, SAND88-0196, Sandia National Laboratories, Albuquerque, New Mexico.

Stein, C. L., 1985. Mineralogy in the Waste Isolation Pilot Plant (WIPP) Facility Stratigraphic Horizon, SAND85-0321, Sandia National Laboratories, Albuquerque, New Mexico.

Stein, C. L., 1985. Preliminary Report on Fluid Inclusions from Halites in the Castile and Lower Salado Formations of the Delaware Basin, Southeastern New Mexico, SAND83-0451, Sandia National Laboratories, Albuquerque, New Mexico.

Stein, C. L., and J. L. Krumhansl, 1984. "Compositions of Brines in Halite from the Lower Salado Formation, Southeastern New Mexico," SAND84-1252A, Sandia National Laboratories, Albuquerque, New Mexico, presented at GSA Annual Meeting, Reno, Nevada, November 5-8, 1984, Abstracts with Programs, 97th Annual Meeting, Geological Society of America, Vol. 16, No. 6, p. 667.

Stein, C. L., and J. L. Krumhansl, 1985. An Interpretational Model for the Evolution of Brines in Salt from the Lower Salado Formation, Southeastern New Mexico, SAND85-1999J, Sandia National Laboratories, Albuquerque, New Mexico, submitted to Geochimica at Cosmochimica Acta.

Stein, C.L., and J. L. Krumhansl, 1986. Chemistry of Brines in Salt From the Waste Isolation Pilot Plant (WIPP), Southeastern New Mexico: A Preliminary Investigation, SAND85-0897, Sandia National Laboratories, Albuquerque, New Mexico.

Sutherland, H. J., and S. Cave, 1978. Gas Permeability of SENM Rock Salt, SAND78-2287, Sandia National Laboratories, Albuquerque, New Mexico.

Szabo, B. J., W. C. Gottschall, J. N. Rosholt, and C. R. McKinney, 1980. Part I: A Preliminary Study of Uranium-Thorium Systematics in Dissolution Residues at the Top of Salado Evaporites--Implications to Process and Time of the Last Vertical Dissolution Cycle of Salt Beds Near the WIPP Site; Part II: Uranium Trend Dating of Surficial Deposits and Gypsum Spring Deposit Near WIPP Site, New Mexico, U.S. Geological Survey Open-File Report 80-879.

Thomson, B. M., and R. J. Heggen, 1983. "Uranium and Water: Managing Common Resources," Chemtech, May.

Torstenfelt, B., H. Kipatsi, K. Anderson, B. Allard, and University Olofsson, 1982. "Transport of Actinides Through a Bentonite Backfill," in Scientific Basis for Radioactive Waste Management V, ed. W. Lutze, Elsevier Science Publishing Company, New York, pp. 659-668.

Tsunashima, A., G. W. Brindley, and M. Bastovanov, 1981. "Adsorption of Uranium from Solutions by Montmorillonite: Compositions and Properties of Uranyl Monmorillonites," Clays and Clay Minerals, Vol. 29, No. 1.

Uhland, D. W., and W. S. Randall, 1986. Annual Water Quality Data Report for the Waste Isolation Pilot Plant, DOE/WIPP-86-006, U.S. Department of Energy.

Uhland, D. W., W. S. Randall, and R. C. Carrascco, 1987. 1987 Annual Water Quality Data Report for the Waste Isolation Pilot Plant, DOE/WIPP 87-006, U.S. Department of Energy, Carlsbad, New Mexico.

Wolery, T. J., 1983. EQ3NR—A Computer Program for Geochemical Aqueous Speciation-Solubility Calculations: User's Guide and Documentation, UCRL-53414, Lawrence Livermore National Laboratory, Livermore, California.

J.5 **GEOLOGY**

Acton, R. V., 1977. "Thermal Conductivity of SE New Mexico Rocksalt and Anhydrite," unpublished Sandia National Laboratories preliminary report, Albuquerque, New Mexico.

Adams, J. E., 1944. "Upper Permian Ochoa Series of the Delaware Basin, West Texas and Southeastern New Mexico," American Association of Petroleum Geologists Bulletin, Vol. 48, pp. 1596-1625.

Adams, J. E., 1965. "Stratigraphic-Tectonic Development of Delaware Basin," American Association of Petroleum Geologists Bulletin, Vol. 49, pp. 2140-2148.

Adams, J. E., 1972. "Semi-Cyclicity in the Castile Evaporite," in Cyclic Sedimentation in the Permian Basin, West Texas Geological Society Publication 72-60, pp. 196-202.

Algermissen, S. T., 1969. Seismic Risk Studies in the United States, Proceedings of the Fourth World Conference on Earthquake Engineering, Santiago, Chile, Vol. 1, pp. 14-27.

Algermissen, S. T., and D. M. Perkins, 1976. A Probabilistic Estimate of Maximum Acceleration in Rock in the Contiguous United States, U.S. Geological Survey Open-File Report 76-416.

Anderson, R. Y., 1976. Correlation and Structural Relationships of the Castile Salado Evaporite Sequence in ERDA 6, AEC 7 and AEC 8 Boreholes, Eddy and Lea Counties, New Mexico, University of New Mexico, Albuquerque, New Mexico.

Anderson, R. Y., 1978. Deep Dissolution of Salt, Northern Delaware Basin, New Mexico, University of New Mexico, prepared for Sandia National Laboratories, Albuquerque, New Mexico.

Anderson, R. Y., 1979. "Notes on Dissolution and Brine Chambers Related to the WIPP," memo submitted to State of New Mexico/Environmental Evaluation Group, Santa Fe, New Mexico.

Anderson, R. Y., 1981. "Deep-Seated Dissolution in the Delaware Basin, Texas and New Mexico," Environmental Geology and Hydrology in New Mexico, eds. S. G. Wells and W. Lambert, New Mexico Geological Society, Special Publication No. 10.

Anderson, R. Y., W. E. Dean, D. N. Kirkland, and H. I. Snyder, 1972. "Permian Castile Varved Evaporite Sequence, West Texas and New Mexico," Bulletin of the Geological Society of America, Vol. 83, pp. 59-86.

Anderson, R. Y., K. K. Kietzke, and D. J. Rhodes, 1978. "Development of Dissolution Breccias, Northern Delaware Basin, New Mexico," Geology and Mineral Deposits of Ochoan Rocks in Delaware Basin and Adjacent Areas, New Mexico Bureau of Mines and Mineral Resources Circular 159, pp. 47-52.

Anderson, R. Y., and D. W. Kirkland, 1980. "Dissolution of Salt Deposits by Brine Density Flow," Geology, Vol. 8, pp. 66-69.

Anderson, R. Y., and D. W. Powers, 1978. "Salt Anticlines in the Castile-Salado Evaporite Sequence, Northern Delaware Basin, New Mexico," Geology and Mineral Deposits of Ochoan Rocks in Delaware Basin and Adjacent Areas, Circular 159, New Mexico Bureau of Mines and Mineral Resources, Socorro, New Mexico.

Ash, S. R., and A. C. Clebsch, 1961. "Cretaceous Rocks in Lea County, New Mexico," U.S. Geological Survey Professional Paper 424-D, pp. 139-142.

Bachman, G. O., 1973. Surficial Features and Late Cenozoic History in Southeastern New Mexico, U.S. Geological Survey Open-File Report 4339-8.

Bachman, G. O., 1974. Geologic Processes and Cenozoic History Related to Salt Dissolution in Southeastern New Mexico, U.S. Geological Survey Open-File Report 74-194.

Bachman, G. O., 1976. "Cenozoic Deposits of Southeastern New Mexico and an Outline of the History of Evaporite Dissolution," U.S. Geological Survey Journal of Research, Vol. 4, No. 2, pp. 135-149.

Bachman, G. O., 1980. Regional Geology and Cenozoic History of Pecos Region, Southeastern New Mexico, U.S. Geological Survey Open-File Report 80-1099.

Bachman, G. O., 1981. Geology of Nash Draw, Eddy County, New Mexico, U.S. Geological Survey Open-File Report 81-31.

Bachman, G. O., 1985. Assessment of Near-Surface Dissolution At and Near the Waste Isolation Pilot Plant (WIPP), Southeastern New Mexico, SAND84-7178, Sandia National Laboratories, Albuquerque, New Mexico.

Bachman, G. O., 1985. Regional Geology of Ochoan Evaporites, Northern Part of Delaware Basin, New Mexico Bureau of Mines and Mineral Resources Circular 184.

Bachman, G. O., 1986. Stratigraphy and Dissolution of the Rustler Formation, SAND86-7077C, Sandia National Laboratories, Albuquerque, New Mexico.

Bachman, G. O., 1987. Karst in Evaporites in Southeastern New Mexico, SAND86-7078, Sandia National Laboratories, Albuquerque, New Mexico.

Bachman, G. O., and R. B. Johnson, 1973. Stability of Salt in the Permian Salt Basin of Kansas, Oklahoma, Texas, and New Mexico, Open-File Report 73-14, U.S. Geological Survey.

Bachman, G. O., R. B. Johnson, and R. A. Swenson, 1972. "Stability of Bedded Salt in the Permian Basin, Colorado, Kansas, Oklahoma, Texas and New Mexico," Oak Ridge National Laboratories Pilot Plant Repository Staff, Alternative Geologic Formations for the Disposal of Radioactive Wastes, ORNL Report 72-6-42.

Bachman, G. O., R. B. Johnson, and F. A. Swenson, 1973. Stability of Salt in the Permian Salt

Basin of Kansas, Oklahoma, Texas, and New Mexico, U.S. Geological Survey Open-File Report 4339-4.

Balazs, E. I., 1982. "Vertical Movement in the Los Medanos and Nash Draw Areas, New Mexico, as Indicated by 1977 and 1981 Leveling Surveys," NOAA Technical Memorandum No. NGS 37, National Geodetic Survey, Rockville, Maryland.

Barnes, H., 1974. "Geologic and Hydrologic Background for Selecting Site of Pilot-Plant Repository for Radioactive Waste," Bulletin of the Association of Engineering Geologists, Vol. 11, No. 1, pp. 83-92.

Barrows, L. J., 1982. "WIPP Geohydrology, The Implications of Karst," unpublished manuscript discussed in Neill et al., 1983, Evaluation of the Suitability of the WIPP Site, EEG-23, State of New Mexico Environmental Evaluation Group, Santa Fe, New Mexico.

Beauheim, R. L., B. W. Hassinger, and J. A. Klaiber, 1983. Basic Data Report for Borehole Cabin Baby-1 Deepening and Hydrologic Testing, DOE/TME-020, Albuquerque, New Mexico.

Bechtel, 1979. Geologic Data and Hole History for Borehole B-25, U.S. Department of Energy Document No. 22-V-510-02.

Bechtel, 1981. "Exploratory Shaft and Station, Instrument Location," Drawing No. 37-X-005, Rev. 5, October 3, 1980, prepared for U.S. Department of Energy Waste Isolation Pilot Plant, Carlsbad, New Mexico.

Bechtel, 1981. "Exploratory Shaft Key, Sections and Details," Drawing No. 37-R-012, Rev. 6, January 6, prepared for U.S. Department of Energy Waste Isolation Pilot Plant, Carlsbad, New Mexico.

Bechtel National Advanced Technology Division, 1985. Quarterly Geotechnical Field Data Report, December 1985, DOE/WIPP-221, U.S. Department of Energy Waste Isolation Pilot Plant Office, Carlsbad, New Mexico.

Bechtel National Inc., 1986. Quarterly Geotechnical Field Data Report, DOE-WIPP-213, U.S. Department of Energy, Carlsbad, New Mexico.

Bell and Murphy and Associates, 1979. Geophysical Investigation of the Waste Isolation Pilot Plant, Eddy County, New Mexico, report to the Department of Energy.

Berg, R. R., 1979. "Reservoir Sandstones of the Delaware Mountain Group, Southeast New Mexico," Guadalupian Delaware Mountain Group of West Texas and Southeast New Mexico, Society of Economic Paleontologists and Mineralogists, Permian Basin Section Publication, 79-18, pp. 75-95.

Bjorklund, L. J., and W. S. Motts, 1959. Geology and Water Resources of Carlsbad Area, New Mexico, U.S. Geolological Survey Open-File Report 59-9.

Bodine, M. W., Jr., 1978. "Clay-Mineral Assemblages from Drill Core of Ochoan Evaporites, Eddy County," in Geology and Mineral Deposits of Ochoan Rocks in the Delaware Basin and Adjacent Areas, Circ. 159, New Mexico Bureau of Mines and Resources, Socorro, New Mexico.

Borns, D. J., 1983. Petrographic Study of Evaporite Deformation Near the Waste Isolation Pilot Plant (WIPP), SAND83-0166, Sandia National Laboratories, Albuquerque, New Mexico.

Borns, D. J., 1985. Marker Bed 139: A Study of Drillcore from a Systematic Array, SAND85-0023, Sandia National Laboratories, Albuquerque, New Mexico.

Borns, D. J., 1987. The Geologic Structures Observed in Drillhole DOE-2 and Their Possible Origins: Waste Isolation Pilot Plant, SAND86-1495, Sandia National Laboratories, Albuquerque, New Mexico.

Borns, D. J., 1987. Rates of Evaporite Deformation: The Role of Pressure Solution, SAND85-1599, Sandia National Laboratories, Albuquerque, New Mexico.

Borns, D. J., L. J. Barrows, D. W. Powers, and R. P. Snyder, 1983. Deformation of Evaporites near the Waste Isolation Pilot Plant (WIPP) Site, SAND82-1069, Sandia National Laboratories, Albuquerque, New Mexico.

Borns, D. J., and S. E. Shaffer, 1985. Regional Well-Log Correlation in the New Mexico Portion of Delaware Basin, SAND83-1798, Sandia National Laboratories, Albuquerque, New Mexico.

Borns, D. J., and J. C. Stormont, 1987. The Delineation of the Disturbed Rock Zone Around Excavations in Salt Waste Isolation Pilot Plant (WIPP), SAND87-1375, Sandia National Laboratories, Albuquerque, New Mexico.

Borns, D. J., and J. C. Stormont, 1988. An Interim Report on Excavation Effect Studies at the Waste Isolation Pilot Plant: The Delineation of the Disturbed Rock Zone, SAND 87-1375, Sandia National Laboratories, Albuquerque, New Mexico.

Borns, D. J., and J. C. Stormont, 1989. "The Delineation of the Disturbed Rock Zone Surrounding Excavations in Salt," Proceedings of the 30th U.S. Rock Mechanics Symposia, Balkema, Rotterdam, The Netherlands.

Brady, B. H. G., and E. T. Brown, 1985. Rock Mechanics for Underground Mining, George Allen and Unwin, London.

Brokaw, A. L., C. L. Jones, M. E. Cooley, and W. H. Hays, 1972. Geology and Hydrology of the Carlsbad Potash Area, Eddy and Lea Counties, New Mexico, U.S. Geological Survey Open-File Report 72-49.

Brookins, D. G., and S. J. Lambert, 1988. "WIPP Site Studies: Secondary Selenite Veins in the Rustler Formation and Dewey Lake Red Beds," Material Resources Society Symposium Proceedings for 1987 (in press).

Brush, L. H., 1988. Feasibility of Disposal of High-Level Radioactive Wastes into the Seabed: Review of Laboratory Investigations of Radionuclide Migration Through Deep-Sea Sediments, SAND87-2438, Sandia National Laboratories, Albuquerque, New Mexico.

Callender, J. F., and T. Ingwell, 1977. Structural Petrology of Undeformed and Experimentally Deformed Halite Samples from USERDA Site #7 and #9, SAND78-7076, Department of Geology, University of New Mexico, Albuquerque, New Mexico.

Calzia, J. P., and W. L. Hiss, 1978. "Igneous Rocks in Northern Delaware Basin, New Mexico and Texas," Geology and Mineral Deposits of Ochoan Rocks in Delaware Basin and Adjacent Areas, Circular 159, New Mexico Bureau of Mines and Mineral Resources, Socorro, New Mexico.

Campbell, J. E., P. C. Kaestner, B. S. Langkopf, and R. B. Lantz, 1979. Risk Methodology for Geologic Disposal of Radioactive Waste: The Network Flow and Transport (NWFT) Model, SAND79-1920, NUREG/CR-1190, Sandia National Laboratories, Albuquerque, New Mexico.

Campbell, J. E., D. E. Longsine, and R. M. Cranwell, 1981. Risk Methodology for Geologic Disposal of Radioactive Waste: The NWFT/DVM Computer Code User's Manual, SAND81-0886, NUREG/CR-2081, Sandia National Laboratories, Albuquerque, New Mexico.

Campbell, J. E., D. E. Longsine, and M. Reeves, 1986. Risk Methodology for Geologic Disposal of Radioactive Waste: The Distributed Velocity Method of Solving the Convective-Dispersion Equation, SAND80-0717, NUREG/CR-1376, Sandia National Laboratories, Albuquerque, New Mexico.

Caravella, F. J., and A. R. Sanford, 1977. An Analysis of Earthquakes North of the Los Medanos Site on July 26, 1972, and November 28, 1974, New Mexico Institute of Mining and Technology, Socorro, New Mexico.

Chaturvedi, Lokesh, 1984. "Occurrence of Gases in the Salado Formation."

Christensen, C. L., and T. O. Hunter, 1979. Waste Isolation Pilot Plant (WIPP) Borehole Plugging Program Description, January 1, 1979, SAND79-0640, Sandia National Laboratories, Albuquerque, New Mexico.

Chugg, J. C., et al., 1971. "Soil Survey of Eddy County Area, New Mexico," Photographs, U.S. Department of Agriculture 82:151.

Cooper, J. B., 1960. Geologic Section from Carlsbad Caverns National Park Through Project Gnome Site, Eddy and Lea Counties, New Mexico, U.S. Geological Survey Trace Element Investigation Report 767.

Cooper, J. B., 1961. Test Holes Drilled in Support of Groundwater Investigations, Project Gnome, Eddy County, New Mexico, basic data report, U.S. Geological Survey Trace Element Investigation Report 786.

Cooper, J. B., 1962. "Ground Water in Cenozoic Fill in Collapse Structures, Southeastern Eddy County, New Mexico," Short Papers in Geology, Hydrology, and Topography, U.S. Geological Survey Professional Paper 450-B, pp. E152-E153.

Cooper, J. B., 1962. Groundwater Investigations of the Project Gnome Area, Eddy and Lea Counties, New Mexico, U.S. Geological Survey Trace Element Investigation Report 802.

Cooper, J. B., et al., 1962. Hydrologic and Geologic Studies for Project Gnome, preliminary report, U.S. Geological Survey Project Gnome Report PNE-130P.

Core Laboratories, 1977. "Special Core Analysis Study," for Sipes, Williamson and Aycock, Inc., unpublished consulting report to Sandia National Laboratories, Albuquerque, New Mexico.

Cox, E. R., 1967. "Geology and Hydrology between Lake McMillan and Carlsbad Springs, Eddy County, New Mexico," U.S. Geological Survey Water-Supply Paper 1828.

Cys, J. M., 1978. "Transitional Nature and Significance of the Castile-Bell Canyon Contact," Geology and Mineral Deposits of Ochoan Rocks in Delaware Basin and Adjacent Areas, New Mexico Bureau of Mines and Mineral Resources Circular 159, pp. 53-56.

D'Appolonia Consulting Engineers, Inc., 1982. Data File Report: ERDA-6 and WIPP-12 Testing, prepared for Westinghouse Electric Co. and U.S. Department of Energy, Albuquerque, New Mexico, 7 Vols.

D'Appolonia Consulting Engineers, Inc., 1983. Data File Report: ERDA-6 and WIPP-12 Testing, Addendum, prepared for Westinghouse Electric Co. and U.S. Department of Energy, Albuquerque, New Mexico.

Davies, P. B., 1983. "Assessing the Potential for Deep-Seated Salt Dissolution and Subsidence at the Waste Isolation Pilot Plant (WIPP)," unpublished manuscript presented at the State of New Mexico Environmental Evaluation Group Conference on WIPP Site Suitability for Radioactive Waste Disposal, Carlsbad, New Mexico.

Davies, P. B., 1983. "Structural Characteristics of a Deep-Seated Dissolution-Subsidence Chimney in Bedded Salt," Sixth International Symposium on Salt, Vol. 1, pp. 331-350.

Dawson, P. R., and P. F. Chavez, 1978. COUPLEFLO - A Computer Code for Coupled Creeping Viscous Flow and Conductive-Convective Heat Transfer, Part I, Theoretical Background, SAND78-1406, Sandia National Laboratories, Albuquerque, New Mexico.

Dawson, P. R., and P. F. Chavez, 1978. COUPLEFLO - Computer Program for Coupled Creeping Viscous Flow and Conductive-Convective Heat Transfer, Part II, Users Manual, SAND78-1407, Sandia National Laboratories, Albuquerque, New Mexico.

Dean, W. E., Jr., 1967. "Petrologic and Geochemical Variations in the Permian Castile Varved Anhydrite, Delaware Basin, Texas and New Mexico," Ph.D. Dissertation, University Microfilms Number 68-3463.

Dean, W. E., and R. Y. Anderson, 1978. "Salinity Cycles: Evidence for Subaqueous Deposition of Castile Formation and Lower Part of Salado Formation, Delaware Basin, Texas and New Mexico," Geology and Mineral Deposits of Ochoan Rocks in Delaware Basin and

Adjacent Areas, New Mexico Bureau of Mines and Mineral Resources Circular 159, pp. 15-20.

Djeddi, R., 1979. "An Integrated Geophysical and Geological Study of the Deep Structure and Tectonics of the Permian Basin of West Texas and Southeastern New Mexico," Master's Thesis, University of Texas at El Paso.

Dobrin, M. B., P. L. Lawrence, and R. L. Sengbush, 1954. "Surface and Near-Surface Waves in the Delaware Basin," Geophysics, Vol. 19, pp. 695-715.

DOE (U.S. Department of Energy), 1983. Geotechnical Field Data Report No. 4 - Geologic Mapping and Water Inflow Testing in the SPDV Ventilation Shaft, Waste Isolation Pilot Plant, compiled for U.S. Department of Energy by TSC-D'Appolonia.

DOE (U.S. Department of Energy), 1983. Geotechnical Field Data Report No. 5 - Geologic Mapping of Access Drifts, Double Box Area, compiled for U.S. Department of Energy by TSC-D'Appolonia.

DOE (U.S. Department of Energy), 1983. Geotechnical Field Data Report No. 6 - Underground Excavation Field Instrumentation Data, compiled for U.S. Department of Energy by TSC-D'Appolonia.

DOE (U.S. Department of Energy), 1983. Geotechnical Field Data Report No. 7 - Geologic Mapping of Exploratory Drift.

DOE (U.S. Department of Energy), 1983. Geotechnical Field Data Report No. 9 - Logging of Vertical Coreholes "Double Box" Area and Exploratory Drift.

DOE (U.S. Department of Energy), 1983. Geotechnical Field Data Report No. 10 - Underground Excavation Field Instrumentation Data.

DOE (U.S. Department of Energy), 1983. Geotechnical Field Data Report, Part II, Gas Testing Data Update, WIPP/DOE-177.

DOE (U.S. Department of Energy), 1985. Geotechnical Field Data Report, DOE/WIPP-218.

DOE (U.S. Department of Energy), 1986. Interim Geotechnical Field Data Report, DOE/WIPP-86-012 (Bechtel National).

DOE (U.S. Department of Energy), 1987. Geotechnical Field Data and Analysis Report, Vol. 2, DOE/WIPP 87-017, WIPP Project Office, Carlsbad, New Mexico.

Drellack, S. L., Jr., 1979. "Geological Data," Basic Data Report for Drillhole WIPP-28, SAND79-0282, Sandia National Laboratories, Albuquerque, New Mexico, pp. 4-31, Fig. 2.

Drellack, S. L., Jr., and R. P. Snyder, 1979. "Geological Data," Basic Data Report for Drillhole WIPP-26, SAND79-0280, Sandia National Laboratories, Albuquerque, New Mexico, pp. 4-29, Fig. 2.

EEG (Environmental Evaluation Group), 1979. A Compilation of Site Slection Criteria, Considerations, and Concerns Appearing in the Literature on the Deep Geologic Disposal of Radioactive Wastes, D. Goad, Health and Environment Department, State of New Mexico, Santa Fe, New Mexico.

EEG (Environmental Evaluation Group), 1980. Geotechnical Considerations for Radiological Hazard Assessment of WIPP, A Report of a Meeting Held on January 17-18, 1980, EEG-6, Health and Environment Department, State of New Mexico, Santa Fe, New Mexico.

EEG (Environmental Evaluation Group), 1980. WIPP Site and Vicinity Geological Field Trip, A Field Trip to the Proposed Waste Isolation Pilot Plant Project in Southeastern New Mexico, June 16 to 18, 1980, EEG-7, L. Chaturvedi, Health and Environment Department, State of New Mexico, Santa Fe, New Mexico.

EEG (Environmental Evaluation Group), 1982. Analysis of the Potential Formation of a Breccia Chimney Beneath the WIPP Repository, EEG-13, P. Spiegler, Health and Environment Department, State of New Mexico, Santa Fe, New Mexico.

EEG (Environmental Evaluation Group), 1983. EEG Review Comments on the Geotechnical Reports Provided by DOE to EEG Under the Stipulated Agreement Through March 1, 1983, EEG-22, Health and Environment Department, State of New Mexico, Santa Fe, New Mexico.

EEG (Environmental Evaluation Group), 1983. Evaluation of the Suitability of the WIPP Site, EEG-23, R. H. Neill, J. K. Channell, L. Chaturvedi, M. S. Little, K. Rehfeldt, and P. Spiegler, Health and Environment Department, State of New Mexico, Santa Fe, New Mexico.

EEG (Environmental Evaluation Group), 1987. Exhaust Stack Monitoring Issues at the Waste Isolation Pilot Plant, EEG-37, J. C. Rodgers, Health and Environment Department, State of New Mexico, Santa Fe, New Mexico.

EEG (Environmental Evaluation Group), 1987. Post Burial Alteration of the Permian Rustler Formation Evaporites, WIPP Site, New Mexico: Textural, Stratigraphic and Chemical Evidence, EEG-36, T. K. Lowenstein, Health and Environment Department State of New Mexico, Santa Fe, New Mexico.

EEG (Environmental Evaluation Group), 1987. The Rustler Formation at the WIPP Site, EEG-34, L. Chatuverdi, Health and Environment Department, State of New Mexico, Santa Fe, New Mexico.

Elliot, C. L., 1976. An Experimental Detail Gravity Survey of Known or Suspected Breccia Pipes at Weaver Hill, Hills A and B, and Hills C and D, Eddy County, New Mexico, Elliot Geophysical Company report to Sandia National Laboratories, Albuquerque, New Mexico, for Waste Isolation Pilot Plant Program, Vol. 1.

Elliot, C. L., 1976. An Experimental Detail Resistivity Survey of Known or Suspected Breccia Pipes and Sinkholes, Eddy County, New Mexico, Elliot Geophysical Company report to Sandia National Laboratories, Albuquerque, New Mexico, for Waste Isolation Pilot Plant Program.

Elliot, C. L., 1976. A Preliminary Geophysical Study of a Trachyte Dike in Close Proximity to the Proposed Los Medanos Nuclear Waste Disposal Site, Eddy and Lea Counties, New Mexico, Elliot Geophysical Company, Tucson, Arizona.

Elliot, C. L., 1977. Evaluation of the Proposed Los Medanos Nuclear Waste Disposal Site by Means of Electrical Resistivity Surveys, Eddy and Lea Counties, New Mexico, Elliot Geophysical Company report to Sandia National Laboratories, Albuquerque, New Mexico, for Waste Isolation Pilot Plant Program, Vol. 1.

Elliot, C. L., 1977. Experimental Resistivity Soundings Near a Known Breccia Pipe, Weaver Area, Eddy County, New Mexico, report prepared for Sandia National Laboratories, Albuquerque, New Mexico.

Earth Technology Corp., 1987. Final Report for Time Domain Electromagnetic (TDEM) Surveys at the WIPP, SAND87-7144, Sandia National Laboratories, Albuquerque, New Mexico.

Ferrall, C. C., and J. F. Gibbons, 1980. Core Study of the Rustler Formation Over the WIPP Site, SAND79-7110, Sandia National Laboratories, Albuquerque, New Mexico.

Fournier, R. O., 1961. "Regular Interlayered Chlorite-Vermiculite in Evaporite of the Salado Formation, New Mexico," U.S. Geological Survey Professional Paper 424-D, pp. 323-327.

\Freeze, R. A., and J. A. Cherry, 1976. Groundwater, Prentice-Hall, Englewood Cliffs, New Jersey.

Gard, L. M., Jr., 1968. "Geologic Studies Project Gnome, Eddy County, New Mexico," U.S. Geological Survey Professional Paper 589.

Gera, F., 1974. On the Origin of the Small Hills in Nash Draw and Clayton Basin, Southeastern New Mexico, Oak Ridge National Laboratory, Report ORNL Central Files No. 74-2-29, Oak Ridge, Tennessee.

Gonzalez, D. D., 1983. Groundwater Flow in the Rustler Formation: Waste Isolation Pilot Plant (WIPP), Southeast New Mexico, SAND82-1012, Sandia National Laboratories, Albuquerque, New Mexico.

Gonzales, J. L., and C. L. Jones, 1979. Geological Data for Borehole WIPP-13, Basic Data Report for Drill Hole WIPP-1, SAND79-0273, Sandia National Laboratories, Albuquerque, New Mexico, pp. 4-15.

Gonzales, M. M., 1988. Compilation and Comparison of Test-Hole Location Surveys in the Vicinity of the Waste Isolation Pilot Plant (WIPP) Site, SAND-1065, Sandia National Laboratories, Albuquerque, New Mexico.

Griswold, G. B., 1977. Site Selection and Evaluation Studies of the Waste Isolation Pilot Plant (WIPP), Los Medanos, Eddy County, New Mexico, SAND77-0946, Sandia National Laboratories, Albuquerque, New Mexico.

Haigler, L. B., 1962. Geologic Notes on the Delaware Basin, Circular 63, New Mexico Bureau of Mines and Mineral Resources, Socorro, New Mexico.

Haigler, L. B., and R. R. Cunningham, 1972. "Structure Contour Map on Top of the Undifferentiated Silurian and Devonian Rocks in Southeastern New Mexico," U.S. Geological Survey Map OM-218.

Haug, A., V. A. Kelley, A. M. LaVenue, and J. F. Pickens, 1987. Modeling of Groundwater Flow in the Culebra Dolomite at the Waste Isolation Pilot Plant (WIPP) Site: Interim Report, SAND86-7167, Sandia National Laboratories, Albuquerque, New Mexico.

Hayes, P. T., 1964. "Geology of the Guadalupe Mountains, New Mexico," U.S. Geological Survey Professional Paper 446.

Hayes, P. T., and G. O. Bachman, 1979. Examination and Reevaluation of Evidence for the Barrera Fault, Guadalupe Mountains, New Mexico, U.S. Geological Survey Open-File Report 79-1520.

Hern, J. L., D. W. Powers, and L. J. Barrows, 1978. Seismic Reflection Data Report Waste Isolation Pilot Plant (WIPP) Site, Southeastern New Mexico, SAND79-0264, Sandia National Laboratories, Albuquerque, New Mexico, Vol. 2.

Hills, J. M., 1963. "Late Paleozoic Tectonics and Mountain Ranges, Western Texas to Southern Colorado," American Association of Petroleum Geologists Bulletin, Vol. 947, pp. 1709-1725.

Hills, J. M., 1976. "Late Paleozoic Structural Directions in Southern Permian Basin, West Texas and Southeastern New Mexico," American Association of Petroleum Geologists Bulletin, Vol. 54 (10), pp. 1809-1827.

Hills, J. M., 1979. "Delaware Basin Sedimentation Tectonism, and Hydrocarbon Generation, Guadalupian Delaware Mountain Group of West Texas and Southeast New Mexico," Society of Economic Paleontologists and Mineralogists, Permian Basin Section Publication, 79-18:1.

Hills, J. M., 1984. "Sedimentation, Tectonism, and Hydrocarbon Generation in Delaware Basin, West Texas and Southeastern New Mexico," American Association of Petroleum Geologists Bulletin, Vol. 68, pp. 250-267.

Hiss, W. L., 1976. "Structure of the Permian Capitan Aquifer, Southeast New Mexico and West Texas," Resource Map 6, New Mexico Bureau of Mines and Mineral Resources, Socorro, New Mexico.

Hiss, W. L., 1976. "Structure of the Permian Ochoan Rustler Formation, Southeast New Mexico and West Texas," Resource Map 7, New Mexico Bureau of Mines and Mineral Resources.

Holcomb, D. J., and M. Shields, 1987. Hydrostatic Creep Consolidation of Crushed Salt with Added Water, SAND87-1990, Sandia National Laboratories, Albuquerque, New Mexico.

Holt, R. M., and D. W. Powers, 1984. Geotechnical Activities in the Waste-Handling Shaft, WTSD-TME-038, U.S. Department of Energy, Carlsbad, New Mexico.

Holt, R., and D. Powers, 1986. Geotechnical Activities in the Exhaust Shaft, DOE/WIPP-86-008, Waste Isolation Pilot Plant Project, Carlsbad, New Mexico.

Holt, R. M., and D. W. Powers, 1987. Rustler Formation, Waste-Handling and Exhaust Shafts, Waste Isolation Pilot Plant (WIPP) Site, EEG-34, State of New Mexico Environmental Evaluation Group, Santa Fe, New Mexico.

Holt, R. M., and D. W. Powers, 1988. Facies Variability and Post-Depositional Alteration Within the Rustler Formation in the Vicinity of the Waste Isolation Pilot Plant, Southeastern New Mexico, DOE/WIPP-88-004, U.S. Department of Energy, Carlsbad, New Mexico.

Hoy, R. N., and G. W. Gross, 1982. A Baseline Study of Oxygen 18 and Deuterium in the Roswell, New Mexico, Groundwater Basin, Report No. 144, New Mexico Water Resources Research Institute.

Jarolimek, L., M. J. Timmer, and R. F. McKinney, 1983. Geotechnical Activities in the Exploratory Shaft - Selection of the Facility Interval.

Jarolimek, L., M. J. Timmer, and D. W. Powers, 1983. Correlation of Drillhole and Shaft Logs, Waste Isolation Pilot Plant (WIPP) Project, Southeastern New Mexico, TME-3179, U.S. Department of Energy, Albuquerque, New Mexico.

Johnson, K. S., 1981. "Dissolution of Salt on the East Flank of the Permian Basin in the Southwestern USA," Journal of Hydrology Vol. 54, pp. 75-93.

Johnson, K. S., and S. Gonzales, 1978. Salt Deposits in the United States and Regional Geologic Characteristics Important for Storage of Radioactive Waste, Y/OWI/SUB-7414/1, Office of Waste Isolation, Union Carbide Corporation, Oak Ridge, Tennessee.

Jones, C. L., 1960. Thickness, Character and Structures of Upper Permian Evaporites in Part of Eddy County, New Mexico, TEM-1033, U.S. Geological Survey.

Jones, C. L., 1981. Geologic Data for Borehole ERDA-6, Eddy County, New Mexico, U.S. Geological Survey Open-File Report 81-468.

Jones, C. L., 1981. Geologic Data for Borehole ERDA-9, Eddy County, New Mexico, U.S. Geological Survey Open-File Report 81-469.

Jones, C. L., C. G. Bowles, and K. G. Bell, 1960. Experimental Drill Hole Logging in Potash Deposits of the Carlsbad District, New Mexico, U.S. Geological Survey Open-File Report 60-84.

Jones, C. L., M. E. Cooley, and G. O. Bachman, 1973. Salt Deposits of Los Medanos Area, Eddy and Lea Counties, New Mexico, sections on "Groundwater Hydrology," and "Surficial Geology," U.S. Geological Open-File Report 4339-7.

Jones, C. L., and J. L. Gonzales, 1980. Basic Data Report for Drillhole WIPP-19, SAND79-0276, Sandia National Laboratories and U.S. Geological Survey.

Jones, C. L., and B. M. Madsen, 1959. "Observations on Igneous Intrusions in Late Permian Evaporites, Southeastern New Mexico," Geological Society of America Bulletin, Vol. 70, pp. 1625-1626.

Jones, C. L., and A. F. McIntyre, 1979. Geological Data, Basic Data Report for Drillhole WIPP-27, SAND79-0281, Sandia National Laboratories, Albuquerque, New Mexico, pp. 5-18.

Katsube, T. J., T. W. Melnyk, and J. P. Hume, 1986. "Pore Structure from Diffusion in Granitic Rocks," Atomic Energy of Canada Ltd., Technical Record 381, p. 28.

Kelley, V. C., 1971. Geology of the Pecos Country, Southeastern New Mexico, Memoir 24, New Mexico Bureau of Mines and Mineral Resources, Socorro, New Mexico.

King, P. B., 1948. Geology of the Southern Guadalupe Mountains, Texas, U.S. Geological Survey Professional Paper 215.

Kirkland, D. W., and R. Y. Anderson, 1970. "Microfolding in the Castile and Todilto Evaporite, Texas and New Mexico," Geological Society of America Bulletin, Vol. 81, pp. 3259-3282.

Lallemand-Barres and P. Peaudecerf, 1978. "Recherche des Relations Entre les Valeurs Mesurées de la Dispersivité Macroscropique d'un Milieu Aquifère, ses Autres Charactéristiques et les Conditions de Mesure," Rech. Bull. Bur. Geo. Min., Ser. 2, Sec.III, 4-1978, pp. 277-284.

Lambert, S. J., 1980. Recommended Analysis Plan for the Borehole Plugging Program Potash Core Test, SAND80-0812, Sandia National Laboratories, Albuquerque, New Mexico.

Lambert, S. J., 1982. Interim Report - Dissolution of Evaporites In and Around the Delaware Basin, Southeastern New Mexico and West Texas, SAND82-01461, Sandia National Laboratories Report, Albuquerque, New Mexico.

Lambert, S. J., 1983. Dissolution of Evaporites In and Around the Delaware Basin, Southeastern New Mexico, SAND82-01, Sandia National Laboratories Report, Albuquerque, New Mexico.

Lambert, S. J., 1983. Dissolution of Evaporites In and Around the Delaware Basin, Southeastern New Mexico and West Texas, SAND82-0461, Sandia National Laboratories, Albuquerque, New Mexico.

Lambert, S. J., 1983. "Evaporite Dissolution Relevant to the WIPP Site, Northern Delaware Basin, Southeastern New Mexico," Material Resources Society Symposium Proceedings, Vol. 15, pp. 291-298.

Lambert, S. J., and J. A. Carter, 1984. Uranium-Isotope Disequilibrium in Brine Reservoirs of the Castile Formation, Northern Delaware Basin, Southeastern New Mexico, I: Principles and Methods, SAND83-0144, Sandia National Laboratories, Albuquerque, New Mexico.

Lang, W. B., 1938. Geology of the Pecos River Between Laguna Grande de la Sal and Pierce Canyon, 12th and 13th Biennial Reports, New Mexico State Engineer, pp. 80-86.

Lappin, A. R., 1988. Summary of Site-Characterization Studies Conducted from 1983 through 1987 at the Waste Isolation Pilot Plant (WIPP), Southeastern New Mexico, SAND88-0157, Sandia National Laboratories, Albuquerque, New Mexico.

Long, G. J., and Associates, 1976. Interpretation of Geophysical Data, Los Medanos and Vicinity, Lea and Eddy Counties, New Mexico, prepared for Sandia National Laboratories, Albuquerque, New Mexico.

Lowenstein, T. K., 1987. Post-Burial Alteration of the Permian Rustler-Formation Evaporites, WIPP Site, New Mexico, EEG-36, State of New Mexico Environmental Evaluation Group, Santa Fe, New Mexico.

Maley, V. C., and R. M. Huffington, 1953. "Cenozoic Fill and Evaporite Solution in Delaware Basin, Texas, and New Mexico," Geological Society of America Bulletin, Vol. 64, No. 5, pp. 539-546.

Mansure, A., and M. Reiter, 1977. An Accurate Equilibrium Temperature Log in AEC #8, A Drill Test in the Vicinity of the Carlsbad Disposal Site, New Mexico Bureau of Mines and Mineral Resources Open-File Report #80.

McGlasson, E. H., 1968. "The Siluro-Devonian of West Texas and Southeast New Mexico," West Texas Geological Society-Delaware Basin Exploration Guidebook, Pub. 68-55, pp. 35-44.

Mercer, J. W., R. L. Beauheim, R. P. Snyder, and G. M. Fairer, 1986. Basic Data Report for Drillhole DOE-2 Drilling and Hydrologic Testing, SAND86-0611, Sandia National Laboratories, Albuquerque, New Mexico.

Mercer, J. W., R. L. Beauheim, R. P. Snyder, and G. M. Fairer, 1987. Basic Data Report for Drilling and Hydrologic Testing of Drillhole DOE-2 at the Waste Isolation Pilot Plant (WIPP) Site, SAND86-0611, Sandia National Laboratories, Albuquerque, New Mexico.

Meyer, R. F., 1966. Geology of Pennsylvania and Wolfcampian Rocks in Southeast New Mexico, Memoir 17, New Mexico Bureau of Mines and Mineral Resources, Socorro, New Mexico.

Morgan, A. M., 1941. "Solution-Phenomena in the Pecos Basin of New Mexico," Transcripts of the American Geophysics Union, pp. 27-35.

Motts, W. S., 1957. "Geology and Groundwater Resources of the Carlsbad Area, New Mexico", unpublished Ph.D. Thesis, University of Illinois, Urbana, Illinois.

Munson, D. E., A. F. Fossum, and P. E. Senseny, 1989. "Approach to First Principles Model Prediction of Measured WIPP In Situ Room Closure in Salt," Proceedings of the 30th U.S. Rock Mechanics Symposium, at the University of West Virginia, June 19-22, to be published by A. A. Balkema, Rotterdam, The Netherlands.

Neretnieks, I., and A. Rasmuson, 1984. "An Approach to Modeling Radionuclide Migration in a Medium with Strongly Varying Velocity and Block Sizes Along the Flow Path," Water Resources Research, Vol. 20, No. 12, pp. 1823-1836.

Nicholson, A., Jr., and A. Clebsch, Jr., 1961. Geology and Groundwater Conditions in Southern Lea County, New Mexico: Groundwater Report 6, U.S. Geological Survey and State Bureau of Mines and Mineral Resources, New Mexico.

NOAA (U.S. National Oceanic and Atmospheric Administration) and USGS (U.S. Geological Survey), United States Earthquakes, published jointly (annually since 1928).

Olive, W. W., 1957. "Solution-Subsidence Troughs Castile Formation of Gypsum Plain, Texas and New Mexico," Geological Society of America Bulletin, Vol. 68, pp. 351-358.

O'Neil, J. R., C. M. Johnson, L. D. White, and E. Roedder, 1986. "The Origin of Fluids in the Salt Beds of the Delaware Basin, New Mexico and Texas," Journal of Applied Geochemistry, Vol. 1, No. 2, pp. 265-271.

Peterson, E. W., et al, 1981. Analysis of Bell Canyon Test Results, S-CUBED Document No. SSS-R-80-4474-2.

Peterson, E. W., P. L. Langus, R. D. Broce, and K. Lie, 1981. In Situ Permeability Study of Rock Salt, SAND81-7073, Sandia National Laboratories, Albuquerque, New Mexico.

Peterson, E. W., P. L. Langus, R. D. Broce, and J. C. Stormont, 1986. Permeability Measurements at the Waste Isolation Pilot Plant (WIPP), paper presented at Waste Management 1986, Tucson, Arizona.

Pettigrew, R. R., and Associates, 1977. WIPP Conceptual Design Report (Soils and Foundation Investigation), prepared for Sandia National Laboratories, Albuquerque, New Mexico.

Pfeifer, M. C., 1987. "Multicomponent Underground DC Resistivity Study at the Waste Isolation Pilot Plant, Southeast New Mexico," Master's Thesis T-3372, Colorado School of Mines, Golden, Colorado.

Pierce, W. G., and E. I. Rich, 1962. "Summary of Rock Salt Deposits in the United States as Possible Storage Sites for Radioactive Waste Materials," U.S. Geological Survey Bulletin 1148.

Piper, A. M., 1973. Subrosion In and About the Four-Township Study Area Near Carlsbad, New Mexico, ORNL Subcontract 3745, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Powers, D. W., 1981. "Geologic Investigations of the WIPP Site--Overview and Issues," Environmental Geology and Hydrology in New Mexico, New Mexico Geological Society Special Publication No. 10, pp. 119-122.

Powers, D. W., and B. W. Hassinger, 1984. "Synsedimentary Dissolution Pits in Halite of the Salado Formation, Southeastern New Mexico," submitted to Journal of Sedimentary Petrology.

Powers, D. W., and R. M. Holt, 1984. "Depositional Environments and Dissolution in the Rustler Formation (Permian), Southeastern New Mexico," Abstract, Geologic Society of America, 97th Annual Meeting Abstracts with Program, Vol. 16, No. 6, p. 627.

Powers, D. W., S. J. Lambert, S. E. Shaffer, L. R. Hill, and W. D. Weart, eds., 1978. Geological Characterization Report, Waste Isolation Pilot Plant (WIPP) Site, Southeastern New Mexico, SAND78-1596, Vols. 1 and 2, Sandia National Laboratories, Albuquerque, New Mexico.

Pratt, W. E., 1954. "Evidences of Igneous Activity in the Northwestern Part of the Delaware Basin," New Mexico Geological Society Guidebook of Southeastern New Mexico Fifth Field Conference, pp. 143-147.

Pray, L. C., 1953. "Upper Ordovician and Silurian Stratigraphy of the Sacramento Mountains, Otero County, New Mexico," American Association of Petroleum Geologists Bulletin, Vol. 37, No. 8, pp. 1894-1918.

Pray, L. C., 1954. "Outline of the Stratigraphy and Structure of the Sacremento Mountain Escarpment," Fifth Field Conference, Southeastern New Mexico, New Mexico Geological Society, pp. 92-106.

Pray, L. C., 1959. "Stratigraphic and Structural Features of the Sacremento Mountain Escarpment, New Mexico," Guidebook for a Joint Field Conference in the Sacramento Mountains of Otero County, New Mexico, Permian Basin Section of Economic Paleontologists and Mineralogists and Roswell Geological Society, Alamogordo, New Mexico.

Pray, L. C., 1961. Geology of the Sacramento Mountains Escarpment, Otero County, New Mexico, New Mexico Bureau of Mines and Mineral Resources Bulletin 35, pp. 1-133.

Reddy, G. R., 1961. "Geology of the Queen Lake Domes Near Malaga, Eddy County, New Mexico," Master's Thesis, University of New Mexico, Albuquerque, New Mexico.

Reddy, R. S., 1966. "Crustal Structure in New Mexico," Master's Thesis, New Mexico Institute of Mining and Technology, Socorro, New Mexico, pp. 1-45.

Reeves, C. C., Jr., 1963. "Subterranean Natural Brines Produce Sodium Sulphate in West Texas," Groundwater, Vol. 1, pp. 35-36.

Register, J. K., 1981. Brine Pocket Occurrences in the Castile Formation, Southeastern New Mexico, U.S. Department of Energy TME-3080.

Robinson, T. W., and W. G. Lang, 1938. Geology and Groundwater Conditions of the Pecos River Valley in the Vicinity of Laguna Grande de la Sal, New Mexico, with Special Reference to the Salt Content of the River Water, 12th and 13th Biennial Reports, New Mexico State Engineer, pp. 77-100.

Roedder, E., and H. E. Belkin, 1978. Fluid Inclusions in Core Samples from ERDA No. 9 Borehole, WIPP Site, New Mexico, Status Report 2, prepared for Sandia National Laboratories, Albuquerque, New Mexico.

Rogers, A. M., and A. Malkiel, 1978. "A Study of Earthquakes in the Permian Basin, Texas - New Mexico," draft of a U.S. Geological Survey Report for Open-File.

Ross, A. C., 1986. "Paleozoic Evolution of Southern Margin of Permian Basin," Geological Society of America Bulletin, Vol. 97, No. 5, pp. 536-554.

Roswell Geological Society, 1958. North-South Stratigraphic Cross-Section: Delaware Basin-Northwest Shelf.

Reynolds, C. B., and Associates, 1976. Summary and Final Report, Shallow Seismic Reflection Survey, Phases I and II, Los Medanos Area, Eddy County, New Mexico, report submitted to Sandia National Laboratories, Albuquerque, New Mexico.

Sandia National Laboratories, and U.S. Geological Survey, 1979. Basic Data Report for Drillhole WIPP-13, SAND79-0273, Albuquerque, New Mexico.

Sandia National Laboratories, and U.S. Geological Survey, 1979. Basic Data Report for Drillhole WIPP-25, SAND79-0279, Albuquerque, New Mexico.

Sandia National Laboratories, and U.S. Geological Survey, 1979. Basic Data Report for Drillhole WIPP-26, SAND78-0280, Albuquerque, New Mexico.

Sandia National Laboratories, and U.S. Geological Survey, 1979. Basic Data Report for Drillhole WIPP-27, SAND79-0281, Albuquerque, New Mexico.

Sandia National Laboratories, and U.S. Geological Survey, 1979. Basic Data Report for Drillhole WIPP-28, SAND79-0282, Albuquerque, New Mexico.

Sandia National Laboratories, and U.S. Geological Survey, 1979. Basic Data Report for Drillhole WIPP-29, SAND79-0283, Albuquerque, New Mexico.

Sandia National Laboratories, and U.S. Geological Survey, 1980. Basic Data Report for Drillhole WIPP-18, SAND79-0275, Albuquerque, New Mexico

Sandia National Laboratories, and U.S. Geological Survey, compilers, 1980. Basic Data Report for Drillhole WIPP-19, SAND79-0276, Albuquerque, New Mexico.

Sandia National Laboratories, and U.S. Geological Survey, 1980. Basic Data Report for Drillhole WIPP-21, SAND79-0277, Albuquerque, New Mexico.

Sandia National Laboratories, and U.S. Geological Survey, 1980. Basic Data Report for Drillhole WIPP-22, SAND79-0278, Albuquerque, New Mexico.

Sandia National Laboratories, and U.S. Geological Society, 1980. Basic Data Report for Drillhole WIPP-30, SAND79-0284, Albuquerque, New Mexico.

Sandia National Laboratories, and U.S. Geological Society, 1980. Basic Data Report for Drillhole WIPP-32, SAND80-1102, Albuquerque, New Mexico.

Sandia National Laboratories, and University of New Mexico, 1981. Basic Data Report for

Drillhole WIPP-15, SAND 79-0274, Albuquerque, New Mexico.

Sandia National Laboratories, and U.S. Geological Survey, 1981. Basic Data Report for Drillhole WIPP-33, SAND80-2011, Albuquerque, New Mexico.

Sandia National Laboratories, and U.S. Geological Survey, 1981. Basic Data Report for Drillhole WIPP-34, SAND81-2643, Albuquerque, New Mexico.

Sandia National Laboratories, and D'Appolonia Consulting Engineers, 1982. Basic Data Report for Drillhole AEC-7 WIPP, SAND79-0268, Albuquerque, New Mexico.

Sandia National Laboratories, and U.S. Geological Survey, 1982. Basic Data Report for Drillhole WIPP-11, SAND79-0272, Albuquerque, New Mexico.

Sandia National Laboratories, and D'Appolonia Consulting Engineers, 1982. Basic Data Report for Drillhole WIPP-12, SAND82-2336, Albuquerque, New Mexico.

Sandia National Laboratories, and D'Appolonia Consulting Engineers, 1982. Basic Data Report for Deepening of Drillhole WIPP-13, SAND82-1880, Albuquerque, New Mexico.

Sandia National Laboratories, and D'Appolonia Consulting Engineers, 1982. Basic Data Report for Drillhole WIPP-14, SAND82-1783, Albuquerque, New Mexico.

Sandia National Laboratories, and D'Appolonia Consulting Engineers, 1983. Basic Data Report for Drillhole AEC - WIPP, SAND79-0269, Albuquerque, New Mexico.

Sandia National Laboratories, and U.S. Geological Survey, 1983. Basic Data Report for Drillhole ERDA-6 - WIPP, SAND79-0267, Albuquerque, New Mexico.

Sandia National Laboratories, and U.S. Geological Survey, 1983. Basic Data Report for Drillhole ERDA-9 - WIPP, SAND79-0270, Albuquerque, New Mexico.

Sandia National Laboratories, and U.S. Geological Survey, 1983. Basic Data Report for Drillhole ERDA-10 - WIPP, SAND79-0271, Albuquerque, New Mexico.

Sanford, A. R., 1965. An Instrumental Study of New Mexico Earthquakes, New Mexico Bureau of Mines and Mineral Resources Circular 78, pp. 1-12.

Sanford, A. R., S. J. Johnson, F. Caravella, and R. Ward, 1976. A Report on Seismic Recording at the Los Medanos Area in Southeastern New Mexico, 1974-1975, New Mexico Institute of Mining and Technology, Socorro, New Mexico.

Sanford, A. R., S. Sandford, F. Caravella, L. Merritt, J. Sheldon, and R. Ward, 1978. A Report on Seismic Studies of the Los Medanos Area in Southeastern New Mexico, New Mexico Institute of Mining and Technology, interim report prepared for Sandia National Laboratories, Albuquerque, New Mexico.

Sanford, A. R., and T. R. Toppozada, 1974. Seismicity of Proposed Radioactive Waste Disposal Site in Southeastern New Mexico, New Mexico Bureau of Mines and Mineral Resources Circular 143, Socorro, New Mexico.

Sares, S. W., and S. G. Wells, 1986. "Geomorphic and Hydrogeologic Development of the Gypsum Plain Karst, Delaware Basin, New Mexico," National Speleological Society Geology Field Trip Guidebook.

Schreiber, J. R., Jr., 1981. A Thin Section Petrographic Study of Rustler Formation Dolomites from the WIPP Site, Eddy County, New Mexico, Hydro Geo Chem., Inc., Tech Report No. 5 as a contractor report to Sandia National Laboratories, Albuquerque, New Mexico.

Semmes, D. R., 1920. "Notes on the Tertiary Intrusives of the Lower Pecos Valley, New Mexico," American Journal of Science, 4th Series, Vol. 50, pp. 415-430.

Seward, P. D., 1982. Abridged Borehole Histories for the Waste Isolation Pilot Plant (WIPP) Studies, SAND82-0080, Sandia National Laboratories, Albuquerque, New Mexico.

Shaffer, S. E., 1982. Bibliography of Geology of Southeastern New Mexico, SAND82-0770, Sandia National Laboratories, Albuquerque, New Mexico.

Shaffer, S. E., and D. W. Powers, 1977. Survey of Background Geology for WIPP Project in Southeastern New Mexico, SAND77-0448, Sandia National Laboratories, Albuquerque, New Mexico.

Shell Exploration and Production Research Center, 1973. "Potassium-Argon Dates on Permian Potash Minerals from Southeastern New Mexico," Isochron/West, No. 6, p. 37.

Silver, B. A., and R. G. Todd, 1969. "Permian Cyclic Strata, Northern Midland and Delaware Basins, West Texas and Southeast New Mexico," American Association of Petroleum Geologists Bulletin, 1966, Vol. 53, No. 11, pp. 2223-2251.

Snider, H. I., 1966. "Stratigraphy and Associated Tectonics of the Upper Permian Castile-Salado-Rustler Evaporite Complex, Delaware Basin, West Texas and Southeast New Mexico," Ph.D. Dissertation, University of New Mexico, Albuquerque, New Mexico.

Snyder, R. P., 1985. Dissolution of Halite and Gypsum, and Hydration of Anhydrite to Gypsum, Rustler Formation, in the Vicinity of the Waste Isolation Pilot Plant, Southeastern New Mexico, U.S. Geological Survey Open-File Report 85-229, Denver, Colorado.

Snyder, R. P., and L. M. Gard, Jr., 1982. Evaluation of Breccia Pipes in Southeastern New Mexico and Their Relation to the Waste Isolation Pilot Plant (WIPP) Site (with a section on drill-stem tests, WIPP-31, by J. W. Mercer), U.S. Geological Survey Open-File Report 82-968.

Spiegler, P., 1982. Hydrologic Analyses of Two Brine Encounters in the Vicinity of the Waste Isolation Pilot Plant (WIPP) Site.

Spiegler, P., 1983. Origin of the Brines Near WIPP from the Drill Holes ERDA-6 and WIPP-12 Based on Stable Isotope Concentrations of Hydrogen and Oxygen.

Stewart, S. W., and L. C. Pakiser, 1962. "Crustal Structure in Eastern New Mexico Interpreted from the Gnome Explosion," Seismological Society of American Bulletin, Vol. 52, pp.

1017-1030.

Stipp, T. F., 1960. *Major Structural Features and Geologic History of Southeastern New Mexico*, A Symposium of the Gas and Oil Fields of Southeastern New Mexico, Roswell Geological Society.

Swenson, F. A., 1973. *Dissolved Salts in Surface Water*, U.S. Geological Survey Open-File Report 73-14.

Szabo, B. J., W. C. Gottschall, J. N. Rosholt, and C. R. McKinney, 1980. *Part I: A Preliminary Study of Uranium-Thorium Systematics in Dissolution Residues at the Top of Salado Evaporites--Implications to Process and Time of the Last Vertical Dissolution Cycle of Salt Beds Near the WIPP Site; Part II: Uranium Trend Dating of Surficial Deposits and Gypsum Spring Deposit Near WIPP Site, New Mexico*, U.S. Geological Survey Open-File Report 80-879.

Teufel, L. W., 1981. *Mechanical Properties of Anhydrite and Polyhalite in Quasi-Static Triaxial Compression*, SAND81-0858, Sandia National Laboratories, Albuquerque, New Mexico.

Toppozada, T. R., and A. R. Sanford, 1976. "Crustal Structure in Central New Mexico Interpreted from the Gasbuggy Explosion," *Seismological Society of American Bulletin*, Vol. 66, pp. 877-886.

Unterberger, R. R., 1981. *Subsurface Radar Applications in the Delaware Basin*, SAND81-7153, Sandia National Laboratories, Albuquerque, New Mexico.

U.S. Geological Survey, 1973. "Aeromagnetic Map of the Carlsbad Area, New Mexico and Texas," U.S. Geological Survey Geophysical Investigations Map GP-861.

Van Devender, T. R., 1980. "Holocene Plant Remains from Rocky Arroyo and Last Chance Canyon, Eddy County, New Mexico," *The Southwestern Naturalist*, Vol. 25.

Vine, J. D., 1960. "Recent Domal Structures in Southeastern New Mexico," *American Association of Petroleum Geologists Bulletin*, Vol. 44, No. 12, pp. 1903-1911.

Vine, J. D., 1963. "Surface Geology of the Nash Draw Quadrangle, Eddy County, New Mexico," *U.S. Geological Survey Bulletin 1141-B*, Government Printing Office, Washington, D.C.

Vine, J. D., 1976. "Breccia Pipes and Burial Metamorphism in Permian Evaporites of the Delaware Basin New Mexico," *Geological Society of American Abstracts*, Vol. 8, No. 6.

Walter, J. C., Jr., 1953. "Paleontology of the Rustler Formation, Culberson County, Texas," *Journal of Paleontology*, Vol. 27, No. 5, pp. 279-702.

Watson, W. G., 1979. "Inhomogeneities of the Ramsey Member of the Permian Bell Canyon Formation, Geraldine Ford Field, Culberson and Reeves Counties, Texas, Guadalupian Delaware Mountain Group of West Texas and Southeast New Mexico," *Permian Basin Society Section of Economic Paleontologists and Mineralogists Publication*, 79-18, pp. 2-38.

Wawersik, W. R., and C. M. Stone, 1986. Experience with Hydraulic Fracturing Test for Stress Measurements in WIPP, SAND85-2273C, Sandia National Laboratories, Albuquerque, New Mexico.

West, R. E., and W. G. Wieduwilt, 1976. Experimental Geophysical Survey of the Los Medanos Site and Surrounding Areas for Waste Isolation Pilot Plant Program, Eddy and Lea Counties, New Mexico, Project 0611, Mining Geophysical Surveys report to Sandia National Laboratories, Albuquerque, New Mexico.

Widdicombe, R., 1980. "Quaternary Surficial Features in the Delaware Basin, Eddy and Lea Counties, New Mexico," unpublished Master's Thesis, University of New Mexico, Albuquerque, New Mexico.

Wood, B. G., R. E. Snow, D. J. Cosler, and S. Haji-Djafari, 1982. Delaware Mountain Group (DMG) Hydrology--Salt Removal Potential, DOE/TME-3166.

J.6 **HYDROLOGY**

Abitz, R. J., J. Myers, and P. Drez, 1989. "Composition of Salado Formation Brines from the WIPP Repository Horizon," DOE-WIPP Publication, in review.

Allison, G. B., W. J. Stone, and M. W. Hughes, 1985. "Recharge in Karst and Dune Elements of a Semi-Arid Landscape as Indicated by Natural Isotopes and Chloride," Journal of Hydrology, Vol. 76, pp. 1-25.

Anderson, R. Y., 1982. Upper Castile Brine Aquifer, Northern Delaware Basin, New Mexico.

Antevs, E., 1954. "Climate of New Mexico During the Last Glacio-Pluvial," Journal of Geology, Vol. 62, pp. 182-191.

Barker, J.A., and S.S.D. Foster, 1981. "A Diffusion Exchange Model for Solute Movement in Fissured Porous Rock," Quarterly Journal of Engineering Geology, London, Vol. 14, pp. 17-26.

Barr, G. E., and D. D. Gonzalez, 1980. "Current Status of the Hydrogeology for the WIPP," SAND80-1539C, in Proceedings of ONWI/LBL Workshop on Thermomechanical-Hydrochemical Modeling for a Hardrock Waste Repository, LBL-1120, pp. 89-94.

Barr, G. E., W. B. Miller, and D. D. Gonzalez, 1983. Interim Report on the Modeling of the Regional Hydraulics of the Rustler Formation, SAND83-0391, Sandia National Laboratories, Albuquerque, New Mexico.

Barrows, L. J., 1982. "WIPP Geohydrology--The Implications of Karst," unpublished manuscript reproduced in Appendix I of this report.

Barrows, L. J., 1982. "WIPP Geohydrology--The Implications of Karst," in Evaluation of the Suitability of the WIPP Site, eds. R. H. Neill, J. K. Channell, L. Chaturvedi, M. S. Little, K. Rehfeldt, and P. Spiegler, 1983, EEG-23, New Mexico Environmental Evaluation Group, Santa Fe, New Mexico.

Basler, J. A, 1983. Instrumentation Used for Hydraulic Testing of Potential Water-Bearing Formations at the Waste Isolation Pilot Plant Site in Southeastern New Mexico, U.S. Geological Survey Open-File Report 83-144, 29 pp.

Bear, J., 1972. Dynamics of Fluids in Porous Media, American Elsevier Publishing Company, New York.

Beauheim, R. L., 1985. Effects of Borehole History on Well-Test Analysis, SAND85-2008A, Sandia National Laboratories, Albuquerque, New Mexico, presented at 1985 AGU Fall Meeting.

Beauheim, R. L., 1986. Hydraulic-Test Results for Well DOE-2 at the Waste Isolation Pilot Plant (WIPP) Site, SAND86-1364, Sandia National Laboratories, Albuquerque, New

Mexico.

Beauheim, R. L., 1987. Analysis of Pumping Tests of the Culebra Dolomite Conducted at the H-3 Hydropad at the Waste Isolation Pilot Plant (WIPP) Site, SAND86-2311, Sandia National Laboratories, Albuquerque, New Mexico.

Beauheim, R. L., 1987. Interpretations of Single-Well Hydraulic Tests Conducted At and Near the Waste Isolation Pilot Plant (WIPP) Site, 1983-1987, SAND87-0039, Sandia National Laboratories, Albuquerque, New Mexico.

Beauheim, R. L., 1987. Interpretation of the WIPP-13 Multipad Pumping Test of the Culebra Dolomite at the Waste Isolation Pilot Plant (WIPP) Site, SAND87-2456, Sandia National Laboratories, Albuquerque, New Mexico.

Beauheim, R. L., B. W. Hassinger, and J. A. Klaiber, 1983. Basic Data Report for Borehole Cabin Baby-1 Deepening and Hydrologic Testing, Waste Isolation Pilot Plant (WIPP) Project, Southeastern New Mexico, WTSD-TME-020, U.S. Department of Energy, Albuquerque, New Mexico.

Bentley, H. W., 1983. "Development and Features of Groundwater Tracers at the Waste Isolation Pilot Plant (WIPP), New Mexico," Transactions, AGU Fall Meeting, Hydrology Section, p. 714.

Bentley, H. W., J. J. Ward, G. R. Walter, S. J. Alweis, and D. D. Gonzalez, 1981. "H-6 Convergent Tracer Test, the WIPP, Southeast New Mexico," (in preparation).

Bjorklund, L. J., and W. S. Motts, 1959. Geology and Water-Resources of the Carlsbad Area, New Mexico, U.S. Geological Survey Open-File Report 59-9.

Bredehoeft, J. D., 1988. "Will Salt Repositories Be Dry?" EOS, Transactions, American Geophysical Union, 69, No. 9, March 1, p. 121.

Brokaw, A. L., C. L. Jones, M. E. Cooley, and W. H. Hays, 1972. Geology and Hydrology of the Carlsbad Potash Area, Eddy and Lea Counties, New Mexico, U.S. Geological Survey, Open-File Report 4339-1.

Buck, A. D., J. E. Rhoderick, J. P. Burkes, K. Mather, R. E. Reinhold, and J. A. Boa, 1983. Modification of Bell Canyon Test (BCT) 1-FF Grout, SAND83-7097, Sandia National Laboratories, Albuquerque, New Mexico.

Bureau of Land Management (BLM), 1978. Groundwater Study for the Proposed Expansion of Potash Mining Near Carlsbad, New Mexico.

Canfield, T. R., 1983. Stress Near Waste Canisters Buried in Salt, SAND82-0525, Sandia National Laboratories, Albuquerque, New Mexico.

Carter, R. D., and C. W. Tracy, 1960. "An Improved Method for Calculating Water Influx," Transactions of the Society of Petroleum Engineers, American Institute of Mining Engineers, Vol. 219, pp. 515-517.

Chaturvedi, L., and J. K. Channell, 1985. The Rustler Formation as a Transport Medium for

Contaminated Groundwater, EEG-32, State of New Mexico Environmental Evaluation Group, Santa Fe, New Mexico.

Chaturvedi, L., and K. Rehfeldt, 1984. "Groundwater Occurrence and the Dissolution of Salt at the WIPP Radioactive Waste Repository Site," EOS, American Geophysical Union Transactions, Vol. 65, No. 31, pp. 457-59.

Christensen, C. L., and E. W. Peterson, 1981. The Bell Canyon Test Summary Report, SAND80-1375, Sandia National Laboratories, Albuquerque, New Mexico.

Cole, C. R., and F. W. Bond, 1980. Assessment of the Effectiveness of Geologic Isolation Systems, Comparison of INTERA and WISAP Model Application, Pacific Northwest Laboratory, PNL-3070.

Colton, I. D., and J. G. Morse, 1985. Water Quality Sampling Plan, DOE/WIPP-215, Waste Isolation Pilot Plant, Carlsbad, New Mexico.

Cooper, J. B., 1962. Groundwater in Cenozoic Fill in Collapse Structures, Southeastern Eddy County, New Mexico, U.S. Geological Survey Professional Paper 450-E, Art 225.

Cooper, J. B., 1962. Groundwater Investigations of the Project Gnome Area, Eddy and Lea Counties, New Mexico, U.S. Geological Survey TEI-802.

Cooper, J. B., and V. M. Glanzman, 1971. Geohydrology of Project Gnome Site, Eddy County, New Mexico, U.S. Geological Survey Professional Paper 712-A.

Cox, E. R., and J. S. Havens, 1965. A Progress Report on the Malaga Bend Experimental Salinity Alleviation Project, Eddy County, New Mexico, U.S. Geological Survey Open-File Report 65-35.

Cox, E. R., and J. L. Kunkler, 1962. Feasibility of Injecting Brine from Malaga Bend into the Delaware Mountain Group, Eddy County, New Mexico, U.S. Geological Survey Open-File Report.

Crawley, M. E., 1985. Hydrostatic Pressure Fluid Density Distribution of the Culebra Dolomite Member of the Rustler Formation Near the Waste Isolation Pilot Plant Site, Southeastern New Mexico, U.S. Department of Energy, Carlsbad, New Mexico.

Crawley, M. E., 1987. Initial Data Release Report for the Pressure-Density Survey Program, WID, WS:87:00002.

Crawley, M. E., 1987. Pressure-Density Survey Program Plan, Waste Isolation Pilot Plant, Carlsbad, New Mexico, International Technology Corporation.

Crawley, M. E., 1987. Second Data Release Report for the Pressure-Density Survey Program, WID, WS:87:00015.

Crawley, M. E., 1988. Third Data Release Report for the Pressure-Density Survey Program, WID, WS:88:00005.

D'Appolonia Consulting Engineers, 1981. Modeling Verification Studies: Long-Term Waste

Isolation Assessment, Waste Isolation Pilot Plant (WIPP) Project, Southeastern New Mexico, Consultant's Report to U.S. Department of Energy.

Davies, P. B., 1986. "Pleistocene-to-Present Flow-System Evolution in the Northern Delaware Basin, Southeastern New Mexico--Analysis Using Transient Cross-Sectional Flow Simulations," Geological Society of America, Abstracts with Programs, Vol. 18, No. 6, p. 580.

Davies, P. B., 1987. "Modeling Area, Variable-Density Groundwater Flow Using Equivalent Freshwater Head-Analysis of Potentially Significant Errors," Proceedings of the NWWA Conference on Solving Groundwater Problems with Models, February 10-12, Denver, Colorado, pp. 888-903.

Davies, P. B., 1988. Variable Density Groundwater Flow and Paleohydrology in the Region Surrounding the Waste Isolation Pilot Plant (WIPP), Southeastern New Mexico, U.S. Geological Survey Open-File Report 88-490.

Deal, D. E., 1988. "Brine Seepage to the Waste Isolation (WIPP) Excavations," Conference on Waste Management, Proceedings of Feb. 28 - Mar. 3, University of Arizona.

Deal, D. E., J. B. Case, R. M. Deshler, P. E. Drez, J. Myers, and J. R.. Tyburski, 1987. Brine-Sampling and Evaluation Program: Phase II Report, DOE-WIPP 87-010, Carlsbad, New Mexico.

De Marsily, G., 1986. Quantitative Hydrogeology. Academic Press, New York.

Dennehy, K. F., 1982. "Results of Hydrologic Tests and Water-Chemistry Analyses, Wells H-6A, H-6B, and H-6C at the Proposed Waste Isolation Pilot Plant Site in Southeastern New Mexico," U.S. Geological Survey Water-Resources Investigations 82-8.

Dennehy, K. F., and J. W. Mercer, 1982. "Results of Hydrologic Tests and Water-Chemistry Analyses, Wells H-5A, H-5B, and H-5C at the Proposed Waste Isolation Pilot Plant Site in Southeastern New Mexico," U.S. Geological Survey Water-Resources Investigations 82-19.

Dillon, R. T., R. B. Lantz, and S. B. Pahwa, 1977. Risk Methodology for Geologic Disposal of Radioactive Waste: The Sandia Waste Isolation Flow and Transport (SWIFT) Model, SAND78-1267, Sandia National Laboratories, Albuquerque, New Mexico.

Domenico, P. A., 1972. Concepts and Models in Groundwater Hydrology, McGraw-Hill Book Company, New York.

Dosch, R. G., 1978. "Radionuclide Migration Studies Associated with the WIPP Site in Southeastern New Mexico," SAND78-1178J, Sandia National Laboratories, Albuquerque, New Mexico, presented at the Symposium on Science Underlying Radioactive Waste Materials Research Society, Boston, Massachusetts, published in a Proceedings Volume, Science Underlying Radioactive Waste Management, by Plenum Publishing Co.

Dosch, R. G., 1980. Assessment of Potential Radionuclide Transport in Site-Specific Geologic Formations, SAND79-2468, Sandia National Laboratories, Albuquerque, New Mexico.

Dosch, R. G., 1981. Solubility and Sorption Characteristics of Uranium (VI) Associated with Rock Samples and Brines/Groundwaters from WIPP and NTS, SAND80-1595, Sandia National Laboratories, Albuquerque, New Mexico.

Dosch, R. G., and A. W. Lynch, 1979. Radionuclide Transport in a Dolomite Aquifer, SAND79-1016C, Sandia National Laboratories, Albuquerque, New Mexico, presented at Materials Research Society Meeting, Boston, Massachusetts.

Duffey, T. A., 1980. The Salt Block I Test: Experimental Details and Comparison with Theory, Final Report, SAND79-7050, Sandia National Laboratories, Albuquerque, New Mexico.

Eaton, R. R., and D. B. Hayes, 1981. Analytical and Numerical Analyses of Hydrologic Well-Bore Experiments, SAND80-1120, Sandia National Laboratories, Albuquerque, New Mexico.

EEG (Environmental Evaluation Group), 1980. The Significance of Certain Rustler Aquifer Parameters for Predicting Long-Term Radiation Doses from WIPP, EEG-8, C. Wofsy, Health and Environment Department, State of New Mexico, Santa Fe, New Mexico.

EEG (Environmental Evaluation Group), 1982. Hydrologic Analyses of Two Brine Encounters in the Vicinity of the Waste Isolation Pilot Plant, (WIPP) Site EEG-17, P. Spiegler, Health and Environment Department, State of New Mexico, Santa Fe, New Mexico.

EEG (Environmental Evaluation Group), 1983. Origin of the Brines Near WIPP from the Drill Holes ERDA-6 and WIPP-12 Based on Stable Isotope Concentrations of Hydrogen and Oxygen, EEG-18, P. Spiegler and D. Updegraff, Health and Environment Department, State of New Mexico, Santa Fe, New Mexico.

EEG (Environmental Evaluation Group), 1984. Sensitivity Analysis of Solute Transport in Fractures and Determination of Anisotrophy Within the Culebra Dolomite, EEG-27, K. Rehfeldt, Health and Environment Department, State of New Mexico, Santa Fe, New Mexico.

EEG (Environmental Evaluation Group), 1986. Stable Isotopes in Southeastern New Mexico Groundwater: Implications for Dating Recharge in the WIPP, EEG-35, J. B. Chapman, Health and Environment Department, State of New Mexico, Santa Fe, New Mexico.

EEG (Environmental Evaluation Group), 1988. Chemical and Radiochemical Characteristics of Groundwater in the Culebra Dolomite, Southeastern New Mexico, EEG-39, J. B. Chapman, Health and Environment Department, State of New Mexico, Santa Fe, New Mexico.

Federal Water Pollution Control Administration (FWPCA), 1967. Immediate Water Pollution Control Needs, Pecos River Basin, New Mexico and Texas, PB-266-387, South Central Region, Dallas, Texas.

Fortney, D. R., 1979. Sensitivity of the Consequences of Pu Migration in the WIPP Environment on Kd Values Used in Current Transport Models, SAND79-1305, Summary of Research and Development Activities in Support of Waste Acceptance Criteria for WIPP, Sandia National Laboratories, Albuquerque, New Mexico.

Fried, S., et al. 1981. "The Radiolytic Oxidation and Reduction of Plutonium, Argonne National Laboratories, 1979," SAND80-2577, Sandia National Laboratories, Albuquerque, New Mexico, presented at Materials Research Society Meeting Symposium, on Scientific Basis for Nuclear Waste Management, Boston, Massachusetts, published in Subseabed Disposal Program Annual Report, ed. D. M. Talbert, Vol. 2, pp. 271-284.

Geohydrology Associates, Inc., 1978. Groundwater Study Related to Proposed Expansion of Potash Mining Near Carlsbad, New Mexico, prepared for U.S. Bureau of Land Management, Denver, Colorado.

Geohydrology Associates, Inc., 1979. Water-Resources Study of the Carlsbad Potash Area, New Mexico, Albuquerque, New Mexico.

Gonzalez, D. D., 1983. Fracture Flow in the Rustler Formation, Waste Isolation Pilot Plant (WIPP), Southeastern New Mexico (SENM): Interim Report, SAND82-1012, Sandia National Laboratories, Albuquerque, New Mexico.

Gonzalez, D. D., 1983. Groundwater Flow in the Rustler Formation, Waste Isolation Pilot Plant (WIPP), Southeast New Mexico (SENM): Interim Report, SAND82-1012, Sandia National Laboratories, Albuquerque, New Mexico.

Gonzalez, D. D., 1983. Hydrogeochemical Parameters of Fluid-Bearing Zones in the Rustler and Bell Canyon Formations: Waste Isolation Pilot Plant (WIPP), Southeast New Mexico (SENM), SAND83-0210, Sandia National Laboratories, Albuquerque, New Mexico.

Gonzalez, D. D., 1983. "Hydrogeologic Characterization of Radioactive Waste Repositories: The WIPP Example," SAND83-2029A, in Transactions, AGU Fall Meeting Hydrology Screen, p. 713.

Gonzalez, D. D., and H. W. Bentley (Hydro Geo Chem, Inc.), 1984. "Field Test for Effective Porosity and Dispersivity in Fractured Dolomite, the WIPP, Southeastern New Mexico," Water Resources Monograph 9, American Geophysical Union, pp. 207-221.

Gonzalez, D. D., H. W. Bentley, and G. R. Walter. "Dispersion and Porosity Determinations at the H-2 Site, the WIPP, Southeastern New Mexico," Symposium of Ferris Water Resources Monograph Series, (in preparation).

Gonzalez, D. D., and L. R. Hill, 1979. "Tracer Tests Performed in the Field for WIPP in Southeastern New Mexico," SAND79-0981C, in Proceedings of the International Symposium on Scientific Basis for Nuclear Waste Management, Materials Research Society, Vol. 2, pp. 487-489.

Grove, D. B., and W. A. Beetum, 1971. "Porosity and Dispersion-Constant Calculations for a Fractured Carbonate Aquifer Using the Two-Well Tracer Method," Water Resources Research, Vol. 7.

Haberman, J. H., and D. J. Frydrych, 1988. "Corrosion Studies of A216 Grade WCA Steel in Hydrothermal Magnesium-Containing Brines," in Scientific Basis for Nuclear Waste Management XI, eds. M. J. Apted and R. E. Westerman, Materials Research Society, Pittsburgh, Pennsylvania, pp. 761-772.

Hale, W. E., 1945. "Groundwater Conditions in the Vicinity of Carlsbad, New Mexico," 16th and 17th Biennial Reports, New Mexico State Engineer, pp. 195-250.

Hale, W. E., 1961. "Availability of Groundwater in New Mexico," Sixth Annual New Mexico Water Conference, November 1-2, 1961, New Mexico State University, Las Cruces, New Mexico, pp. 11-22.

Hale, W. E., and A. Clebsch, Jr., 1958. "Preliminary Appraisal of Groundwater Conditions in Southeastern Eddy County and Southwestern Lea County, New Mexico," U.S. Geological Survey Trace Element Memorandum Report 1045.

Hale, W. E., L. S. Hughes, and E. R. Cox, 1954. Possible Improvement of Quality of Water of the Pecos River by Diversion of Brine at Malaga Bend, Eddy County, New Mexico, Pecos River Commission, New Mexico and Texas, in cooperation with U.S. Geological Survey Water Resources Division, Carlsbad, New Mexico.

Haug, A., V. A. Kelley, A. M. LaVenue, and J. F. Pickens, 1987. Modeling of Groundwater Flow in the Culebra Dolomite at the Waste Isolation Pilot Plant (WIPP) Site, Interim Report, SAND86-7167, Sandia National Laboratories, Albuquerque, New Mexico.

Havens, J. S., 1972. Malaga Bend Experimental Salinity Alleviation Project--A Comprehensive Interim Report--Eddy County, New Mexico, U.S. Geological Survey Water Resources Division, in cooperation with the Pecos River Commission.

Havens, J. S., and D. W. Wilkins, 1980. "Experimental Salinity Alleviation at Malage Bend of the Pecos River, Eddy County, New Mexico," U. S. Geological Survey Water-Resources Investigations 80-4.

Henrickson, G. E., and R. S. Jones, 1952. "Geology and Groundwater Resources of Eddy County, New Mexico," New Mexico Bureau of Mines and Mineral Resources Groundwater Report 3.

Herkenhoff, Gordon, and Associates, Inc., 1975. Water and Wastewater Master Plan, 1975-1995, Water Supply Study, City of Hobbs, New Mexico.,

Higgo, J. J., L. V. C. Rees, T. G. Coles, and D. S. Cronan, 1987. Diffusion of Radionuclides Through Deep-Sea Sediments, DOE/RW87.053, Department of the Environment, London, U.K.

Hiss, W. L., 1971. "Capitan Aquifer Observation-Well Network, Carlsbad to Jal, New Mexico," New Mexico State Engineer Technical Report 38.

Hiss, W. L., 1975. "Chloride-Ion Concentration in Groundwater in Permian Guadalupian Rocks, Southeast New Mexico and West Texas," Resource Map 5, New Mexico Bureau of Mines and Mineral Resources, Socorro, New Mexico.

Hiss, W. L., 1975. "Stratigraphy and Groundwater Hydrology of the Capitan Aquifer, Southeastern New Mexico and Western Texas," Ph.D. Thesis, University of Colorado, Boulder, Colorado.

Hiss, W. L., 1975. "Thickness of the Permian Guadalupian Aquifer, Southeastern New Mexico and West Texas," Resource Map 5, New Mexico Bureau of Mines and Mineral Resources, Socorro, New Mexico.

Hiss, W. L., 1975. Water Quality Data from Oil and Gas Wells in Part of the Permian Basin, Southeastern New Mexico and Western Texas, U.S. Geological Survey Open-File Report 75-579.

Hiss, W. L., 1976. "Structure of the Permian Guadalupian Aquifer, Southeastern New Mexico and Western Texas," Resource Map 6, New Mexico Bureau of Mines and Mineral Resources, Socorro, New Mexico.

Hiss, W. L., 1980. "Movement of Groundwater in Permian Guadalupian Aquifer Systems, Southeastern New Mexico and Western Texas, Trans-Pecos Region," New Mexico Geological Society, 31st Field Conference, pp. 289-294.

Hohlfelder, J. J., A. C. Beattie, and H. C. Shefelbine, 1982. Water Release and Mechanical Failure in Heated Geologic Salt, SAND81-1488, Sandia National Laboratories, Albuquerque, New Mexico.

Hohlfelder, J. J., and G. R. Hadley, 1979. "Laboratory Studies of Water Transport in Rock Salt," SAND79-1044J, Letters in Heat and Mass Transfer, Pergamon Press, Great Britain, Vol. 6, p. 271.

Hohlfelder, J. J., and W. M. McMurtry, 1983. Water Sources in Southeast New Mexico (SENM) Rock Salt, SAND82-0530, Sandia National Laboratories, Albuquerque, New Mexico.

Hunter, R. L., 1985. A Regional Water Balance for the Waste Isolation Pilot Plant (WIPP) Site and Surrounding Area, SAND84-2233, Sandia National Laboratories, Albuquerque, New Mexico.

Hydro Geo Chem, Inc., 1985. WIPP Hydrology Program Waste Isolation Pilot Plant, SENM: Hydrology Data Report No. 1, SAND85-7206, Sandia National Laboratories, Albuquerque, New Mexico.

Hydro Geo Chem, Inc., 1986. Two-Well Recirculation Tracer Test at the H-2 Hydropad, Waste Isolation Pilot Plant (WIPP), Southeastern New Mexico, SAND86-7092, Sandia National Laboratories, Albuquerque, New Mexico.

Intera Environmental Consultants, Inc., 1981. Consequence Assessment of Hydrological Communications Through Borehole Plugs, SAND81-7164, Sandia National Laboratories, Albuquerque, New Mexico.

Intera Technologies, Inc., 1986. WIPP Hydrology Program Waste Isolation Pilot Plant, Southeastern New Mexico, Hydrologic Data Report No. 3, SAND86-7109, Sandia National Laboratories, Albuquerque, New Mexico.

Intera Technologies, Inc. and Hydro Geo Chem, Inc., 1985. WIPP Hydrology Program, Waste Isolation Pilot Plant, Southeastern New Mexico, Hydrologic Data Report No. 2, SAND85-7263, Sandia National Laboratories, Albuquerque, New Mexico.

Jackson, K. J., 1988. Verification and Validation Studies of the Addition of Pitzers' Equations to the EQ3/6 Brine Model, UCRL-53841, Lawrence Livermore National Laboratory, Livermore, California.

Jennings, A. H., 1963. "Maximum Recorded U.S. Point Rainfall, Weather Bureau Technical Paper No. 2 (revised)," U.S. Department of Commerce, Washington, D.C.

Kelley, V. A., and J. F. Pickens, 1986. Interpretation of the Convergent-Flow Tracer Tests Conducted in the Culebra Dolomite at the H-3 and H-4 Hydropads at the Waste Isolation Pilot Plant (WIPP) Site, SAND86-7161, Sandia National Laboratories, Albuquerque, New Mexico.

Krauskopf, K. B., 1986. "Thorium and Rare-Earth Metals as Analogs for Actinide Elements," Chemical Geology, Vol. 55.

Kunkler, J. L., 1980. Evaluation of the Malaga Bend Salinity Alleviation Project, Eddy County, New Mexico, U.S. Geological Survey Open-File Report 80-1111.

Lambert, S. J., 1980. Mineralogical Aspects of Fluid Migration in the Salt Block II Experiment, SAND79-2423, Sandia National Laboratories, Albuquerque, New Mexico.

Lambert, S. J., 1980. Proposed Standards, Controls and Procedures for the Collection of Hydrologic Data for the Waste Isolation Pilot Plant (WIPP), Southeastern New Mexico, SAND80-0436, Sandia National Laboratories, Albuquerque, New Mexico, (Internal Distribution Only).

Lambert, S. J., 1987. "Stable-Isotope Studies of Groundwaters in Southeastern New Mexico," SAND85-1978C, Sandia National Laboratories, Albuquerque, New Mexico, in The Rustler Formation at the WIPP Site, ed. L. Chaturvedi, EEG-34, New Mexico Environmental Evaluation Group, Santa Fe, New Mexico.

Lambert, S. J., and J. W. Mercer, 1977. Hydrologic Investigations of Los Medanos Area, Southeast New Mexico, SAND77-1401, Sandia National Laboratories, Albuquerque, New Mexico.

Lambert, S. J., and J. W. Mercer, 1978. Hydrologic Investigations of the Los Medanos Area, Southeastern New Mexico, 1977, SAND77-1401, Sandia National Laboratories, Albuquerque, New Mexico.

Lambert, S. J., and H. C. Shefelbine, 1980. A Strategy for Investigation of Fluid Migration in Evaporites (Waste Isolation Pilot Plant WIPP), SAND79-1889, Sandia National Laboratories, Albuquerque, New Mexico.

Langmuir, D., and J. S. Herman, 1980. "The Mobility of Thorium in Natural Waters at Low Temperatures," Geochimica et Cosmochimica Acta, Vol. 44, pp. 1753-1766.

Lappin, A. L., 1988. Summary of Site-Characterization Studies Conducted from 1983 through 1987 at the Waste Isolation Pilot Plant (WIPP) Site, Southeastern New Mexico, SAND88-0157, Sandia National Laboratories, Albuquerque, New Mexico.

LaVenue, A. M., A. Haug, and V. A. Kelley, 1988. Numerical Simulation of Groundwater Flow in the Culebra Dolomite at the Waste Isolation Pilot Plant (WIPP) Site, Second Interim Report, SAND88-7002, Sandia National Laboratories, Albuquerque, New Mexico.

Lee, W. J., 1982. Well Testing, SPE Textbook Series, Vol. I, Society of Petroleum Engineers of AIME, Dallas, Texas.

Lynch, A. W., 1978. Migration of Cesium in Dolomite from the Rustler Formation in the Los Medanos Area in Southeastern New Mexico, SAND78-1179C, Sandia National Laboratories, Albuquerque, New Mexico, presented at the Materials Research Society - Science Underlying Radioactive Waste Management, Boston, Massachusetts.

Lynch, A. W., and R. G. Dosch, 1980. Cesium Migration Through Solid Cores of Magenta Dolomite, SAND80-1335C, Sandia National Laboratories, Albuquerque, New Mexico, presented at Materials Research Society Meeting, Boston, Massachusetts.

Lynch, A. W., and R. G. Dosch, 1980. Sorption Coefficients for Radionuclides on Samples from the Water-Bearing Magenta and Culebra Members of the Rustler Formation, SAND80-1064, Sandia National Laboratories, Albuquerque, New Mexico.

Lynch, A. W., R. G. Dosch, and C. R. Hills, 1981. Migration of Cesium-137 Through a Solid Core of Magenta Dolomite Taken from the Rustler Formation in SENM, SAND80-1259, Sandia National Laboratories, Albuquerque, New Mexico.

Magara, K., 1971. "Permeability Considerations in Generation of Abnormal Pressures," Journal of the Society of Petroleum Engineers, Vol. 11, pp. 236-242.

Mercer, J. W., 1983. "Geohydrology of the Proposed Waste Isolation Pilot Plant Site, Los Medanos Area, Southeastern New Mexico," U.S. Geological Survey Water-Resources Investigations Report 83-4016.

Mercer, J. W., 1987. Compilation of Hydrologic Data from Drilling the Salado and Castile Formations Near the WIPP Site, Southeastern New Mexico, SAND86-0954, Sandia National Laboratories, Albuquerque, New Mexico.

Mercer, J. W., R. L. Beauheim, R. P. Snyder, and G. M. Fairer, 1987. Basic Data Report for Drilling and Hydrologic Testing of Drillhole DOE-2 at the Waste Isolation Pilot Plant (WIPP) Site, SAND86-0611, Sandia National Laboratories, Albuquerque, New Mexico.

Mercer, J. W., P. Davis, K. F. Dennehy, and C. L. Geotz, 1981. "Results of Hydrologic Tests and Water-Chemistry Analyses, Wells H-4A, H-4B, and H-4C at the Proposed Waste Isolation Pilot Plant Site, Southeastern New Mexico," U.S. Geological Plant Site, Southeastern New Mexico," U.S. Geological Survey, Water-Resources Investigations, 81-3666.

Mercer, J. W., and D. D. Gonzalez, 1981. "Geohydrology of the Proposed Waste Isolation Pilot Plant in Southeastern New Mexico," in Environmental Geology and Hydrology in New Mexico, New Mexico Geological Society Special Publication No. 10, pp. 123-131.

Mercer, J. W., and B. R. Orr, 1977. Review and Analysis of Geohydrology Conditions Near the Site of a Potential Nuclear-Waste Repository, Eddy and Lea Counties, New Mexico, U.S. Geological Survey Open-File Report, 77-123, 35 pp.

Mercer, J. W., and B. R. Orr, 1978. "Geohydrology of the Proposed Waste Isolation Pilot Plant, Southeastern New Mexico," U.S. Geological Survey, Preliminary Water-Resources Investigations.

Mercer, J. W., and B. R. Orr, 1979. "Interim Data Report on the Geohydrology of the Proposed Waste Isolation Pilot Plant Site, Southeastern New Mexico," U.S. Geological Survey Water-Resources Investigations 79-98.

Mercer, J. W., and B. R. Orr. Interim Data Report on the Geohydrology of the Proposed Waste Isolation Pilot Plant Site, Southeastern New Mexico, U.S. Geological Survey.

Mercer, J. W., and B. R. Orr. Interim Data Report on the Geohydrology of the Proposed Waste Isolation Pilot Plant Site, Los Medanos Area, Southeastern New Mexico, U.S. Geological Survey Water Investigations Report 83-4016, U.S. Geological Survey (not funded through Sandia).

Motts, W. S., 1968. "The Control of Groundwater Occurrence by Lithofacies in the Guadalupian Reef Complex Near Carlsbad, New Mexico," Geological Society of America Bulletin, Vol. 79, pp. 283-298.

Narasimhan, T. N., and B. Y. Kanehiro, 1980. "A Note on the Meaning of Storage Coefficient," Water Resources Research, Vol. 16, No.2, pp.423-429.

Nash, K., S. Fried, and A. Friedman, 1981. "Radiolysis of Np in Environmental Solutions, Argonne National Laboratories, 1979," SAND80-2577, Sandia National Laboratories, Albuquerque, New Mexico, published in Subseabed Disposal Program Annual Report, ed. D. M. Talbert, Vol. 2, pp. 271-284.

New Mexico Interstate Stream Commission and New Mexico State Engineer's Office, 1975. County Profile for Eddy County and County Profile for Lea County.

New Mexico State Engineer's Office, 1967. Water Resources of New Mexico Occurrence, Development and Use.

Nowak, J. E., 1984. Diffusion of Plutonium in Compacted, Brine-Saturated Bentonite, SAND84-1118C, Sandia National Laboratories, Albuquerque, New Mexico.

Nowak, J. E. The Migration of Eu Through Geologic Media, SAND70-1746J, Sandia National Laboratories, Albuquerque, New Mexico.

Pearson, F. J., V. A. Kelley, and J. F. Pickens, 1986. Preliminary Design for A Sorbing Tracer Test in the Culebra Dolomite at the H-3 Hydropad at the Waste Isolation Pilot Plant, SAND86-7177, Sandia National Laboratories, Albuquerque, New Mexico.

Pearson, F. J., V. A. Kelley, and J. F. Pickens, 1987. Preliminary Design for a Sorbing Tracer Test in the Culebra Dolomite at the H-3 Hydropad at the Waste Isolation Pilot Plant (WIPP) Site, SAND86-7177, Sandia National Laboratories, Albuquerque, New Mexico.

Phillips, R. H., 1989. "The Prospects for the Regional Groundwater Contamination Due to Karst Landforms in Mescalero Caliche at the WIPP Site near Carlsbad, New Mexico," in preparation.

Pickens, J. F., and G. E. Grisak, 1981. "Scale-Dependent Dispersion in a Stratified Granular Aquifer," Water Resources Research, Vol. 17, No.4, pp. 1191-1211.

Pickens J. F., G. E. Grisak, J. D. Avis, D. W. Belanger, and M. Thury, 1987. "Analysis and Interpretation of Borehole Hydraulic Tests in Deep Boreholes: Principles, Model Development, and Applications," Water Resources Research, Vol. 23, No. 7, pp. 1341-1375.

Popielak, R. S., R. L. Beauheim, S. R. Black, W. E. Coons, C. T. Ellingson, and R. L. Olsen, 1983. "Brine Reservoirs in the Castile Formation, Waste Isolation Pilot Plant (WIPP) Project, Southeastern New Mexico," TME 3153, U.S. Department of Energy.

Ramey, D. S., 1985. Chemistry of Rustler Fluids, EEG-31. New Mexico Environmental Evaluation Group, Santa Fe, New Mexico.

Randall, W. S., M. E. Crawley, and M. L. Lyon, 1988. 1988 Annual Water Quality Data Report for the Waste Isolation Pilot Plant, DOE/WIPP 88-006, Carlsbad, New Mexico.

Reeves, M., V. A. Kelley, and J. F. Pickens, 1987. Regional Double-Porosity Solute Transport in the Culebra Dolomite: An Analysis of Parameter Sensitivity and Importance at the Waste Isolation Pilot Plant (WIPP) Site, SAND87-7105, Sandia National Laboratories, Albuquerque, New Mexico.

Reeves, M., D. S. Ward, N. D. Johns, and R. M. Cranwell, 1986. Data Input Guide for SWIFT II, the Sandia Waste Isolation Flow and Transport Model for Fractured Media, Release 4.84, SAND84-1586, Sandia National Laboratories, Albuquerque, New Mexico (also NUREG/CR-3925).

Reeves, M., D. S. Ward, N. D. Johns, and R. M. Cranwell, 1986. Theory and Implementation for SWIFT II, the Sandia Waste Isolation Flow and Transport Model, Release 4.84, SAND83-1159, Sandia National Laboratories, Albuquerque, New Mexico (also NUREG/CR-3328).

Register, J. K., 1980. "Subsurface Hydrology of Strata Overlying the Salado Formation at the Waste Isolation Pilot Plant Site, Eddy County, New Mexico," TME-3059, Waste Isolation Pilot Plant, Carlsbad, New Mexico.

Register, J. K., 1981. "Brine Pocket Occurrences in the Castile Formation, Southeastern New Mexico," TME-3080, U.S. Department of Energy, Albuquerque, New Mexico.

Richey, S. F., 1986. Hydrologic-Test Data from Wells at Hydrologic-Test Pads H-7, H-8, H-9 and H-10 Near the Proposed Waste Isolation Pilot Plant, Southeastern New Mexico, U.S. Geological Survey Open-File Report 86-413.

Richey, S. F., 1987. Preliminary Hydrologic Data for Wells Tested in Nash Draw, Near the Proposed Waste Isolation Pilot Plant, Southeastern New Mexico, U.S. Geological

Survey Open-File Report 87-37, Albuquerque, New Mexico.

Richey, S. F., 1987. Water-Level Data from Wells in the Vicinity of the Waste Isolation Pilot Plant, Southeastern New Mexico, USGS Open-File Report 87-120, Albuquerque, New Mexico.

Richey, S. F., J. G. Wells, and K. T. Stephens, 1985. "Geohydrology of the Delaware Basin and Vicinity, Texas and New Mexico," U.S. Geological Survey Water Resources Investigation Report 84-4077, Albuquerque, New Mexico.

Rose, W., 1977. "Development of a Conceptual Model to Explain Brine Flowage into Wells Drilled Through the Salado-Castile Formation: Contact in Locations such as the ERDA #6 Site Near Carlsbad in Southeastern New Mexico," Report to Sandia National Laboratories, Albuquerque, New Mexico.

Sares, S. W., 1984. "Hydrologic and Geomorphic Development of a Low Relief Evaporite Karst Drainage Basin, Southeastern New Mexico," University of New Mexico, Master's Thesis, Albuquerque, New Mexico.

Saulnier, G. J., Jr., 1987. Analysis of Pumping Tests of the Culebra Dolomite Conducted at the H-11 Hydropad at the Waste Isolation Pilot Plant (WIPP) Site, SAND87-7124, Sandia National Laboratories, Albuquerque, New Mexico.

Saulnier, G. J., Jr., and J. D. Avis, 1988. Interpretation of Hydraulic Tests Conducted in the Waste-Handling Shaft at the Waste Isolation Pilot Plant (WIPP) Site, SAND87-7001, Sandia National Laboratories, Albuquerque, New Mexico.

Saulnier, G. J., Jr., et al., 1987. Program Waste Isolation Pilot Plant Southeastern New Mexico, Hydrologic Data Report No. 4, SAND86-7166, Sandia National Laboratories, Albuquerque, New Mexico.

Saulnier, G. J., Jr., G. A. Freeze, and W. A. Stensrud, 1987. WIPP Hydrology Program, Waste Isolation Pilot Plant, Southeastern New Mexico, Hydrologic Data Report No. 4, SAND86-7166, Sandia National Laboratories, Albuquerque, New Mexico.

Scientists Review Panel on WIPP, 1988. Evaluation of the Waste Isolation Pilot Plant (WIPP) as a Water-Saturated Nuclear Waste Repository, Albuquerque, New Mexico.

Seitz, M. G., D. Bowers, and D. R. Fortney, 1981. A Groundwater Stream Experiment for the Waste Isolation Pilot Plant, ANL-81-28, Argonne National Laboratory, Argonne, Illinois.

Serne, R. J., D. Rai, M. J. Mason, and M. A. Molecke, 1977. "Batch Kd Measurements of Nuclides to Estimate Migration Potential at the Proposed Waste Isolation Pilot Plant in New Mexico," Battelle Pacific Northwest Laboratories Report PNL-2448.

Shefelbine, H. C., 1982. Brine Migration: A Summary Report, SAND82-0152, Sandia National Laboratories, Albuquerque, New Mexico.

Soudek, A., 1984. Binary and Ternary Ion Exchange on Sodium Montmorillonite for Application to Backfill in Nuclear Water Disposal," Master's Thesis, University of California, Berkeley, California.

Stensrud, W. A., M. A. Bame, K. D. Lantz, T. L. Cauffman, J. B. Palmer, and G. J. Saulnier, Jr., 1988. WIPP Hydrology Program, Waste Isolation Pilot Plant Site, Southeastern New Mexico, Hydrologic Data Report No. 6, SAND87-7166, Sandia National Laboratories, Albuquerque, New Mexico.

Stensrud, W. A., M. A. Bame, K. D. Lantz, A. M. LaVenue, J. B. Palmer, and G. J. Saulnier, Jr., 1987. WIPP Hydrology Program, Waste Isolation Pilot Plant Site, Southeastern New Mexico, Hydrologic Data Report No. 5, SAND87-7125, Sandia National Laboratories, Albuquerque, New Mexico.

Stopps, G. J., and M. McLaughlin, 1967. American Ind. Hyg. Assoc. J., Vol. 23, p. 43.

Tomasko, D., 1986. The Effect of Side-Wall Boundary Conditions on Convective Heat Transfer in a Porous Medium Heated from Below, SAND86-1549A, Sandia National Laboratories, Albuquerque, New Mexico, presented at CUBE Symposium, Livermore, California.

Tomasko, D., 1987. The Effects of a Bimodel Pore Distribution on Matrix Diffusion in a Fractured Porous Medium, SAND87-0265, Sandia National Laboratories, Albuquerque, New Mexico.

Tomasko, D., 1987. Optimally Orienting a Parallel-Strip Aquifer Using Multiwell Test Data, SAND86-1440J, Sandia National Laboratories, Albuquerque, New Mexico, presented at Solving Groundwater Problems with Models, Denver, Colorado.

Tomasko, D., and A. L. Jensen, 1987. A Linear-Flow Interpretation of the H-3 Multiwell Pumping Test Conducted at the Waste Isolation Pilot Plant (WIPP) Site, SAND87-0630, Sandia National Laboratories, Albuquerque, New Mexico.

Trucano, T. G., 1981. Computer Modeling of Stress States Association with Fluid Migration Experiments, SAND81-2008, Sandia National Laboratories, Albuquerque, New Mexico.

Uhland, D. W., and W. S. Randall, 1986. Annual Water Quality Data Report, DOE/WIPP-86-006, Waste Isolation Pilot Plant, Carlsbad, New Mexico.

Uhland, D. W., W. S. Randall, and R. C. Carrasco, 1987. Annual Water Quality Data Report, DOE/WIPP-87-007, Waste Isolation Pilot Plant, Carlsbad, New Mexico.

U.S. Army Corps of Engineers, 1976. "Hydrology, Pecos River and Tributaries at and in the Vicinity of Carlsbad, New Mexico," Feasibility Report for the Water Resources Development, Albuquerque, New Mexico.

USGS (U.S. Geological Survey), 1976. Water Resources Data for New Mexico, Water Year 1975, NM-75-1, Department of the Interior, Albuquerque, New Mexico.

USGS (U.S. Geological Survey), 1977. Water Resources Data for New Mexico, Water Year 1976, NM-76-1, Albuquerque, New Mexico.

Van Everdingen, A. F., and W. Hurst, 1949. "Application of the Laplace Transformation to Flow Problems in Reservoirs," Transactions of the Society of Petroleum Engineers, American Institute of Mining Engineers, Dallas, Texas, Vol. 186, p. 305-324.

Walter, G. R., 1983. Aquifer Tests to Determine the Principal Components of Trans-missivity in the Culebra Dolomite at Hydropads H-4, H-5 and H-6: Waste Isolation Pilot Plant (WIPP) Southeastern New Mexico, SAND83-7009, Sandia National Laboratories, Albuquerque, New Mexico.

Walter, G. R. "Convergent-Flow Tracer Test at H-6: Waste Isolation Pilot Plant (WIPP), Southeastern New Mexico," SAND83-7015, (in preparation), Sandia National Laboratories, Albuquerque, New Mexico.

Walter, G. R., H. W. Bentley, and J. J. Ward, Hydro Geo Chem, Inc., and D. D. Gonzalez, 1983. "Continuum and Non-Continuum Aspects of Groundwater Movement at the Waste Isolation Pilot Plant, Southeastern New Mexico," Sandia National Laboratories, Albuquerque, New Mexico, in Transactions, AGU Fall Meeting Hydrology Section, p. 713.

Walter, G. R., and Hydro Geo Chem, Inc., 1983. A Three-Well Technique for Determining Formation Anisotropy at the Waste Isolation Pilot Plant (WIPP) Southeastern New Mexico, SAND83-7011, Sandia National Laboratories, Albuquerque, New Mexico.

Ward, J. J., and D. D. Gonzales, 1981. "Anisotrophy Tests in the Culebra Dolomite at F-4, H-5, and H-6 Sites, The WIPP, Southeast New Mexico," (in preparation).

Ward, J. J., and G. R. Walter, 1983. A Three Well Technique for Determining Formation Anisotropy at the Waste Isolation Pilot Plant (WIPP) Southeast New Mexico, SAND83-7011, Sandia National Laboratories, Albuquerque, New Mexico.

Ward, J. J., G. R. Walter, Hydro Geo Chem Inc., and D. D. Gonzalez, 1983. "Heterogeneity and Anisotrophy of a Groundwater System in Carbonate Rocks at the Waste Isolation Pilot Plant (WIPP)," Sandia National Laboratories, Albuquerque, New Mexico, in Transactions AGU Fall Meeting Hydrology Section.

Welder, G. E, 1983. "Geohydrologic Framework of the Roswell Groundwater Basin, Chavez and Eddy Counties, New Mexico," New Mexico State Engineer Technical Report 42.

Wilson B. H., G. G. Smith, and J. F. Rees, 1986. "Biotransformation of Selected Alkylbenzenes and Halogenated Aliphatic Hydrocarbons in Methanogenic Aquifer Material: A Microcosm Study," Environ. Sc. Technology, Vol. 20, pp. 997-1002.

Winstanley, D. J., and R. C. Carrasco, 1986. Annual Hydrogeologic Data Report 1985/1986, DOE/WIPP-004, Waste Isolation Pilot Plant, Carlsbad, New Mexico.

Wofsy, C., 1980. The Significance of Certain Rustler Aquifer Parameters for Predicting Long-Term Radiation Doses from WIPP, EEG-8, State of New Mexico Environmental Evaluation Group, Santa Fe, New Mexico.

Wood, B. J., R. E. Snow, D. J. Cosler, and S. Haji-Djafari, 1982. "Delaware Mountain Group (DMG) Hydrology - Salt Removal Potential," TME-3166, U.S. Department of Energy,

Albuquerque, New Mexico.

J.7 **REPOSITORY TECHNOLOGY**

Abrego, L., and J. W. Braithwaite, 1980. Corrosion of Titanium in Saline Nuclear Waste Isolation Environments, SAND80-0325A, Sandia National Laboratories, Albuquerque, New Mexico, presented at the American Nuclear Society Annual Meeting, Las Vegas, Nevada.

Abrego, L., and H. J. Rack, 1981. The Slow Strain Rate Behavior of TiCode 12 (ASTM Gr. 12) in Aqueous Chloride Solutions, SAND80-1738C, Sandia National Laboratories, Albuquerque, New Mexico, presented at Corrosion/81, National Association of Corrosion Engineers Meeting, Toronto, Canada.

Acton, R. V., 1977. "Thermal Conductivity of SE New Mexico Rock Salt and Anhydrite," unpublished Sandia National Laboratories preliminary report, Albuquerque, New Mexico.

Anderson, H. M., J. Pietz, and D. Smith, 1983. Solutions to Radionuclide Migration with Nonlinear Retardation Mechanisms in Backfill Material, Interim Progress Report, June-December 1982, SAND82-7089, Sandia National Laboratories, Albuquerque, New Mexico.

Anderson, H. M., J. M. Pietz, and D. M. Smith, 1984. "Radionuclide Transport Modeling of Diffusion Cell Experiments Specific to a Backfill Barrier in a Salt Repository," Scientific Basis for Nuclear Waste Management VII, Materials Research Society Symposia Proceedings, Vol. 26, ed. G. L. McVay, North-Holland, New York.

Arguello, J. G., 1987. Numerical Simulation of WIPP Panel Entryway Seal/Formation Structural Interaction, SAND87-2257A, Sandia National Laboratories, Albuquerque, New Mexico, presented at the Ninth U.S. Symposium on Rock Mechanics at Minneapolis, Minnesota.

Arguello, J.G., 1987. WIPP Panel Entryway Seal---Numerical Simulation of Seal Composite Interaction for Preliminary Seal Design Evaluation, SAND87-2804, Sandia National Laboratories, Albuquerque, New Mexico.

Arguello, J. G., and R. Beraum, 1986. Numerical Simulation of Drift Response Resulting from the Emplacement of RH TRU Waste in an Array of Horizontal Long Boreholes in a Separate Panel, SAND86-2224A, Sandia National Laboratories, Albuquerque, New Mexico, presented at the U.S. Symposium on Rock Mechanics, Tucson, Arizona.

Arguello, J. G., and R. Beraun, 1987. Numerical Simulation of Drift Response in Rock Salt Resulting From the Emplacement of RH TRU Waste in an Array of Horizontal Long Boreholes in a Separate Panel at the WIPP, SAND86-2224C, Sandia National Laboratories, Albuquerque, New Mexico.

Arguello, J. G., and H. S. Morgan, 1986. Numerical Simulation of the Creep Response of Model Salt Pillars, SAND85-2429, Sandia National Laboratories, Albuquerque, New

Mexico.

Arguello, J. G., D. E. Munson, and D. S. Preece, 1987. Preliminary Results of the Three-Dimensional Modeling of the WIPP Room D Excavation Sequence, SAND86-2380C, Sandia National Laboratories, Albuquerque, New Mexico, presented at U.S. Symposium on Rock Mechanics, Tucson, Arizona.

Arguello, J. G., and T. M. Torres, 1987. WIPP Panel Entryway Seal---Numerical Simulation of Seal Component/Formation Interaction for Preliminary Seal Design Evaluation, SAND87-2591, Sandia National Laboratories, Albuquerque, New Mexico.

Arnold, G. W., 1980. Radiation Hardening of Rock Salt, SAND80-1566C, Sandia National Laboratories, Albuquerque, New Mexico, presented at Annual Meeting of the Materials Research Society, Boston, Massachusetts.

Baes, C. F., Jr., L. O. Gilpatrick, F. G. Kitts, H. R. Bronstein, and A. J. Shor, 1983. The Effect of Water in Salt Repositories: Final Report, ORNL-5950, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Bard, S. T., 1982. Estimated Radiation Doses Resulting if an Exploratory Borehole Penetrates a Pressurized Brine Reservoir Assumed to Exist Below the WIPP Repository Horizon---A Single Borehole Scenario, EEG-15, New Mexico Environmental Evaluation Group, Santa Fe, New Mexico.

Barnhart, B. J., D. E. Caldwell, et al., 1978-79. (Los Alamos Scientific Laboratory and University of New Mexico) Potential Microbial Impact on Transuranic Wastes Under Conditions Expected in the Waste Isolation Pilot Plant (WIPP), Progress Reports: LA-7788-PR, LA-7839-PR, LA-7918-PR, LA-8297-PR.

Barr, G. E., F. W. Bingham, and M. S. Tierney, 1980. The Use and Misuse of Generic Scenarios, SAND80-1537A, Sandia National Laboratories, Albuquerque, New Mexico, presented at OECD Workshop on Radionuclide Release Scenarios for Geologic Repositories in Paris, France.

Barrows, L. J., and C. W. Cook, 1980. WIPP Statement of Work for Subsurface Geophysical Survey of the Hill C Breccia Pipe, Sandia National Laboratories, Albuquerque, New Mexico.

Barthel, J. M., 1988. "Supercompaction and Repackaging Facility for Rocky Flats Plant Transuranic Waste," Proceedings of the Symposium on Waste Management at Tucson, Arizona, February, Vol. II, pp. 317-323.

Beall, G. W., 1982. Chemistry and Solution Effects on Radionuclides in Backfill Materials, Final Report, draft, SAND82-7222, Sandia National Laboratories, Albuquerque, New Mexico.

Beall, G. W., 1982. A Literature Survey of Near-Field Effects in a Bedded Salt Nuclear Waste Repository Site, draft, SAND81-7218, Sandia National Laboratories, Albuquerque, New Mexico.

Beauheim, R. L., 1988. Salado Permeability Measurements in Holes Drilled from the Surface and the Waste-Handling Shaft, presentation to the National Research Council/National

Academy of Science WIPP Panel.

Beauheim, R. L., S. R. Black, C. L. Christensen, T. O. Hunter, L. Jarolimek, A. K. Kuhn, S. J. Lambert, R. F. McKinney, D. Meyer, J. B. Murray, R. S. Newton, D. W. Powers, C. L. Stein, D. E. Stephenson, B. R. Sullivan, M. J. Timmer, and W. D. Weart, 1983. Results of Site Validation Experiments, Waste Isolation Pilot Plant (WIPP) Project, Southeastern New Mexico, Vols. I and II, TME-3177, U.S. Department of Energy, Albuquerque, New Mexico.

Bechtel National, Inc., 1983. "Waste Isolation Pilot Plant, Preliminary Design Validation Report," compiled for U.S. Department of Energy, Albuquerque, New Mexico.

Beraun, R., and M. A. Molecke, 1987. Numerical Thermal Modeling of the WIPP In Situ Room A1 Simulated DHLW Experiments, SAND87-0467C, Sandia National Laboratories, Albuquerque, New Mexico, presented at Waste Management 1987, Tucson, Arizona.

Beraun, R., and M. A. Molecke, 1987. Thermal Analysis of the WIPP In Situ Room A1 DHLW Package Experiments, SAND86-0681, Sandia National Laboratories, Albuquerque, New Mexico.

Bibler, N. E., 1977. TRU Waste Form and Package Criteria Meeting, Sheraton Old Town, Albuquerque, New Mexico, (cassettes only), SAND77-1178, Sandia National Laboratories, Albuquerque, New Mexico.

Biffle, J. H., 1981. Second Benchmark Problem for WIPP Structural Computation with the JAC Computer Program, SAND81-0767, Sandia National Laboratories, Albuquerque, New Mexico.

Biffle, J. H., 1984. JAC---A Two-Dimensional Finite Element Computer Program for the Non-Linear Quasistatic Response of Solids with the Conjugate Gradient Method, SAND81-0998, Sandia National Laboratories, Albuquerque, New Mexico.

Bingham, F. W., and G. E. Barr, 1979. "Development of Scenarios for the Long-Term Release of Radionuclides from the Proposed Waste Isolation Pilot Plant in Southeastern New Mexico," SAND79-0955c, Sandia National Laboratories, Albuquerque, New Mexico, published in Proceedings of International Symposium on Scientific Basis for Nuclear Waste Management, Materials Research Society, Vol. 2, pp. 771-778.

Black, S. R., R. S. Newton, and D. K. Shukla, eds., 1983. Results of Site Validation Experiments, Vols. I and II, TME-3177, prepared for U.S. Department of Energy by TSC-D'Appolonia.

Boa, J. A., Jr., 1978. Borehole Plugging Program (Waste Disposal), Report 1, Initial Investigations and Preliminary Data, SAND77-7005, Sandia National Laboratories, Albuquerque, New Mexico.

Boa, J. A., Jr., 1981. Borehole Plugging Program (Waste Disposal), Report 2, Petrographic Examination of Several Four-Year-Old Laboratory-Developed Grout Mixtures, MPC-78-1, prepared for Sandia National Laboratories, Albuquerque, New Mexico.

Borns, D. J., and J. C. Stormont, 1987. "Delineation of the Disturbed Rock Zone Around

Excavations in Salt, Waste Isolation Pilot Plant (WIPP), Southeastern New Mexico," Geological Society of America Abstracts with Programs, Vol. 19, No. 6.

Borns, D. J., and J. C. Stormont, 1987. "Delineation of the Disturbed Rock Zone Around Excavations in Salt," Geological Society of America Abstracts with Programs, Vol. 19, No. 7.

Borns, D. J., and J. C. Stormont, 1988. "An Interim Report on Excavation Effects Studies at the Waste Isolation Pilot Plant," SAND86-xxx, (in preparation), Sandia National Laboratories, Albuquerque, New Mexico.

Boyd, R. D., 1978. Forced-Air Cooling of a WIPP Mine Drift, SAND78-1122, Sandia National Laboratories, Albuquerque, New Mexico.

Bradshaw, R. L., and W. C. McClain, 1971. Project Salt Vault, A Demonstration of the Disposal of High-Activity Solidified Waste in Underground Salt Mines, Oak Ridge National Laboratories, No. 4555, Oak Ridge, Tennessee.

Bradshaw, R. L., and F. Sanchez, 1969. "Migration of Brine Cavities in Rock Salt," Journal of Geophysical Research, Vol. 74, pp. 4209-4212.

Braithwaite, J. W., 1980. "Brine Chemistry Effects on the Durability of a Simulated Nuclear Waste Glass," SAND79-0900C, Sandia National Laboratories, Albuquerque, New Mexico, presented at Materials Research Society, Boston, Massachusetts, published in Scientific Basis for Nuclear Waste Management, Vol. 2, ed. C. J. Northrup, Plenum Press, New York.

Braithwaite, J. W., 1987. The Corrosion Behavior of Metals in Terminal Nuclear Waste Storage Environments, SAND77-1812C, Sandia National Laboratories, Albuquerque, New Mexico, presented at Corrosion/78, National Association of Corrosion Engineers, Houston, Texas.

Braithwaite, J. W., and J. K. Johnstone, 1980. "Chemical Durability and Characterization of Nuclear Waste Forms in a Hydrothermal Environment," SAND78-2259C, Sandia National Laboratories, Albuquerque, New Mexico, presented at American Ceramic Society Conference, Cincinnati, Ohio, published in Ceramics in Nuclear Waste Management, CONF-79420.

Braithwaite, J. W., and N. J. Magnani, 1979. "Corrosion Considerations for Nuclear Waste Isolation Canisters," SAND78-1093C, Sandia National Laboratories, Albuquerque, New Mexico, presented at Materials Research Society, Boston, Massachusetts, published in Scientific Basis for Nuclear Waste Management, Vol. 1, ed. G. J. McCarthy, Plenum Press, New York.

Braithwaite, J. W., N. J. Magnani, and J. W. Munford, 1979. Corrosion of Candidate Nuclear Waste Canister Alloys, SAND79-1560C, Sandia National Laboratories, Albuquerque, New Mexico, presented at AIME Meeting, Las Vegas, Nevada.

Braithwaite, J. W., N. J. Magnani, and J. W. Munford, 1980. Titanium Alloy Corrosion in Nuclear Waste Environments, SAND79-2023C, Sandia National Laboratories, Albuquerque, New Mexico, presented at Corrosion 80, National Association of

Corrosion Engineers, Chicago, Illinois.

Braithwaite, J. W., and M. A. Molecke, 1979. Corrosion and Anti-Corrosion Coatings Studies for TRU Waste Containers, Section 7 in Summary of Research and Development Activities in Support of Waste Acceptance Criteria for WIPP, SAND79-1305, Sandia National Laboratories, Albuquerque, New Mexico.

Braithwaite, J. W., and M. A. Molecke, 1978. High-Level Waste Canister Corrosion Studies Pertinent to Geologic Isolation, SAND78-2111, Sandia National Laboratories, Albuquerque, New Mexico.

Braithwaite, J. W., and M. A. Molecke, 1980. "Nuclear Waste Canister Corrosion Studies Pertinent to Geologic Isolation," SAND79-1935J, Sandia National Laboratories, Albuquerque, New Mexico, published in Nuclear and Chemical Waste Management, Vol. 1, No. 1, pp. 37-50.

Brandshaug, Terje (RE/SPEC, Inc.), 1986. A Thermomechanical Analysis of WIPP DHLW Experiments: Waste Container/Borehole Interactions, SAND85-7210, Sandia National Laboratories, Albuquerque, New Mexico.

Branstetter, L. J., 1983. Pretest Parametric Calculations for the Heated Pillar Experiment in the WIPP In Situ Experimental Area, SAND82-2781, Sandia National Laboratories, Albuquerque, New Mexico.

Branstetter, L. J., 1985. Second Reference Calculation for the WIPP, SAND82-2461, Sandia National Laboratories, Albuquerque, New Mexico.

Branstetter, L. J., R. D. Krieg, and C. M. Stone, 1981. A Method for Modeling Regional Scale Deformation and Stresses Around Radioactive Waste Depositories in Bedded Salt, SAND81-0237, Sandia National Laboratories, Albuquerque, New Mexico.

Branstetter, L. J., R. D. Krieg, and C. M. Stone, 1982. The Effect of a Random Variation of Rock Salt Creep on Calculations of Storage Room Closure for the WIPP Project, SAND82-0024, Sandia National Laboratories, Albuquerque, New Mexico.

Branstetter, L. J., R. D. Krieg, and C. M. Stone, 1983. The Effects of a Random Variation of Rock Salt Creep on Closure Calculations for an Excavated Opening in a Complex Stratigraphy, SAND82-2256A, presented at ASCE Engineering Mechanics Division, Specialty Conference, Purdue University.

Branstetter, L. J., C. M. Stone, and R. D. Krieg, 1981. WIPP Benchmark II Results Using SANCHO, SAND81-0853, Sandia National Laboratories, Albuquerque, New Mexico.

Brausch, L. M., J. S. Hart, K. R. Porter, and J. K. Register, 1980. Technical and Environmental Analyses, Development of the Site and Preliminary Design Validation Program, Los Medanos Site, Southeastern New Mexico, TME 3058, WIPP Technical Support Contractor, Albuquerque, New Mexico.

Brunton, G. D., and W. C. McClain, 1977. Geological Criteria for Radioactive Waste Repositories, Y/OWI/TM-47, Office of Waste Isolation, Union Carbide Corporation, Oak Ridge, Tennessee.

Buck, A. D., 1985. Development of a Sanded Expansive Salt Grout for Repository Sealing Application, SL-85-6, Office of Nuclear Waste Isolation, Columbus, Ohio.

Buck, A. D., 1985. Development of Two Candidate Concrete Mixtures (Salt, Nonsalt) for Repository Sealing Applications, SL-85-8, Office of Nuclear Waste Isolation, Columbus, Ohio.

Buck, A. D., 1987. Effect of Salt-Saturated Mixing Water on Long-Term Durability of Hydraulic-Cement Mixtures, Structures Laboratory, USAE Waterways Experiment Station Draft Report to Sandia National Laboratories, Albuquerque, New Mexico.

Buck, A. D., J. A. Boa, and D. M. Walley, 1985. Development of a Sanded Expansive Salt Grout for Repository Sealing Application, SL-85-11, Office of Nuclear Waste Isolation, Columbus, Ohio.

Buck, A. D., J. P. Burkes, and J. E. Rhoderick, 1981. Examination of ERDA-10 Grout Specimens at Different Ages, USAE Waterways Experiment Station Miscellaneous Paper SL-81-20 (also issued as ONWI-341).

Buck, A. E., et al., 1983. Modification of Bell Canyon Test (BCT) 1-FF Grout, miscellaneous paper SL-83-18, SAND83-7097, U.S. Army Engineer Waterways.

Burkes, J. P., and J. E. Rhoderick, 1983. Petrographic Examination of Bell Canyon Tests (BCT) 1-FF Field Grouts Over a Three-Year Period, SL-83-12, Office of Nuclear Waste Isolation, Columbus, Ohio. Also published as SAND83-7115, Sandia National Laboratories, Albuquerque, New Mexico.

Butcher, B. M., 1980. Creep Consolidation of Nuclear Depository Backfill Materials, SAND79-22-12, Sandia National Laboratories, Albuquerque, New Mexico.

Butcher, B. M., 1981. An Elastic-Plastic Mechanical Constitutive Description for Rock Salt Triaxial Compression, SAND80-1935, Sandia National Laboratories, Albuquerque, New Mexico.

Butcher, B. M., and J. P. Hill, 1984. A Method for Defining the Distribution of Triaxial Stress States in Underground Structures, SAND83-2506, Sandia National Laboratories, Albuquerque, New Mexico.

Caldwell, D. E., M. A. Molecke, R. C. Hallet, E. Martinez, and B. J. Barnhart, 1987. Rates of CO_2 Production from the Microbial Degradation of Transuranic Wastes Under Simulated Geologic Isolation Conditions, SAND87-7170, Sandia National Laboratories, Albuquerque, New Mexico.

Canfield, T. R., 1983. Stresses Near Waste Canisters Buried in Salt, SAND82-0525, Sandia National Laboratories, Albuquerque, New Mexico.

Case, J. B., S. M. Dass, J. G. Franzone, and A. K. Kuhn, 1982. Analysis of Potential Impacts of Brine Flow Through Boreholes Penetrating the WIPP Storage Facility, TME-3155, U.S. Department of Energy, Albuquerque, New Mexico.

Case, J. B., P. C. Kelsall, and J. L. Withiam, 1987. "Laboratory Investigation of Crushed Salt Consolidation," Proceedings of the 28th U.S. Rock Mechanics Symposium, Tucson, Arizona, eds. I. W. Farmer, J. K. Daemen, C. S. Desai, C. E. Glass, and S. P. Neuman, Pub. A. A. Balkema, pp. 189-196.

Channell, J. K., 1982. Calculated Radiation Doses from Radionuclides Brought to the Surface if Future Drilling Intercepts the WIPP Repository and Pressurized Brine, EEG-11, New Mexico Environmental Evaluation Group, Santa Fe, New Mexico.

Chaturvedi, L., J. K. Channell, and J. B. Chapman, 1988. "Potential Problems Resulting from the Plans for the First Five Years of the WIPP Project," in Waste Management '88, ed. R. G. Post and M. E. Weeks, University of Arizona, Tucson, Arizona, pp. 355-364.

Christensen, C. L., 1979. Bell Canyon Test, WIPP Experimental Program Borehole Plugging, Sandia National Laboratories, Albuquerque, New Mexico.

Christensen, C. L., 1979. Field Test Programs of Borehole Plugs in Southeastern New Mexico, SAND79-1634C, Sandia National Laboratories, Albuquerque, New Mexico, presented at ONWI, Columbus, Ohio.

Christensen, C. L., 1979. Potash Core Test, WIPP Experimental Program Borehole Plugging, Sandia National Laboratories, Albuquerque, New Mexico.

Christensen, C. L., 1979. Test Plan---Bell Canyon Test, WIPP Experimental Program Borehole Plugging, SAND79-0739, Sandia National Laboratories, Albuquerque, New Mexico.

Christensen, C. L., 1979. Test Plan---Potash Core Test, WIPP Experimental Program Borehole Plugging, SAND79-0739, Sandia National Laboratories, Albuquerque, New Mexico.

Christensen, C. L., 1980. Results from the Bell Canyon Borehole Plugging Test, SAND80-2376C, presented at 1980 National Waste Terminal Storage Program Information Meeting, Columbus, Ohio.

Christensen, C. L., 1980. Sandia Borehole Plugging Program for the Waste Isolation Pilot Plant (WIPP), SAND80-0390C, Sandia National Laboratories, Albuquerque, New Mexico, presented at Workshop on Borehole and Shaft Plugging, Columbus, Ohio.

Christensen, C. L. 1980. Thermal Conductivity Probe Repeatability Assessment, Sandia National Laboratories, Albuquerque, New Mexico.

Christensen, C. L., 1981. Trapped Gas Experiment, Sandia National Laboratories, Albuquerque, New Mexico.

Christensen, C. L., 1984. Compliance with Prescribed Plugging Criteria for Reentry, Preparation, and Cementing of the B-25 Borehole at the WIPP Site, SAND83-2189A, Sandia National Laboratories, Albuquerque, New Mexico, presented at Waste Management 1984, Tucson, Arizona.

Christensen, C. L., C. W. Cook, and C. W. Gulick, 1979. Borehole Plugging Program Status Report, October 1, 1978 - September 30, 1979 (January 1980), SAND79-2141, Sandia

National Laboratories, Albuquerque, New Mexico.

Christensen, C. L., C. W. Gulick, and S. J. Lambert, 1981. Sealing Concepts for the Waste Isolation Pilot Plant (WIPP) Site, SAND81-2195, Sandia National Laboratories, Albuquerque, New Mexico.

Christensen, C. L., and T. O. Hunter, 1980. Bell Canyon Test and Results, SAND80-2414C, presented at Materials Research Society Meeting, Boston, Massachusetts.

Christensen, C. L., and T. O. Hunter, 1980. The Bell Canyon Test and Results, SAND80-1557A, Sandia National Laboratories, Albuquerque, New Mexico, presented at Scientific Basis for Nuclear Waste Management Symposium, Boston, Massachusetts.

Christensen, C. L., and E. W. Peterson, 1981. Bell Canyon Test Summary Report, SAND80-1375, Sandia National Laboratories, Albuquerque, New Mexico.

Christensen, C. L., and E. W. Peterson, 1981. "Field-Test Programs of Borehole Plugs in Southeastern New Mexico," in Advances in the Science and Technology of the Management of High-Level Nuclear Waste, Vol. I, Office of Nuclear Waste Isolation (ONWI), DOE/Tic-4621-Vol. 1, pp. 354-369.

Christensen, C. L., R. D. Statler, and E. W. Peterson, 1980. Downhole Television Applications in Borehole Plugging, SAND80-0459, Sandia National Laboratories, Albuquerque, New Mexico.

Christensen, C. L., and J. C. Stormont, 1984. Compliance with Prescribed Plugging Critera for the Reentry, Preparation, and Cementing of the B-25 Borehole at the WIPP Site, Memorandum of Record, Sandia National Laboratories, Albuquerque, New Mexico.

Claiborne, H. C., et al., 1980. Expected Environments in High-Level Nuclear Waste and Spent Fuel Repositories in Salt, ORNL/TM-7201, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Claiborne, H. C., and F. Gera, 1974. Potential Containment Failure Mechanisms and Their Consequences at a Radioactive Waste Repository in Bedded Salt in New Mexico, ORNL-TM-4639, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Clements, T. L., Jr., and D. E. Kudera, 1985. TRU-Waste Sampling Program, Vol. I: Waste Characterization, and Vol II: Gas Generation Studies, Idaho National Engineering Laboratory, prepared for U.S. Department of Energy, EEG-WM-6503.

Comes, G. D., L. D. Wakeley, and E. F. O'Neil, 1987. Laboratory-Tested Specimens of Concrete from Small Scale Seal Performance Tests at the Waste Isolation Pilot Plant, MP SL-87-9, prepared for Sandia National Laboratories, Albuquerque, New Mexico.

Cook, C. W., 1979. Instrumentation Development for the Waste Isolation Pilot Plant (WIPP) Borehole Plugging Program, SAND79-1902, Sandia National Laboratories, Albuquerque, New Mexico.

Cook, C. W., 1980. Impulse Radar Scanning of Intact Salt at the Avery Island Mine, SAND80-0632, Sandia National Laboratories, Albuquerque, New Mexico.

Cook, C. W., 1982. Underground Scanning with an Impulse Radar, SAND82-0470, Sandia National Laboratories, Albuquerque, New Mexico.

Cook, C. W. and E. S. Ames, 1979. "Borehole-Inclusion Stressmeter Measurements in Bedded Salt," SAND79-0377C, Sandia National Laboratories, Albuquerque, New Mexico, presented at Proceedings of the 20th U.S. National Symposium in Rock Mechanics, Austin, Texas.

Cook, C. W., and E. S. Ames, 1980. Borehole-Inclusion Stressmeter Measurements in Bedded Salt, SAND79-0377, Sandia National Laboratories, Albuquerque, New Mexico.

Cook, C. W., C. B. Kinabrew, P. L. Lagus, and R. D. Broce, 1980. Bell Canyon Test (BCT) Instrumentation Development, SAND80-0408C, Sandia National Laboratories, Albuquerque, New Mexico, presented at the International BHP Symposium, Columbus, Ohio.

Cook, C. W., and A. R. Sattler, 1979. Scanning of Intact Salt in the Avery Island Mine with Radar, Sandia National Laboratories, Albuquerque, New Mexico.

Cooley, C. H., and S. W. Butters, 1979. Pressure Time Effects on Permeability of Salt Cores, TR79-45, prepared for Sandia National Laboratories, Albuquerque, New Mexico.

Cooley, C. H., S. W. Butters, and A. H. Jones, 1978. Permeability of Rock Salt to Gases and Liquids, TR78-62, prepared for Sandia National Laboratories, Albuquerque, New Mexico.

Coons, W., A. Bertgstrom, P. Gnirk, M. Gray, B. Knecht, R. Pusch, J. Steadman, B. Stillborg, M. Tokonami, and M. Vaajasaari, 1987. State-of-the-Art Report on Potentially Useful Materials for Sealing Nuclear Waste Repositories, Stripa Project Report 87-12, Swedish Nuclear Fuel and Waste Management Co., Stockholm, Sweden.

Costin, L. S., 1979. Summary of Research and Development Activities in Support of Waste Acceptance Criteria for WIPP, SAND79-1305, Sandia National Laboratories, Albuquerque, New Mexico.

Costin, L. S., and W. R. Wawersik, 1980. Creep Healing of Fractures in Rock Salt, SAND80-0392, Sandia National Laboratories, Albuquerque, New Mexico.

D'Appolonia Consulting Engineers, Inc., 1979. Repository Sealing Design Approach--1979, ONWI-55, Office of Nuclear Waste Isolation, Columbus, Ohio.

D'Appolonia Consulting Engineers, Inc. 1981. Sealing Consideration for Repository Shafts in Bedded and Dome Salt, ONWI-255, Office of Nuclear Waste Isolation, Columbus, Ohio.

Dawson, P. R., 1979. Constitutive Models Applied in the Analysis of Creep of Rock Salt, SAND79-0137, Sandia National Laboratories, Albuquerque, New Mexico.

Dawson, P. R., and D. E. Munson, 1983. "Numerical Simulation of Creep Deformations Around a Room in a Deep Potash Mine," SAND82-0184J, Sandia National

Laboratories, Albuquerque, New Mexico, published in International Journal of Rock Mechanics & Mineral Science, Vol. 20, pp. 33-42.

Dawson, P. R., and J. R. Tillerson, 1977. Comparative Evaluations of the Thermomechanical Responses for Three High-Level Waste Canister Emplacement Alternatives, SAND77-0388, Sandia National Laboratories, Albuquerque, New Mexico.

Dawson, P. R., and J. R. Tillerson, 1978. Nuclear Waste Canister Thermally Induced Motion, SAND78-0566, Sandia National Laboratories, Albuquerque, New Mexico.

Dawson, P. R., and J. R. Tillerson, 1978. Salt Motion Following Nuclear Waste Disposal, SAND77-1226C, Sandia National Laboratories, Albuquerque, New Mexico, presented at the Conference of Evaluation and Prediction of Subsidence, Pensacola, Florida.

Deal, D. E., and J. B. Case, 1987. Brine Sampling and Evaluation Program, Phase I Report, DOE/WIPP-87-008, U.S. Department of Energy, Carlsbad, New Mexico.

Dennis, A. W., et al., 1978. Revised Concept for the Waste Isolation, SAND78-1429, Sandia National Laboratories, Albuquerque, New Mexico.

DOE (U.S. Department of Energy), 1980. "Statement of the Position of the United States Department of Energy in the Matter of Proposed Rulemaking on the Storage and Disposal of Nuclear Waste (NRC Waste Confidence Rulemaking)," DOE/NE-0007, Washington, D.C.

DOE (U.S. Department of Energy), 1981. Waste Isolation Pilot Plant Safety Assessment: DOE Report, Vol. 1.

DOE (U.S. Department of Energy), 1982. "Site Validation Program," DOE/WIPP-116, Revision 1, Waste Isolation Pilot Plant Project Office.

DOE (U.S. Department of Energy), 1983. The Defense Waste Management Plan, DOE/DP-0015, Washington, D.C.

DOE (U.S. Department of Energy), 1983. DOE Responses to the State of New Mexico's Comments on Summary of Results of the Evaluation of the WIPP Site and Preliminary Design Validation Program, DOE/WIPP-161, DOE/WIPP-174, Albuquerque, New Mexico.

DOE (U.S. Department of Energy), 1983. "Geotechnical Activities in the Exploratory Shaft," TME-3178.

DOE (U.S. Department of Energy), 1983. Results of Site Validation Experiments, Waste Isolation Pilot Plant (WIPP) Project, Southeastern New Mexico, Technical Support Contract DE-AC04-78AL05346, TME-3177 DOE/AL, Albuquerque, New Mexico.

DOE (U.S. Department of Energy), 1984. "Geotechnical Activities in the Waste Handling Shaft," WSTD-TME-038.

DOE (U.S. Department of Energy), 1984. "Radioactive Waste Management," DOE Order 5820.2.

DOE (U.S. Department of Energy), 1985. "Radioactive Waste Management," DOE Order, AL 5820.2.

DOE (U.S. Department of Energy), 1986. "Design Validation Final Report," WIPP/DOE 86-010, DOE Carlsbad, New Mexico.

DOE (U.S. Department of Energy), 1987. "Exhaust Shaft Grouting Program `C', Report for Week Ending June 21," WIPP, Carlsbad, New Mexico.

DOE (U.S. Department of Energy), 1987. Integrated Data Base for 1987: Spent Fuel and Radioactive Waste Inventories, Projections, and Characteristics, DOE/RW-0006, Rev. 3.

DOE (U.S. Department of Energy), 1988. Draft Radionuclide Source Term for the WIPP, DOE/WIPP-88-049, Carlsbad, New Mexico.

DOE (U.S. Department of Energy), 1988. "Final Safety Analysis Report, Waste Isolation Pilot Plant," draft, DOE/WIPP 87-013, Carlsbad, New Mexico.

DOE (U.S. Department of Energy), 1988. Integrated Data Base for 1988: Spent Fuel and Radioactive Waste Inventories, Projections, and Characteristics, DOE/RW-0006, Rev. 4.

DOE (U.S. Department of Energy), 1988. Long Range Master Plan for Defense Transuranic Waste Program, DOE/WIPP 88-028.

DOE (U.S. Department of Energy), 1988. "Panel One Monitoring Plan for the Waste Isolation Pilot Plant," draft, Carlsbad, New Mexico.

DOE (U.S. Department of Energy), 1988. "Safety Analysis and Review System," DOE/AL Order 5481.1B, Albuquerque, New Mexico.

DOE (U.S. Department of Energy), 1989. Draft Plan for the Waste Isolation Pilot Plant Test Phase: Performance Assessment and Operations Demonstration, DOE/WIPP 89-011, Carlsbad, New Mexico.

DOE (U.S. Department of Energy), 1989. Final Safety Analysis Report, Waste Isolation Pilot Plant, Carlsbad, New Mexico, draft, WP02-9, Rev. 0, Carlsbad, New Mexico.

DOE (U.S. Department of Energy), 1989. Final Safety Analysis Report, Waste Isolation Pilot Plant, draft, DOE/WIPP 89-XXX, Carlsbad, New Mexico.

DOE (U.S. Department of Energy), 1989. Integrated Data Base for 1989: Spent Fuel and Radioactive Waste Inventories, Projections, and Characteristics, DOE/RW-0006, Rev. 5.

DOE (U.S. Department of Energy), 1989. Radionuclide Source Term for the WIPP, 88-005, Carlsbad, New Mexico.

DOE (U.S. Department of Energy), 1989. "TRU Waste Acceptance Criteria for the Waste Isolation Pilot Plant," WIPP-DOE-069, Revision 3, U.S. Department of Energy, Carlsbad, New Mexico.

Duffy, T. A., 1980. The Salt Block I Test: Experimental Details and Comparison with Theory, Los Alamos Technical Associates, Inc. (LATA 102), SAND79-7050, Sandia National Laboratories, Albuquerque, New Mexico.

Dunn, L. A., 1981. Development Report for Contact-Handled Waste Container and Handling System, SAND81-0141, Sandia National Laboratories, Albuquerque, New Mexico.

Eaton, R. R., 1984. Three-Dimensional Thermal Analysis for the WIPP In Situ Test Room A Heater Configuration, SAND84-2220, Sandia National Laboratories, Albuquerque, New Mexico.

EEG (Environmental Evaluation Group), 1980. "Geotechnical Considerations for Radiological Hazard Assessment of WIPP: A Report of a Meeting Held on January 17-18, 1980."

EEG (Environmental Evaluation Group), 1980. Review Comments on the Report of the Steering Committee on Waste Acceptance Criteria for the Waste Isolation Pilot Plant, M. S. Little, EEG-4, Health and Environment Department, State of New Mexico, Santa Fe, New Mexico.

EEG (Environmental Evaluation Group), 1981. "Radionuclide Release, Transport and Consequence Modeling for WIPP: A Report of a Workshop Held on September 16-17, 1981", EEG-27, Santa Fe, New Mexico.

EEG (Enviromental Evaluation Group), 1983. "Review Comments on the Geotechnical Reports Provided by DOE to EEG Under the Stipulated Agreement through March 1, 1983", EEG-22, Santa Fe, New Mexico.

EEG (Environmental Evaluation Group), 1984. Radiation Shielding in the Hot Cell Facility at the Waste Isolation Pilot Plant: A Review, H. B. Knowles, EEG-28, Health and Environment Department, State of New Mexico, Santa Fe, New Mexico.

EEG (Environmental Evaluation Group), 1985. Evaluation of the Safety Analysis Report for the Waste Isolation Pilot Plant Project, M. S. Little, EEG-29, Health and Environment Department, State of New Mexico, Santa Fe, New Mexico.

EEG (Environmental Evaluation Group), 1985. Evaluation of the Waste Isolation Pilot Plant Classification of Systems, Structures, and Components, EEG-30, Submitted by the Tenera Corporation, Health and Environment Department, State of New Mexico, Santa Fe, New Mexico.

EEG (Environmental Evaluation Group), 1987. Exhaust Stack Monitoring Issuses at the Waste Isolation Pilot Plant, EEG-37, Health and Environment Department, State of New Mexico, Santa Fe, New Mexico.

EEG (Environmental Evaluation Group), 1988. A Critical Assessment of Continuous Air Monitoring Systems at the Waste Isolation Pilot Plant, J. C. Rodgers and J.W. Kenney, EEG-38, Health and Environment Department, State of New Mexico, Santa Fe, New

Mexico.

Ekren, E. B., 1974. Geologic and Hydrologic Considerations for Various Concepts of High-Level Radioactive Waste Disposal in Conterminous United States, U.S. Geological Survey Open-File Report 74-158.

Evans, B., 1983. "Alteration of Porosity and Permeability by Plastic Flow Processes in Rocks," No. 27664, Geological Society of America Abstracts with Programs, Vol. 15, No. 6.

Ewing, R. I., 1979. Test of a Radiant Heater in the Avery Island Salt Mine, Sandia National Laboratories, Albuquerque, New Mexico.

Ewing, R. I., 1981. Preliminary Moisture Release Experiment in a Potash Mine in Southeastern New Mexico, SAND81-1318, Sandia National Laboratories, Albuquerque, New Mexico.

Ewing, R. I., 1982. Environmental Gamma Ray Spectra from Selected Sites in Southeastern New Mexico, SAND82-0738, Sandia National Laboratories, Albuquerque, New Mexico.

Ewing, R. I., and A. R. Sattler, 1979. Measurement of Gamma Radiation Background at Avery Island Mine, Sandia National Laboratories, Albuquerque, New Mexico.

Ewing, R. I., and H. C. Shefelbine, 1980. Preliminary Moisture Release Experiment in the Mississippi Chemical Potash Mine, Sandia National Laboratories, Albuquerque, New Mexico.

Ewing, R. I., and H. C. Shefelbine, 1981. Test Plan for Three Heater Moisture Release Experiment, Sandia National Laboratories, Albuquerque, New Mexico.

Fenix & Scisson, Inc., 1978. WIPP 18, 19, 21, and 22 Field Engineering Summary, Vols. I and II, Sandia National Laboratories, Albuquerque, New Mexico.

Ferrall, C., and J. F. Gibbons (Civil Systems, Inc.), 1980. Core Study of Rustler Formation Over the WIPP Site, CSI-2055-03, SAND79-7110, Sandia National Laboratories, Albuquerque, New Mexico.

Fortney, D. R., and M. G. Seitz, 1981. Test Plan: Development of Dynamic Stream Simulation Methodology, Sandia National Laboratories, Albuquerque, New Mexico.

Fossum, A. F., 1986. Computer Calculations to Develop Quality Assurance (QA) Methodology for Nuclear Waste Repository Design Calculations: Isothermal Parallel Calculations, SAND85-7262, Sandia National Laboratories, Albuquerque, New Mexico.

Franke, C., 1987. "Excavation Effects and Fracture Mapping," Input to 1987 GFDAR, August 18 Memorandum to R. McKinney, International Technology Corporation WIPP Site, Carlsbad, New Mexico.

Gartling, D. K., 1981. MERLIN: A Computer Program to Transfer Data Between Finite Element Meshes, SAND81-0463, Sandia National Laboratories, Albuquerque, New Mexico.

George, O. L., Jr., 1980. Computer Thermal Modeling for the Salt Block II Experiment, SAND79-2250, Sandia National Laboratories, Albuquerque, New Mexico.

George, O. L., Jr., 1980. Predicted Temperature/Time Histories Resulting from the Burial of Nuclear Waste Canisters in Bedded Salt, SAND79-1773, Sandia National Laboratories, Albuquerque, New Mexico.

Gerstle, W. H., and A. K. Jones, 1986. Mechanical Properties of Crushed Salt/Bentonite Blocks, SAND86-0707, Sandia National Laboratories, Albuquerque, New Mexico.

Gillins, R. L., and M. M. Larsen, 1987. "Characterization of INEL Compactible Low-Level Wastes and Evaluation of Compactor Options," Proceedings of the Symposium on Waste Management at Tucson, Arizona, March, Vol. III, pp. 623-629.

Glass, R. S., 1981. Effects of Radiation on the Chemical Environment Surrounding Waste Canisters in Proposed Repository Sites and Possible Effects on the Corrosion Process, SAND81-1677, Sandia National Laboratories, Albuquerque, New,Mexico.

Glass, R. S., 1983. "Effect of Intermetallic Ti2Ni on the Electrochemistry of TiCode-12 in Hydrochloric Acid," SAND83-0053J, Sandia National Laboratories, Albuquerque, New Mexico, published in Electrochimica Acta, Vol. 28, No. 11, pp. 1507-13.

Glass, R. S., 1984. "Technical Note: The Interpretation of Weight-Loss Measurements for Alloys Showing Time-Delay Activation," SAND83-0053J, Sandia National Laboratories, Albuquerque, New Mexico, published in Corrosion, Vol. 40, No. 6, pp. 322-323.

Glass, R. S., 1985. "Passivation of Titanium by Molybdate Ion," SAND83-2541J, Sandia National Laboratories, Albuquerque, New Mexico, presented at Electrochemical Society Meeting, Fall 1983, published in Corrosion, Vol. 41, No. 2, pp. 89-93.

Glass, R. S., and Y. K. Hong, 1983. Mechanism of Transpassive Dissolution of Titanium-Molydenum Alloys, SAND82-2757A, Sandia National Laboratories, Albuquerque, New Mexico, presented at the Electrochemical Society Meeting, San Francisco, California.

Glass, R. S., and Y. K. Hong, 1984. "Transpassive Behavior of Titanium-Molybdenum Alloys in H2SO4," SAND83-1901J, Sandia National Laboratories, Albuquerque, New Mexico, published in Electrochimica Acta, Vol. 29, No. 10, pp. 1465-70.

Glass, R. S., J. A. Ruppen, and R. B. Diegle, 1982. Sensitization to Corrosion of Heat Treated TiCode-12, SAND82-1021A, Sandia National Laboratories, Albuquerque, New Mexico, presented at the Electrochemical Society Meeting, Detroit,,Michigan.

Gnirk, P. F., et al., 1981. State-of-the-Art Review of Brine Migration Studies in Salt, SAND81-7054, prepared for Sandia National Laboratories, Albuquerque, New Mexico, by RE/SPEC, Inc.

Gnirk, P. F., K. D. Mellegard, and H. Waldman, 1978. "Deformation and Strength Characteristics of Granulated Salt from Laboratory Consolidation Tests," RE/SPEC Technical Letter Memorandum, RSI-0020, P.O. Box 725, Rapid City, South Dakota, 57709.

Gnirk, P. F., W. G. Pariseau, J. E. Russel, W. R. Wawersik, and H. V. Hovland, 1973. Analysis and Evaluation of the Rock Mechanics Aspects of the Proposed Salt Mine Repository, RE/SPEC Report RSI-005.

Greenhalgh, W. O., R. J. Cash, and M. A. Christie, 1988. "Product Characteristics of TRU Waste Immobilized in Grout," Proceedings of the Symposium on Waste Management at Tucson, Arizona, February, Vol. II, pp. 309-316.

Griswold, G. B., 1977. Site Selection and Evaluation Studies of the Waste Isolation Pilot Plant (WIPP), Los Medanos, Eddy County, New Mexico, SAND77-0946, Sandia National Laboratories, Albuquerque, New Mexico.

Griswold, G., 1980. "Summary of Brine Flow," Geotechnical Consideration for Radiological Hazard Assessment of WIPP, State of New Mexico Environmental Evaluation Group Report EEG-6, pp. 16-22.

Gulick, C. W., Jr., 1978. Borehole Plugging Materials Development Program, SAND78-0715, Sandia National Laboratories, Albuquerque, New Mexico.

Gulick, C. W., Jr., 1979. Borehole Plugging Program---Plugging of ERDA No. 10 Drill Hole, SAND79-0789, Sandia National Laboratories, Albuquerque, New,Mexico.

Gulick, C. W., Jr., 1984. Test Plan: Plug Test Matrix, Sandia National Laboratories, Albuquerque, New Mexico.

Gulick, C. W., Jr., J. A. Boa, Jr., and A. D. Buck, 1980. Bell Canyon Test (BCT) Cement Development Report, SAND80-0358C, Sandia National Laboratories, presented at the International BHP Symposium, Columbus, Ohio.

Gulick, C. W., Jr., J. A. Boa, Jr., and A. D. Buck, 1980. Bell Canyon Test (BCT) Cement Grout Development Report, SAND80-1928, Sandia National Laboratories, Albuquerque, New Mexico.

Gulick, C. W., Jr., J. A. Boa, Jr., and A. D. Buck, 1982. Borehole Plugging Materials Development Program---Report 3, SAND81-0065, Sandia National Laboratories, Albuquerque, New Mexico.

Gulick, C. W., Jr., J. A. Boa, Jr., D. M. Walley, and A. D. Buck, 1980. Borehole Plugging Materials Development Program---Report No. 2, SAND79-1514, Sandia National Laboratories, Albuquerque, New Mexico.

Gulick, C. W., and L. D. Wakeley, 1988. "Reference Properties of Cement-Based Plugging and Sealing Materials for Waste Isolation Pilot Plant (WIPP)," SAND87-2817, (in preparation), Sandia National Laboratories, Albuquerque, New Mexico.

Hadley, G. R., 1981. Salt Block II: Brine Migration Modeling, SAND81-0438, Sandia National Laboratories, Albuquerque, New Mexico.

Hadley, G. R., and G. W. Faris, 1981. Revised Theory of Water Transport in Rock Salt, SAND80-2398, Sandia National Laboratories, Albuquerque, New Mexico.

Hansen, F. D., 1976. Experimental Consolidation of Granulated Rock Salt with Application to Sleeve Buckling, ORNL/Sub-42-69/21, prepared for U.S. Department of Energy by Oak

Ridge National Laboratory, Oak Ridge, Tennessee.

Hansen, F. D., 1977. Triaxial Quasi-Static Compression and Creep Behavior of Bedded Salt from Southeastern New Mexico, Technical Memorandum Report, RSI-0055, Sandia National Laboratories, Albuquerque, New Mexico.

Hansen, F. D., and P. F. Gnirk, 1975. Design Aspects of Radioactive Waste Repository; III Uniaxial Quasi-Static and Creep Properties of the Site Rock, Technical Memorandum Report RSI-0029, Sandia National Laboratories, Albuquerque, New Mexico.

Hansen, F. D., and K. D. Mellegard, 1977. Creep Behavior for Bedded Salt from Southeastern New Mexico at Elevated Temperatures, Technical Memorandum Report, RSI-0062, Sandia National Laboratories, Albuquerque, New Mexico.

Hansen, F. D., and K. D. Mellegard, 1980. Further Creep Behavior of Bedded Salt from Southeastern New Mexico at Elevated Temperature, SAND80-7114, Sandia National Laboratories, Albuquerque, New Mexico.

Henderson, J. T., 1982. Waste-Management QA Training and Motivation, SAND82-0358C, Sandia National Laboratories, Albuquerque, New Mexico, presented at 1982 ASQC Western Region Conference, Phoenix, Arizona.

Herrmann, W., and H. S. Lauson, 1981. Analysis of Creep Data for Various Natural Rock Salts, SAND81-2567, Sandia National Laboratories, Albuquerque, New Mexico.

Herrmann, W., and H. S. Lauson, 1981. Analysis of Stress Drop in Multistage Creep Experiments: Predictions of Exponential Creep Law, SAND81-1612, Sandia National Laboratories, Albuquerque, New Mexico.

Herrmann, W., and H. S. Lauson, 1981. Review and Comparison of Transient Creep Laws Used for Natural Rock Salt, SAND81-0738, Sandia National Laboratories, Albuquerque, New Mexico.

Herrmann, W., W. R. Wawersik, and H. S. Lauson, 1980. Analysis of Steady State Creep of Southeastern New Mexico Bedded Salt, SAND80-0558, Sandia National Laboratories, Albuquerque, New Mexico.

Herrmann, W., W. R. Wawersik, and H. S. Lauson, 1980. Creep Curves and Fitting Parameters for Southeastern New Mexico Bedded Salt, SAND80-0087, Sandia National Laboratories, Albuquerque, New Mexico.

Herrmann, W., W. R. Wawersik, and H. S. Lauson, 1980. A Model for Transient Creep of Southeastern New Mexico Rock Salt, SAND80-2172, Sandia National Laboratories, Albuquerque, New Mexico.

Herrmann, W., W. R. Wawersik, and S. T. Montgomery, 1982. Review of Creep Modeling for Rock Salt, SAND82-2178C, Sandia National Laboratories, Albuquerque, New Mexico, presented at the International Conference on Constitutive Laws for Engineering Materials: Theory and Application, Toronto, Canada.

Hill, L. R., 1980. "Characterization of a Site in Bedded Salt for Isolation of Radioactive

Wastes," in IAEA-SM243/38, 20, Underground Disposal of Radioactive Wastes, Vol. 1, pp. 269-287.

Hohlfelder, J. J., 1979. Measurement of Water Lost from Heated Geologic Salt, SAND79-0462, Sandia National Laboratories, Albuquerque, New Mexico.

Hohlfelder, J. J., 1980. Salt Block II: Description and Results, SAND79-2226, Sandia National Laboratories, Albuquerque, New Mexico.

Hohlfelder, J. J., 1981. Volatile Content of Rock Salt, SAND79-2349, Sandia National Laboratories, Albuquerque, New Mexico.

Hohlfelder, J. J., and G. R. Hadley, 1979. Laboratory Studies of Water Transport in Rock Salt, SAND79-1519, Sandia National Laboratories, Albuquerque, New Mexico.

Holcomb, D. J., 1985. "Results of the 55-Day Consolidation Test," internal memo to J. Stormont, 6332, Sandia National Laboratories, Albuquerque, New Mexico.

Holcomb, D. J., 1988. Time-Dependent Changes in Velocity and Attenuation in Salt at the Waste Isolation Pilot Plant, SAND87-3016, Sandia National Laboratories, Albuquerque, New Mexico, presented at the 29th U.S. Symposium on Rock Mechanics, Minneapolis, Minnesota.

Holcomb, D. J., and D. W. Hannum, 1982. Consolidation of Crushed Salt Backfill Under Conditions Appropriate to the WIPP Facility, SAND82-0630, Sandia National Laboratories, Albuquerque, New Mexico.

Holcomb, D. J., and A. K. Jones. Data Acquisition for the Rock Mechanics Laboratory, SAND83-0646, Sandia National Laboratories Report, Albuquerque, New Mexico.

Holcomb, D. J., and M. Shields, 1987. Hydrostatic Creep Consolidation of Crushed Salt with Added Water, SAND87-1990, Sandia National Laboratories, Albuquerque, New Mexico.

Holdoway, K. A., 1974. "Behavior of Fluid Inclusions in Salt During Heating and Irradiation," Fourth Symposium on Salt, Vol. 1, ed. A. H. Coogan, Northern Ohio Geological Society, Cleveland, Ohio, pp. 303-312.

Hong, Y. K., and R. S. Glass, 1983. Active-Passive Transition Study of Titanium-Molybdenum Alloys, SAND82-2884A, Sandia National Laboratories, Albuquerque, New Mexico, presented at the Electrochemical Society Meeting, San Francisco, California.

Hudson, T., 1987. Preliminary Personal Communication to J. S. Stormont (Sandia National Laboratories) on USGS Moisture and Density Measurements in Bedded Rock Salt at the WIPP Site.

Hughes, P. S., and L. D. Rigdon, 1980. Waste Isolation Pilot Plant Title I Operator Dose Calculations, LATA Report No. 90, SAND79-7107, Sandia National Laboratories, Albuquerque, New Mexico.

Hunter, T. O., 1979. Summary of Research and Development Activities in Support of Waste

Acceptance Criteria for WIPP, SAND79-1305, Sandia National Laboratories, Albuquerque, New Mexico.

Hunter, T. O., 1979. Summary of Results and Plans of the Experimental Program for the Waste Isolation Pilot Plant, SAND79-1305, Sandia National Laboratories, Albuquerque, New Mexico.

Hunter, T. O., 1979. Technical Issues of Nuclear Waste Isolation in the Waste Isolation Pilot Plant WIPP, SAND79-1117C, Sandia National Laboratories, Albuquerque, New Mexico.

Hunter, T. O., 1980. Borehole Plugging Test Activities in Southeastern New Mexico, SAND80-2462C, Sandia National Laboratories, Albuquerque, New Mexico, presented at 1980 NWTS Annual Information Meeting, Columbus, Ohio.

Hunter, T. O., 1980. The Perspective of Waste Isolation Research Issues and Assessment of Consequences for Radionuclide Release, SAND79-1076C, Sandia National Laboratories, Albuquerque, New Mexico, presented at the Materials Research Society International Symposium on the Scientific Basis for Nuclear Waste Management, Boston, Massachusetts, November 26-29, 1979.

Hunter, T. O., 1980. Role of Borehole Plugging Program in the Evaluation of the Waste Isolation Pilot Plant, SAND80-0502C, Sandia National Laboratories, Albuquerque, New Mexico, presented at the International BHP Symposium, Columbus, Ohio.

Hunter, T. O., 1982. Borehole and Facility Sealing Activities for the Waste Isolation Pilot Plant, SAND81-2034, Sandia National Laboratories, Albuquerque, New Mexico.

Hunter, T. O., 1982. Nuclear Waste Disposal: Technical Issues & Status, SAND82-2238C, Sandia National Laboratories, Albuquerque, New Mexico, presented at the Spent Fuel Management and Waste Disposal Seminar by the Institute for Nuclear Materials Management, Washington, D.C.

Hunter, T. O., 1982. A Summary of Research & Development for Disposal of U.S. Defense Wastes in the Waste Isolation Pilot Plant (WIPP), SAND82-2313C, Sandia National Laboratories, Albuquerque, New Mexico.

Hunter, T. O., 1983. The WIPP Research and Development Program: Providing the Technical Basis for Defense Waste Disposal, SAND83-0178C, Sandia National Laboratories, Albuquerque, New Mexico.

Hunter, T. O., 1984. Engineering and Construction Management Strategies on the WIPP, SAND83-2118A, Sandia National Laboratories, Albuquerque, New Mexico, presented at the Waste Management 1984 Conference, Tucson, Arizona.

Hunter, T. O., C. L. Christensen, R. V. Matalucci, W. R. Wawersik, and M. A. Molecke, 1980. Summary of Research and Development Activities Related to the US/FRG Cooperative Program on Nuclear Waste Mangement, SAND81-1204C, Sandia National Laboratories, Albuquerque, New Mexico, presented at the US/FRG Bilateral Meeting, Bonn, Federal Republic of Germany.

Intera Environmental Consultants, Inc., 1978. Radiological Waste Storage Analysis, Intera Environmental Consultants, Inc., prepared for Sandia National Laboratories, Albuquerque, New Mexico.

Intera Environmental Consultants, Inc., 1981. Consequence Assessment of Hydrological Communications through Borehole Plugs, SAND81-7164, Sandia National Laboratories, Albuquerque, New Mexico.

INTERA Technologies, Inc., 1985. Field Operations Plan for the H-8 Hydropad Pumping Test at the WIPP Site, Sandia National Laboratories, Albuquerque, New Mexico.

INTERA Technologies, Inc., 1985. Field Operations Plan for the Southern Long-Term Multipad Interference Test at the WIPP Site, Sandia National Laboratories, Albuquerque, New Mexico.

INTERA Technologies, Inc., 1986. Field Operations Plan for the H-7 Hydropad Pumping Test at the WIPP, Sandia National Laboratories, Albuquerque, New Mexico.

INTERA Technologies, Inc., 1986. Field Operations Plan for Hydrologic Testing at P-14, P-15, P-17, H-14, H-15, ERDA-9 and Cabin Baby-1, Sandia National Laboratories, Albuquerque, New Mexico.

INTERA Technologies, Inc., 1986. Field Operations Plan for Well Development and Pumping Tests at DOE-2, and WIPP-12, WIPP-13, WIPP-18, WIPP-19, WIPP-21, and WIPP-2A, Sandia National Laboratories, Albuquerque, New Mexico.

INTERA Technologies, Inc., 1987. Field Operations Plan for Monitoring of Ground-Water Observation Wells at the Waste Isolation Pilot Plant (WIPP) Site, Sandia National Laboratories, Albuquerque, New Mexico.

INTERA Technologies, Inc., 1987. Field Operations Plan for the Northern Long-Term Multipad Interference Test at the WIPP Site, Sandia National Laboratories, Albuquerque, New Mexico.

INTERA Technologies, Inc., 1987. Field Operations Plan for Permeability Testing in the Waste-Handling Shaft at the Waste Isolation Pilot Plant (WIPP) Site, Sandia National Laboratories, Albuquerque, New Mexico.

INTERA Technologies, Inc., 1987. Field Operations Plan for Well Development and Hydrologic Testing at H-1, H-12, P-18 and WIPP, Sandia National Laboratories, Albuquerque, New Mexico.

International Atomic Energy Agency, 1977. Site Selection Factors for Repositories of Solid High-Level and Alpha-Bearing Waste in Geological Formations, Technical Report Series No. 177, Vienna, Austria.

IT Corporation, 1984. Assessment of Crushed Salt Consolidation and Fracture Healing Processes in a Nuclear Waste Repository in Salt, BMI/ONWI-546, prepared for the Office of Nuclear Waste Isolation, Columbus, Ohio.

IT Corporation, 1987. Laboratory Investigation of Crushed Salt Consolidation and Fracture

Healing, BMI/ONWI-631, Office of Nuclear Waste Isolation, Columbus, Ohio.

Jacobson, A., and R. Pusch, 1977. Deposition of High-Level Radioactive Waste Products in Boreholes with Buffer Substance, KBS Report 03, KBS, Stockholm, Sweden.

Jenks, G. H., 1979. Effects of Temperature, Temperature Gradients, Stress, and Irradiation on Migration of Brine Inclusions in a Salt Repository, ORNL-5526, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Kelsall, P. C., J. B. Case, D. Meyer, and W. E. Coons, 1982. Schematic Designs for Penetration Seals for a Reference Repository in Bedded Salt, ONWI-405, Office of Nuclear Waste Isolation, Columbus, Ohio.

Kent, D., and W. R. Wawersik, 1976. "Laboratory Velocity Measurements on Rock Salt Core Samples," interoffice memo, Sandia National Laboratories, Albuquerque, New Mexico.

Key, S. W., C. M. Stone, and R. D. Krieg, 1980. Dynamic Relaxation Applied to the Quasi-Static, Large Deformation, Inelastic Response of Axisymmetric Solids, SAND80-1605C, Sandia National Laboratories, Albuquerque, New Mexico, proceedings of U.S. European Workshop, Non-Linear Finite Element Analysis and Structural Mechanics, West Germany.

Kircher, J. F., R. G. Easterling, N. Hubbard, J. Krager, N. J. Magnani, D. P. Moak, R. E. Westerman, B. J. Woodand, and M. A. Molecke (U.S. Department of Energy), 1981. Nuclear Waste Package Material Degeneration Modes and Accelerated Testing, DOE/NWTS-13.

Knorovsky, G. A., and T. J. Headley, 1985. Evaluation of Titanium Alloy Resistance Upset Welds for Defense High-Level Waste Canisters, SAND85-0735, Sandia National Laboratories, Albuquerque, New Mexico.

Kosiewicz, S. T. (Los Alamos Scientific Laboratory), 1980. "Cellulose Thermally Decomposes at 70°C," Thermochimica Acta, Vol. 40, pp. 319-326.

Kosiewicz, S. T. (Los Alamos Scientific Laboratory), 1980. "Gas Generation from the Alpha Radiolysis of Bitumen," Nuclear and Chemical Waste Management, Vol. 2, pp. 139-141.

Kosiewicz, S. T. (Los Alamos Scientific Laboratory), 1981. "Gas Generation from Organic Transuranic Wastes, I: Alpha Radiolysis at Atmospheric Pressure," Nuclear Technology, Vol. 54, No. 1, pp. 92-99.

Kosiewicz, S. T., and A. Zerwekh (Los Alamos Scientific Laboratory), 1979. "Transuranic Waste Degradation Under Waste Isolation Pilot Plant Conditions," published in the Transactions of 1979 Winter Meeting, American Nuclear Society 33, San Francisco, California, p. 426.

Kosiewicz, S. T., A. Zerwekh, and B. Barraclough (Los Alamos Scientific Laboratory), 1978-80. Studies of Transuranic Waste Storage Under Conditions Expected in the Waste Isolation Pilot Plant (WIPP), Progress Reports: LA-7582-PR, LA-7649-PR, LA-7775-PR, LA-7931-PR, LA-8106-PR, LA-8224-PR, LA-8345-PR.

Koski, J. A., and D. F. McVey, 1981. Application of Parameter Estimation Techniques to Thermal Conductivity Probe Data Reduction, SAND80-2601C, Sandia National Laboratories, Albuquerque, New Mexico, presented at the International Joint Conferences on Thermophysical Properties, Gaithersburg, Maryland.

Krieg, R. D., 1980. A Unified Creep-Plasticity Model for Halite, SAND80-1195, Sandia National Laboratories, Albuquerque, New Mexico, presented at the Mechanical Testing for Deformation Model Development Symposium by ASTM, Bal Harbor, Florida.

Krieg, R. D., 1983. "Implementation of Creep Equations for a Metal Into a Finite Element Computer Program," Computer Methods for Nonlinear Solids and Structural Mechanics, eds. S. Atluri and N. Perrone, ASME Special Publication AMD, Vol. 54.

Krieg, R. D., 1984. Reference Stratigraphy and Rock Properties for the Waste Isolation Pilot Plant (WIPP) Project, SAND83-1908, Sandia National Laboratories, Albuquerque, New Mexico.

Krieg, R. D., and T. O. Hunter, 1980. Geomechanics Activities Conducted at Sandia National Laboratories in Support of the Waste Isolation Pilot Plant, SAND80-1592C, Sandia National Laboratories, Albuquerque, New Mexico, presented at ONWI/LBL Workshop on Thermomechanical Hydrochemical Modeling for a Hardrock Waste Repository, Berkeley, California.

Krieg, R. D., H. S. Morgan, and T. O. Hunter, 1980. Second Benchmark Problem for WIPP Structural Computations, SAND80-1331, Sandia National Laboratories, Albuquerque, New Mexico.

Krieg, R. D., C. M. Stone, and S. W. Key, 1981. Comparisons of the Structural Behavior of Three Storage Room Designs for the WIPP Project, SAND80-1629, Sandia National Laboratories, Albuquerque, New Mexico.

Krumhansl, J. L., 1982. "Mineralogic Stability of a Bentonite Backfill in a Bedded Salt Repository," SAND82-0951A, Sandia National Laboratories, Albuquerque, New Mexico, Geological Society of America Abstracts, Vol. 14, No. 7.

Krumhansl, J., 1984. Observations Regarding the Stability of Bentonite-Based Backfills in a High-Level Waste Repository in Rock Salt, SAND83-1293, Sandia National Laboratories, Albuquerque, New Mexico.

Labreche, D. A., and L. L. Van Sambeek, 1988. "Analysis of Data from Expansive Salt-Water Concrete Seals in Series B Small Scale Seal Performance Tests," SAND87-7155, (in preparation), Sandia National Laboratories, Albuquerque, New Mexico.

Lambert, S. J., 1980. Mineralogical Aspects of Fluid Migration in the Salt Block II Experiment, SAND79-2423, Sandia National Laboratories, Albuquerque, New Mexico.

Lambert, S. J., 1980. Recommended Analysis Plan for the Borehole Plugging Program Potash Core Test, SAND80-0812, Sandia National Laboratories, Albuquerque, New Mexico.

Lambert, S. J., 1980. A Strategy for Evaluating the Long-Term Stability of Hole-Plugging

Materials in Their Geological Environments, SAND80-0359C, Sandia National Laboratories, Albuquerque, New Mexico, presented at the International Workshop on Borehole and Shaft Plugging, Columbus, Ohio.

Lambert, S. J., and H. C. Shefelbine, 1980. A Strategy for Investigation of Fluid Migration in Evaporites (Waste Isolation Pilot Plant (WIPP), SAND79-1889, Sandia National Laboratories, Albuquerque, New Mexico.

Long, G. J., and Associates, Inc., 1977. Waste Isolation Pilot Plant (WIPP), Los Medanos Area, Eddy and Lea Counties, New Mexico, Final Report submitted to Sandia National Laboratories, Albuquerque, New Mexico.

Longsine, D. E., E. J. Bonano, and C. P. Harlan, 1987. User's Manual for the NEFTRAN Computer Code, SAND86-2405 and NUREG/CR-4766, Sandia National Laboratories, Albuquerque, New Mexico.

Looper, M. G., and D. L. Charlesworth, 1988. "Demonstration of Remotely Operated Waste Size Reduction and Material Handling Equipment," Proceedings of the Symposium on Waste Management at Tucson, Arizona, March 1987, University of Arizona.

Loukota, J. J., 1986. Software to Obtain Polonomial Coefficients from Gage Calibration Data, SAND85-1796, Sandia National Laboratories, Albuquerque, New Mexico.

Loukota, J. J., 1986. Technical Manual Data Reduction Software for Gage Comparison Experiments in the WIPP, SAND85-1795, Sandia National Laboratories, Albuquerque, New Mexico.

Lucas, J. P., 1984. Analysis of Rock Bolt Material Failures at the WIPP Site, SAND84-0224, Sandia National Laboratories, Albuquerque, New Mexico.

Magnani, N. J., 1983. "Corrosion Resistant Canisters for Nuclear Waste Isolation," SAND82-1384C, Sandia National Laboratories, Albuquerque, New Mexico, presented at Materials Research Society, Boston, Massachusetts, published in Scientific Basis for Nuclear Waste Management VI, Vol. 15, ed. D. Brookins, North-Holland, New York.

Magnani, N. J., and J. W. Braithwaite, 1980. "Corrosion Resistant Metallic Canisters for Nuclear Waste Isolation," SAND79-1429C, Sandia National Laboratories, Albuquerque, New Mexico, presented at Materials Research Society, Boston, Massachusetts, published in Scientific Basis for Nuclear Waste Management, Vol. 2, ed. C. J. Northrup, Plenum Press, New York.

Matalucci, R. V., 1982. Simulated-Waste Experiments Planned for Waste Isolation Pilot Plant (WIPP), SAND82-0547, Sandia National Laboratories, Albuquerque, New Mexico.

Matalucci, R. V., 1983. Field Evaluation of Strain Gaged Stressmeter and USBM Hydraulic Pressure Cells at WIPP, Sandia National Laboratories, Albuquerque, New Mexico.

Matalucci, R. V., 1984. The Waste Isolation Pilot Plant (WIPP) In Situ Testing Program, SAND84-0661A, Sandia National Laboratories, Albuquerque, New Mexico, presented at the Geotechnical Group, New Mexico, Section of the American Society of Civil Engineers.

Matalucci, R. V., 1986. Planning, Developing, and Organizing In Situ Tests for the Waste Isolation Pilot Plant (WIPP), SAND84-1390, Sandia National Laboratories, Albuquerque, New Mexico, presented at the Second Conference on Mechanical Behavior of Salt, Hannover, Federal Republic of Germany.

Matalucci, R. V., 1987. Summary Report: The Integrated In Situ Testing Program for the Waste Isolation Pilot Plant (WIPP), SAND86-2716, Sandia National Laboratories, Albuquerque, New Mexico.

Matalucci, R. V., 1987. Test Plan: WIPP Air Intake Shaft Performance Tests, Sandia National Laboratories, Albuquerque, New Mexico.

Matalucci, R., J. R. Ball, and L. K. Shepard, 1987. User's Manual for the UNDERDOG Data Reduction Software, SAND87-7149, Sandia National Laboratories, Albuquerque, New Mexico.

Matalucci, R., C. L. Christensen, T. O. Hunter, M. A. Molecke, and D. M. Munson, 1979. Waste Isolation Pilot Plant (WIPP) Research and Development Program: In Situ Testing Plan, SAND81-2628, Sandia National Laboratories, Albuquerque, New Mexico.

Matalucci, R. V., and T. O. Hunter, 1981. Geomechanical Applications for the Waste Isolation Pilot Plant (WIPP) Project, SAND81-1203C, Sandia National Laboratories, Albuquerque, New Mexico, presented at the First Conference on the Mechanical Behavior of Salt, University Park, Pennsylvania.

Matalucci, R., H. S. Morgan, and R. D. Krieg, 1981. The Role of Bench-marking in Assessing the Capability to Predict Room Response in Bedded Salt Repositories, SAND81-1293C, Sandia National Laboratories, Albuquerque, New Mexico, presented at the OECD Nuclear Energy Workshop on Near-Field Phenomena in Geologic Repositories for Radioactive Waste, Seattle, Washington.

Matalucci, R. V. and F. G. Yost, 1987. The Scientific Program at the Waste Isolation Pilot Plant, SAND85-1699, Sandia National Laboratories, Albuquerque, New Mexico.

Mattus, A. J., R. D. Doyle, and D. P. Swindlehurst, 1988. "Asphalt Solidification of Mixed Wastes," Proceedings of the Symposium on Waste Management at Tucson, Arizona, February, Vol. I, pp. 229-234.

Maxwell, D., K. K. Wahi, and B. Dial, 1980. The Thermomechanical Response of WIPP Repositories, SAND79-7111, Sandia National Laboratories, Albuquerque, New Mexico.

McClain, W. C. et al., 1975. Geologic Disposal Program Semi-Annual Report for Period Ending March 31, 1975, Chemical Technology Division, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

McIlmoyle, J. T., R. V. Matalucci, and C. O. Harvey, 1987. The Data Acquisition System for the Waste Isolation Pilot Plant In Situ Tests, SAND86-1031, Sandia National Laboratories, Albuquerque, New Mexico.

McIlmoyle, J. T., R. V. Matalucci, and H. C. Ogden, 1986. "Developing a Data Acquisition

System for the Waste Isolation Pilot Plant In Situ Tests," SAND85-2505C, Sandia National Laboratories, Albuquerque, New Mexico, Proceedings of the 32nd International Instrumentation Symposium, pp. 445-456, Seattle, Washington.

McKinley, K. B., and J. D. McKinney, 1978. "Initial Drum Retrieval-Final Report," TREE-1286, August, Idaho National Engineering Laboratory, Idaho.

McKinney, R. F., and R. S. Newton, 1983. "Site Validation Field Program Plan," Revision 1.2, prepared for the U.S. Department of Energy by the WIPP Technical Support Contractor.

McTigue, D. F., 1985. A Linear Theory for Porous Thermoelastic Materials, SAND85-1159, Sandia National Laboratories, Albuquerque, New Mexico.

McTigue, D. F., and E. J. Nowak, 1987. Brine Transport Studies in the Bedded Salt of the Waste Isolation Pilot Plant (WIPP), SAND87-1274C, Sandia National Laboratories, Albuquerque, New Mexico, presented at the 1987 Fall Meeting of the Materials Research Society, Boston, Massachusetts.

McTigue, D. F., and E. J. Nowak, 1988. "Brine Transport in the Bedded Salt of the Waste Isolation Pilot Plant (WIPP) Field Measurements and a Darcy Flow Model," Scientific Basis for Nuclear Waste Management X, Materials Research Society Symposia Proceedings, Vol. 84, Materials Research Society, Pittsburgh, Pennsylvania.

McVey, D. F., 1981. Analysis of Data from Line Source Thermal Conductivity Measurements Taken In Situ and Dome Salt at the Avery Island Mine, SAND81-1232, Sandia National Laboratories, Albuquerque, New Mexico.

Mercer, J. W., 1986. Field Operations Plan of Sandia National Laboratories WIPP Site Investigations Hydrologic Test Holes H-14 and H-15, Sandia National Laboratories, Albuquerque, New Mexico.

Mercer, J. W., 1987. Field Operations Plan of Sandia National Laboratories WIPP Site Investigations Hydrologic Test Holes H-3B4 and H-3B5, Sandia National Laboratories, Albuquerque, New Mexico.

Mercer, J. W., 1987. Field Operations Plan of Sandia National Laboratories WIPP Site Investigations Hydrologic Test Holes H-17 and H-18, Sandia National Laboratories, Albuquerque, New Mexico.

Mercer, D. D., P. L. Baker, J. S. Cockman, N. T. Fischer, D. T. Flynn, J. P. Harvill, K. L. Knudtsen, and E. T. Louderbough, 1989. Operational Environmental Monitoring Plan for the Waste Isolation Pilot Plant, DOE/WIPP 88-025, Carlsbad, New Mexico.

Mercer, J. W., and R. L. Beauheim, 1987. Field Operations Plan of Sandia National Laboratories WIPP Site Investigations Hydrologic Test Hole H-16, Sandia National Laboratories, Albuquerque, New Mexico.

Meyer, D., and J. J. Howard, eds., 1983. Evaluation of Clays and Clay Minerals for Application to Repository Sealing, ONWI-486, Office of Nuclear Waste Isolation, Columbus, Ohio.

Miller, J. D., C. M. Stone, and L. J. Branstetter, 1982. Reference Calculations for Underground Rooms of the WIPP, SAND82-1176, Sandia National Laboratories, Albuquerque, New Mexico.

Mishima, J., and L. C. Schwendiman, 1973. Fractional Airborne Release of Uranium (Representing Plutonium) During the Burning of Contaminated Wastes, Battelle Pacific Northwest Laboratory, BNWL-1730.

Mogford, J. A., and W. L. Garner, 1977. Waste Isolation Pilot Plant (WIPP), Southeastern New Mexico, SAND77-0701, Sandia National Laboratories, Albuquerque, New Mexico.

Molecke, M. A., 1977. The Waste Isolation Pilot Plant High-Level Waste Experimental Program, SAND77-1301A, Sandia National Laboratories, Albuquerque, New Mexico, presented at the American Nuclear Society National Meeting, San Francisco, California, published in Transactions of the ANS, Vol. 27, p. 440.

Molecke, M. A., 1977. "Waste Isolation Pilot Plant (WIPP) Transuranic Waste Experimental Program," SAND77-1300, Sandia National Laboratories, Albuquerque, New Mexico, published in Transactions of the ANS, Vol. 27, p. 441.

Molecke, M. A., 1977. "WIPP In Situ Experimental Program for HLW," Sandia National Laboratories, Albuquerque, New Mexico, published in Proceedings of the NWTS Program Conference on Waste-Rock Interactions, Pennsylvania State University, Y/OWI/SUB-77/142.

Molecke, M. A., 1978. Waste Isolation Pilot Plant Transuranic Wastes Experimental Characterization Program: Executive Summary, SAND78-1356, Sandia National Laboratories, Albuquerque, New Mexico.

Molecke, M. A., 1979. Gas Generation from Transuranic Waste Degradation: Data Summary of Interpretation, SAND79-1245, Sandia National Laboratories, Albuquerque, New Mexico.

Molecke, M. A., 1979. Gas Generation from Transuranic Waste Degradation: Interim Assessment, SAND79-0117, Sandia National Laboratories, Albuquerque, New Mexico.

Molecke, M. A., 1979. "WIPP Waste Characterization and Experimental Programs," Sandia National Laboratories, Albuquerque, New Mexico, published in Proceedings of the Task 2 Workshop Waste Isolation Safety Assessment Program, D. J. Bradley, Chairman, Seattle, Washington, PNL-SA-7992, pp. 205-229.

Molecke, M. A., 1980. Gas Generation from Transuranic Waste Degradation, SAND79-0911C, Sandia National Laboratories, Albuquerque, New Mexico, presented at Materials Research Society, Boston, Massachusetts, November 1979, published in Scientific Basis for Nuclear Waste Management, Vol. 2, Plenum Press, New York.

Molecke, M. A., 1981. "Degradation of Transuranic-Contaminated Wastes Under Geologic Isolation Conditions," SAND79-2216, Sandia National Laboratories, Albuquerque, New Mexico, published in Management of Alpha-Contaminated Wastes, Proceedings of an International Symposium on the Management of Alpha-Contaminated Wastes, International Atomic Energy Commission, Vienna, Austria, 1983, IAEA-SM-246/37, pp.

653-666.

Molecke, M. A., 1983. A Comparison of Brines Relevant to Nuclear Waste Experimentation, SAND83-0516, Sandia National Laboratories, Albuquerque, New Mexico.

Molecke, M. A., 1984. Test Plan: Waste Package Performance Technology Experiments for Simulated DHLW, Sandia National Laboratories, Albuquerque, New Mexico.

Molecke, M. A., 1984. WIPP Waste Package Testing on Simulated DHLW: Emplacement, SAND84-1263A, Sandia National Laboratories, Albuquerque, New Mexico, presented at MRS Symposium, Boston, Massachusetts.

Molecke, M. A., 1985. "WIPP Waste Package Testing on Simulated DHLW: Emplacement," SAND84-1263C, Sandia National Laboratories, Albuquerque, New Mexico, presented at Materials Research Sumposium, Boston, Massachusetts, published in Scientific Basis for Nuclear Waste Management VIII, p. 44.

Molecke, M. A., 1986. "Overview of Waste Package Performance Experiments in WIPP," SAND86-2024C, Sandia National Laboratories, Albuquerque, New Mexico, presented at the Third International Symposium on Ceramics in Nuclear Waste Management, Chicago, Illinois, published in Advances in Ceramics (20).

Molecke, M. A., 1986. Test Plan: WIPP CH and RH TRU Waste Technology Experiments, Sandia National Laboratories, Albuquerque, New Mexico.

Molecke, M. A., 1986. "WIPP Waste Package Performance Testing on Simulated DHLW: Early Data," SAND85-0559C, Sandia National Laboratories, Albuquerque, New Mexico, presented at the High-Level Waste Meeting, Pasco, Washington, published in High-Level Nuclear Waste Disposal, ed. H. C. Burkholder, Battelle Press.

Molecke, M. A., 1988. WIPP/SRI International In Situ Testing Program, SAND87-2654A, Sandia National Laboratories, Albuquerque, New Mexico, presented at Waste Management 1988, Tucson, Arizona.

Molecke, M.A., 1989. Test Plan: WIPP Bin-Scale CH TRU Waste Tests, draft, May, Sandia National Laboratories, Albuquerque, New Mexico.

Molecke, M. A., 1989. Test Plan: WIPP in Situ Room-Scale CH TRU Waste Tests, draft, June, Sandia National Laboratories, Albuquerque, New Mexico.

Molecke, M. A., 1989. "Test Plan: WIPP In Situ Room-Scale CH TRU Waste Tests," peer review draft, July, Sandia National Laboratories, Albuquerque, New Mexico.

Molecke, M. A., 1989. "WIPP Bin-Scale CH TRU Waste Tests," review draft, August, Sandia National Laboratories, Albuquerque, New Mexico.

Molecke, M. A., and L. Abrego, 1980. "Status of Sandia HLW Canister/Overpack Program Studies," SAND80-2191C, Sandia National Laboratories, Albuquerque, New Mexico, presented at 1980 NWTS Annual Information Meeting, Columbus, Ohio, published in Proceedings of the National Waste Terminal Storage Program Information Meeting, ONWI-212, p. 230.

Molecke, M. A., and R. Beraun, 1986. WIPP Simulated DHLW Tests: Status and Initial In Situ Backfill Thermal Conductivities, SAND86-0489C, Sandia National Laboratories, Albuquerque, New Mexico, presented at the Waste Management 1986 Meeting.

Molecke, M. A., D. J. Bradley, and J. W. Shade, 1982. "PNL-Sandia HLW Package Interactions Test: Phase One," SAND80-1442C, Sandia National Laboratories, Albuquerque, New Mexico, published in Scientific Basis for Nuclear Waste Management, Vol. 6, ed. S. V. Topp, North-Holland, New York.

Molecke, M. A., and T. Brandshang, 1986. A Comparison of Measured and Predicted Stresses on Simulated DHLW Package Tests Emplaced in WIPP Bedded Salt, SAND86-1165, Sandia National Laboratories, Albuquerque, New Mexico.

Molecke, M. A., and R. V. Matalucci, 1984. Preliminary Requirements for the Emplacement and Retrieval of Defense High-Level Waste (DHLW) Tests in the Waste Isolation Pilot Plant (WIPP), SAND84-1444, Sandia National Laboratories, Albuquerque, New Mexico.

Molecke, M. A., and G. H. Miller, 1981. Waste Package Materials Test in SENM Salt, CFT/WPMT/CIA, Sandia National Laboratories, Albuquerque, New Mexico.

Molecke, M. A., and D. E. Munson, 1987. Test Plan Appendix: WIPP Simulated RH TRU Waste Add-On Tests, Sandia National Laboratories, Albuquerque, New Mexico.

Molecke, M. A., and E. J. Nowak, 1980. Applicability of a Backfill-Getter Barrier for Radioactive Waste Isolation: Chemical and Physical Functions, SAND80-0838A, Sandia National Laboratories, Albuquerque, New Mexico, presented at American Chemical Society Annual Meeting, San Francisco, California.

Molecke, M. A., and E. J. Nowak, 1980. "Status of Sandia Backfill-Getter Development Studies," SAND80-2190C, Sandia National Laboratories, Albuquerque, New Mexico, presented at 1980 NTWTS Annual Information Meeting, Columbus, Ohio, published in Proceedings of the National Waste Terminal Storage Program Information Meeting, ONWI-22, p. 212.

Molecke, M., J. A. Ruppen, and R. B. Diegle, 1983. "Materials for High-Level Waste Canister/Overpacks in Salt Formations," SAND82-0429, Sandia National Laboratories, Albuquerque, New Mexico, published in Nuclear Technology, Vol. 63, pp. 476-506.

Molecke, M. A., D. W. Schaefer, R. S. Glass, and J. A. Ruppan, 1983. "Sandia HLW Canister/Overpack Studies Applicable for a Salt Repository," SAND81-1584, Sandia National Laboratories, Albuquerque, New Mexico, published in Advances in Science and Technology of the Management of High-Level Nuclear Waste, Vol. 1, ed. P. Hofmann, Battelle Press.

Molecke, M. A., and T. M. Torres, 1984. "The Waste Package Materials Field Test in S. E. New Mexico Salt," SAND83-1516C, Sandia National Laboratories, Albuquerque, New Mexico, presented at Materials Research Society Symposium, Boston, Massachusetts, published in Scientific Basis for Nuclear Waste Management, Vol. 26, pp. 69-76.

Molecke, M. A., and J. A. Van Den Avyle, 1987. FY 1985 Annual Summary of WIPP Waste Package Corrosion and Metallurgy Laboratory Research, SAND86-2636, Sandia National Laboratories, Albuquerque, New Mexico.

Molecke, M. A., and G. G. Wicks, 1986. Test Plan: WIPP Materials Interface Interactions Test (MIIT), Sandia National Laboratories, Albuquerque, New Mexico.

Monsees, J. E., and D. E. Munson, 1982. Design Considerations for a Nuclear Waste Repository in Salt, Tunneling Technology, Sandia National Laboratories, Albuquerque, New Mexico, presented at National Academy of Sciences.

Montgomery, S. T., 1981. Implementation of a Transient Creep Model for Natural Rock Salt as a Material-Response Subroutine for SANCHO, SAND81-1163, Sandia National Laboratories, Albuquerque, New Mexico.

Moody, N. R., and S. L. Robinson, 1982. The Effect of Hydrogen on TiCode-12 Ti-0.3 Mo-0.8Ni, SAND82-8705A, Sandia National Laboratories, Albuquerque, New Mexico.

Moody, N. R., and S. L. Robinson, 1982. "Internal Hydrogen Effects in TiCode-12 Canister/Overpack Materials," SAND82-8732A, presented at the Materials Research Society, Boston, Massachusetts, published in Scientific Basis for Nuclear Waste Management VI, Vol. 15, ed. D. Brookins, North-Holland, New York.

Moody, N. R., and S. L. Robinson, 1986. Compositional and Microstructural Effects on Hydrogen Embrittlement of ASTM Grade 12 Titanium (T1 0.3 Mo---0.8 Ni), SAND85-8247, Sandia National Laboratories, Albuquerque, New Mexico.

Moody, N. R., S. L. Robinson, and F. A. Greulich, 1985. Hydrogen Effects on TiCode-12 Properties and Microstructure, SAND85-8203, Sandia National Laboratories, Albuquerque, New Mexico.

Morgan, H. S., 1981. MARC Calculations for the Second WIPP Structural Benchmark Problem, SAND81-0925, Sandia National Laboratories, Albuquerque, New Mexico.

Morgan, H. S., 1982. Analysis of Borehole Inclusion Stress Measurement Concepts Proposed for Use in the Waste Isolation Pilot Plant In Situ Testing Program, SAND82-1192, Sandia National Laboratories, Albuquerque, New Mexico.

Morgan, H. S., 1984. Analysis of Borehole Inclusion Stress Measurement Concepts Proposed for Use in the Waste Isolation Pilot Plant (WIPP), SAND83-2342C, Sandia National Laboratories, Albuquerque, New Mexico, presented at the 25th U.S. Symposium on Rock Mechanics, Northwestern University, Evanston, Illinois.

Morgan, H. S., R. D. Krieg, and R. V. Matalucci, 1981. Comparative Analysis of Nine Structural Codes Used in the Second WIPP Benchmark Problem, SAND81-1389, Sandia National Laboratories, Albuquerque, New Mexico.

Morgan, H. S., and C. M. Stone, 1985. Pretest Reference Calculation for the 18-W/M2 Mockup for Defense High-Level Waste (WIPP Room A In Situ Experiment), SAND85-0807, Sandia National Laboratories, Albuquerque, New Mexico.

Morgan, H. S., and C. M. Stone, 1985. Pretest Reference Calculation for the Heated Axisymmetric Pillar (WIPP Room H In Situ Experiment), SAND85-1146, Sandia National Laboratories, Albuquerque, New Mexico.

Morgan, H. S., and C. M. Stone, 1985. Pretest Reference Calculation for the Overtest for Simulated Defense High-Level Waste (WIPP Room B In Situ Experiment), SAND85-0213, Sandia National Laboratories, Albuquerque, New Mexico.

Morgan, H. S., and C. M. Stone, 1985. Pretest Reference Calculation for the 6.1 Meter (20 ft) Wide Drifts for the Geomechanical Evaluation (WIPP Room G In Situ Experiment), SAND85-0893, Sandia National Laboratories, Albuquerque, New Mexico.

Morgan, H. S., C. M. Stone, and R. D. Krieg, 1985. The Use of Field Data to Evaluate and Improve Drift Response Models for the Waste Isolation Pilot Plant (WIPP), SAND84-2126C, Sandia National Laboratories, Albuquerque, New Mexico, presented at 26th U.S. Symposium on Rock Mechanics, Rapid City, South Dakota.

Morgan, H. S., C. M. Stone, and R. D. Krieg, 1986. A Comparison of Computed and Measured Responses for a Long Isolated Excavation at the Waste Isolation Pilot Plant (WIPP) Facility Computers and Geotechnics, SAND86-223858J, Sandia National Laboratories, Albuquerque, New Mexico.

Morgan, H. S., C. M. Stone, and R. D. Krieg, 1986. An Evaluation of WIPP Structural Modeling Capabilities Based on Comparisons with South Drift Data, SAND85-0323, Sandia National Laboratories, Albuquerque, New Mexico.

Morgan, H. S., C. M. Stone, R. D. Krieg, and D. E. Munson, 1986. Thermal/Structural Modeling of a Large Scale In Situ Overtest Experiment for Defense High-Level Waste at the WIPP Facility, SAND85-1926C, Sandia National Laboratories, Albuquerque, New Mexico, presented at the Second International Symposium on Numerical Models in Geomechanics, Ghent, Belgium.

Morgan, H. S., W. Wallner, and D. E. Munson, 1987. Results of an International Parallel Calculations Exercise Comparing Creep Responses Predicted with Three Computer Codes for Two Excavations in Rock Salt, SAND87-2125, Sandia National Laboratories, Albuquerque, New Mexico.

Morse, J. G., and B. W. Hassinger, 1985. "Brine Testing Program Plan: Waste Isolation Pilot Plant (WIPP) Project, Carlsbad, New Mexico, Revision 22," Internal Document WD:85:01214, transmitted as a letter from W. R. Cooper to R. H. Neill, WIPP:AEH:85:086.

Moss, M., 1980. Energy of Gamma-Irradiated 97% Pure Rock Salt, SAND80-1616C, Sandia National Laboratories, Albuquerque, New Mexico, presented at ASME Symposium on Thermophysical Properties of Composite Materials, Chicago, Illinois, Paper 80-WA/HT-53.

Moss, M., 1980. Stored Energy of Gamma Irradiated WIPP Salt, SAND80-0416, Sandia National Laboratories, Albuquerque, New Mexico.

Moss, M., and G. M. Haseman, 1981. Thermal Conductivity of Polyhalite and Anhydrite from the Site of the Proposed Waste Isolation Pilot Plant, SAND81-0856, Sandia National Laboratories, Albuquerque, New Mexico.

Moss, M., and M. A. Molecke, 1983. "Thermal Conductivity of Bentonite/Quartz High-Level-Waste-Package Backfill," SAND82-1274C, Sandia National Laboratories, Albuquerque, New Mexico, presented at the Materials Research Society Meeting, Boston, Massachusetts, published in Scientific Basis for Nuclear Waste Management, Vol. 15, ed. D. Brookins, North-Holland, New York.

MSHA (Mine Safety and Health Administration), Federal Mining Code for Metal and Nonmetallic Underground Mines, Title 30, Code of Federal Regulations, Part 57.

Munson, D. E., 1979. Preliminary Deformation-Mechanism Map for Salt (With Application to WIPP), SAND79-0076, Sandia National Laboratories, Albuquerque, New Mexico.

Munson, D. E., 1981. Establishing Parametric Bounds for Groundwater Transport of Nuclides from a Mine Geologic Repository, SAND81-2626J, Sandia National Laboratories, Albuquerque, New Mexico.

Munson, D. E., 1982. Application of Mechanism Maps to Response of Storage Caverns in Salt, SAND82-1218A, Sandia National Laboratories, Albuquerque, New Mexico, presented at 6th International Symposium on Salt, Toronto, Canada.

Munson, D. E., 1982. "Rock Mechanics Considerations for Design of a Nuclear Waste Repository in Salt," SAND82-0185J, Sandia National Laboratories, Albuquerque, New Mexico, published in NAS Tunneling Technology Journal.

Munson, D. E., 1982. "A Work Hardening/Recovery Model of Transient Creep of Salt During Stress Loading and Unloading, "SAND82-0810J, Sandia National Laboratories, Albuquerque, New Mexico, Proceedings 23rd U.S. Symposium on Rock Mechanics, University of California, Berkeley, California, pp. 299-306.

Munson, D. E., 1983. Geomechanical Evaluation, Sandia National Laboratories, Albuquerque, New Mexico.

Munson, D. E., 1983. Overtest for Simulated Defense High-Level Waste (DHLW), Sandia National Laboratories, Albuquerque, New Mexico.

Munson, D. E., 1983. 12-w/m2 Mockup for Defense High-Level Waste (DHLW), Sandia National Laboratories, Albuquerque, New Mexico.

Munson, D. E., and P. R. Dawson, 1979. Constitutive Model for the Low Temperature Creep of Salt (With Application to WIPP), SAND79-1853, Sandia National Laboratories, Albuquerque, New Mexico.

Munson, D. E., and P. R. Dawson, 1980. Numerical Simulation of Creep Closure of Deep

Potash Mine (With Application to Waste Isolation), SAND80-0467, Sandia National Laboratories, Albuquerque, New Mexico.

Munson, D. E. and P. R. Dawson, 1981. Salt-Constitutive Modeling Using Mechanism Maps, SAND81-2196C, Sandia National Laboratories, Albuquerque, New Mexico, presented at first Conference on the Mechanical Behavior of Salt, Pennsylvania State University, University Park, Pennsylvania.

Munson, D. E., and P. R. Dawson, 1982. A Transient Creep Model for Salt During Stress Loading and Unloading, SAND82-0962, Sandia National Laboratories, Albuquerque, New Mexico.

Munson, D. E., and P. R. Dawson, 1982. "A Work Hardening/Recovery Model of Transient Creep of Salt During Stress Loading and Unloading", SAND82-0370A, Sandia National Laboratories, Albuquerque, New Mexico, presented at the 23rd U.S. Symposium on Rock Mechanics, University of California, Berkeley, California.

Munson, D. E., and A. F. Fossom, 1986. "Comparison Between Predicted and Measured South Drift Closures at the WIPP Using a Transient Creep Model for Salt," Rock Mechanics: Key to Energy Production, Proceedings of the 27th U.S. Symposium on Rock Mechanics, University of Alabama, Tuscaloosa, Alabama.

Munson, D. E., and R. V. Matalucci, 1984. Planning, Developing, and Fielding of the WIPP Thermal/Structural Interactions In Situ Tests, SAND83-2048C, Sandia National Laboratories, Albuquerque, New Mexico, presented at Waste Management 1984, Tucson, Arizona.

Munson, D. E., and R. V. Matalucci, 1984. "Planning, Developing, and Fielding of Thermal/Structural Interactions In Situ Tests for the Waste Isolation Pilot Plant (WIPP)," SAND84-0379J, Sandia National Laboratories, Albuquerque, New Mexico, presented in Proceedings of the Symposium on Waste Management, Tucson, Arizona.

Munson, D. E., and R. V. Matalucci, 1986. Planning, Developing, and Fielding of Thermal/Structural Interactions In Situ Tests for the Waste Isolation Pilot Plant (WIPP), SAND83-2048, Sandia National Laboratories, Albuquerque, New Mexico.

Munson, D. E., R. V. Matalucci, and T. M. Torres, 1984. Implementation of Thermal/Structural Interactions In Situ Tests at the Waste Isolation Pilot Plant Facility, SAND 84-0804C, Sandia National Laboratories, Albuquerque, New Mexico, Second Conference on Mechanical Behavior of Salt, Hannover, Federal Republic of Germany.

Munson, D. E., R. V. Matalucci, and T. M. Torres, 1984. Thermal Structural Interactions In Situ Tests Implementation at the Waste Isolation Pilot Plant Facility, SAND84-0807A, Sandia National Laboratories, Albuquerque, New Mexico, Second Conference on Mechanical Behavior of Salt, Hannover, Federal Republic of Germany.

Munson, D. E., and H. S. Morgan, 1986. Methodology for Performing Parallel Design Calculations (Nuclear Waste Repository Application), SAND85-0324, Sandia National Laboratories, Albuquerque, New Mexico.

Munson, D. E., T. M. Torres, and D. Blankenship, 1986. Early Results from the

Thermal/Structural In Situ Test Series at the WIPP, SAND85-2237C, Sandia National Laboratories, Albuquerque, New Mexico, presented at the 27th U.S. Rock Mechanics Symposium.

Munson, D. E., T. M. Torres, and R. L. Jones, 1987. Pseudostrain Representation of Multipass Excavations in Salt, SAND86-22397C, Sandia National Laboratories, Albuquerque, New Mexico, presented at the U.S. Symposium on Rock Mechanics.

Munson, D. E., T. M. Torres, and R. L. Jones, 1987. Results of a Large, In Situ, Heated Axisymmetric Pillar Test at the Waste Isolation Pilot Plant (WIPP), SAND87-2257A, Sandia National Laboratories, Albuquerque, New Mexico, presented at the 29th U.S. Symposium on Rock Mechanics, Minneapolis, Minnesota.

NAS/NRC, 1970. "Report by the Committee on Radioactive Waste Management, Disposal of Solid Radioactive Waste in Bedded Salt Deposits."

Neill, R. H., J. K. Channell, L. Chaturvedi, M. S. Little, K. Rehfeldt, and P. Spiegler, 1983. Evaluation of the Suitability of the WIPP Site, EEG-23, New Mexico Environmental Evaluation Group, Santa Fe, New Mexico.

Nowak, E. J., 1978. "Migration of Eu Through Geologic Media," Sandia National Laboratories, Albuquerque, New Mexico, published in the American Geophysics Union, Vol. 59, p. 1224.

Nowak, E. J., 1979. The Backfill Barrier as a Component in a Multiple Barrier Nuclear Waste Isolation System, SAND79-1109, Sandia National Laboratories, Albuquerque, New Mexico.

Nowak, E. J., 1980. "The Backfill as an Engineered Barrier for Nuclear Waste Management," in Scientific Basis for Nuclear Waste Management, Vol. 2, ed. C. J. Northrup, Plenum Press, New York.

Nowak, E. J., 1980. "Diffusion of Colloids and Other Waste Species in Brine Saturated Backfill Materials," SAND83-1843C, Sandia National Laboratories, Albuquerque, New Mexico, published in Scientific Basis for Nuclear Waste Management, Vol. 2.

Nowak, E. J., 1980. Radionuclide Sorption and Migration Studies of Getters for Backfill Barriers, SAND79-1110, Sandia National Laboratories, Albuquerque, New Mexico.

Nowak, E. J., 1981. Backfill Barriers for Nuclear Waste Repositories in Salt, SAND81-0377A, Sandia National Laboratories, Albuquerque, New Mexico, presented at the International Seminar on Chemistry and Process Engineering for High-Level Liquid Waste Solidification, Julich, Federal Republic of Germany.

Nowak, E. J., 1981. "Composite Backfill Materials for Radioactive Waste Isolation by Deep Burial in Salt," ed. J. G. Moore, Scientific Basis for Nuclear Waste Management, Vol. 3, Plenum Press, New York.

Nowak, E. J., 1981. Predicting the Performance of a Backfill Barrier to Radionuclide Migration for a Repository in Salt, SAND80-2565A, Sandia National Laboratories, Albuquerque, New Mexico, presented at the NRC Symposium on Uncertainties Associated with the

Regulation of the Geologic Disposal of High-Level Radioactive Waste, Gatlinburg, Tennessee.

Nowak, E. J., 1982. The Diffusion of Cs(I) and Sr(II) in Liquid-Saturated Beds of Backfill Materials, SAND82-0750, Sandia National Laboratories, Albuquerque, New Mexico.

Nowak, E. J., 1982. The Diffusion of Leached Constituents in Backfill Barriers, SAND82-1156A, Sandia National Laboratories, Albuquerque, New Mexico, Second Semiannual Workshop on the Leaching Mechanisms of Nuclear Waste Forms.

Nowak, E. J., 1983. "Diffusion of Radionuclides in Brine-Saturated Backfill Barrier Materials," SAND82-1258C, Sandia National Laboratories, Albuquerque, New Mexico, presented at the Materials Research Society Meeting, Boston, Massachusetts, published in Scientific Basis for Nuclear Waste Management VI, Vol. 15, ed. D. Brookins, North-Holland, New York, pp. 735-744.

Nowak, E. J., 1984, "Diffusion of Colloids and Other Waste Species in Brine-Saturated Backfill Materials," Scientific Basis for Nuclear Waste Management VII, ed. G. L. McVay, Materials Research Society Symposium Proceedings, Vol. 26, pp. 59-68.

Nowak, E. J., 1985. "Diffusion of Plutonium in Compacted, Brine-Saturated Bentonite," Scientific Basis for Nuclear Waste Management VIII, ed. C. M. Jantzen, J. A. Stone, and R. C. Ewing, Materials Research Society Symposium Proceedings, Pittsburgh, Pennsylvania, Vol. 44, pp. 351-358.

Nowak, E. J., 1985. Test Plan Moisture Release Experiment for Rooms Al and B, Sandia National Laboratories, Albuquerque, New Mexico.

Nowak, E. J., 1986. "Brine Migration Studies in the Waste Isolation Pilot Plant (WIPP)," SAND85-1987c, Sandia National Laboratories, Albuquerque, New Mexico, in Waste Management 1986---Waste Isolation in the U.S. Technical Programs and Public Education, Proceedings of the Symposium on Waste Management, Tuscon, Arizona, Vol. 2, pp. 53-158.

Nowak, E. J., 1986. "Brine Movement in Waste Isolation Pilot Plant (WIPP) Tests," SAND86-1094A, Sandia National Laboratories, Albuquerque, New Mexico, presented at the Symposium on Origin & Evaluation of Brines in the Subsurface, American Chemical Society National Meeting, Anaheim, California.

Nowak, E. J., 1986. Preliminary Results of Brine Migration Studies in the Waste Isolation Pilot Plant (WIPP), SAND86-0720, Sandia National Laboratories, Albuquerque, New Mexico.

Nowak, E. J., 1988. "Assessment of Brine Inflow to WIPP Disposal Rooms," Memo from Experimental Programs Division, Sandia National Laboratories, Albuquerque, New Mexico.

Nowak, E. J., 1988. Test Plan: Brine Inflow and Humidity Experiments in Room D, Sandia National Laboratories, Albuquerque, New Mexico.

Nowak, E. J. The Diffusion of Cs(I) and Sr(II) in Liquid-Saturated Backfill Materials, SAND82-0750, Sandia National Laboratories, Albuquerque, New Mexico.

Nowak, E. J., and J. L. Krumhansl, 1982. Backfill Barriers for Radioactive Waste Isolation: Effectiveness and Research Needs, SAND82-2158J, Sandia National Laboratories, Albuquerque, New Mexico.

Nowak, E. J., and D. F. McTigue, 1987. Interim Results of Brine Transport Studies in the Waste Isolation Pilot Plant (WIPP), SAND87-0880, Sandia National Laboratories, Albuquerque, New Mexico.

Nowak, E. J., D. F. McTigue, and R. Beraun. "Brine Inflow to WIPP Disposal Rooms: Data Modeling and Assessment," SAND88-0112, (in preparation), Sandia National Laboratories, Albuquerque, New Mexico.

Nowak, E. J., and M. A. Molecke, 1979. WIPP Radionuclide Getter Development Study Thermal Transport Properties of Backfill Materials, Sandia National Laboratories, Albuquerque, New Mexico.

Nowak, E. J., and J. C. Stormont, 1987. Scoping Model Calculations of the Reconsolidation of Crushed Salt in the WIPP Shafts, SAND87-0879, Sandia National Laboratories, Albuquerque, New Mexico.

NRC (Nuclear Regulatory Commission), 1986. "Safety Goals for the Operations of Nuclear Power Plants; Correction and Republication of Policy Statement," Title 10, Part 50, 51 FR 30028.

Oak Ridge National Laboratory, 1973. Site Selection Factors for the Bedded Salt Pilot Plant, ORNL-TM-4219, Oak Ridge, Tennessee.

Oak Ridge National Laboratory, 1988. Integrated Data Base for 1988: Spent Fuel and Radioactive Waste Inventories, Projections, and Characteristics, DOE/RW-0006, Rev. 4, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Olander, D. R. et al., 1980. Thermal Gradient Migration of Brine Inclusions in Salt, ONWI-208, Office of Nuclear Waste Isolation, Columbus, Ohio.

ONWI (Office of Nuclear Waste Isolation), 1980. NWTS Criteria for the Geologic Disposal of Nuclear Wastes: Site-Qualification Criteria, draft, ONWI-33(2), Battelle Memorial Institute, Columbus, Ohio.

OSTP (Office of Science and Technology Policy), 1978. Isolation of Radioactive Wastes in Geologic Repositories: Status of Scientific and Technological Knowledge, draft, Executive Office of the President, Washington, D.C.

Pace, C. E., and C. W. Gulick, 1985. Expansive Grout Plug Effects in Restrained Environments, TR SL-85-?, prepared for Sandia National Laboratories, Albuquerque, New Mexico.

Pahwa, S. B., and J. R. Wayland, 1979. Far Field Thermal Calculations for the WIPP Site in SENM, SAND79-1013C, Sandia National Laboratories, Albuquerque, New Mexico, presented at Materials Research Society Meeting, Boston, Massachusetts.

Petersen, W. B., 1982. Estimating Concentrations Downwind from an Instantaneous PUFF Release, EPA 600/3-82-078, Office of Research and Development, Research Triangle Park, North Carolina.

Peterson, E., and S. Kelkar, 1983. Laboratory Tests to Determine Thermal and Hydraulic Properties of Bentonite-Based Backfill Systems, SAND82-7221, Terra Tek, Inc., Albuquerque, New Mexico.

Peterson, E. W., 1980. Analysis of Borehole Plug Performance, presented at 1980 NWTS Annual Information Meeting, Columbus, Ohio.

Peterson, E. W., 1987. Summary of N1100 Drift Tracer Gas Travel Time Tests, SSS-LR-88-9078, prepared for Sandia National Laboratories, Albuquerque, New Mexico.

Peterson, E. W., and C. L. Christensen, 1980. Analysis of Bell Canyon Test Results, SAND80-7044C, Sandia National Laboratories, Albuquerque, New Mexico, presented at the International BHP Symposium, Columbus, Ohio.

Peterson, E. W., and P. L. Lagus, 1985. "WIPP Horizon In Situ Permeability Measurements/Permeability Test System," S-CUBED Document No. 31062-08-01.

Peterson, E. W., P. L. Lagus, J. Brown, and K. Lie, 1985. WIPP Horizon In Situ Permeability Measurements, SAND85-7166, prepared for Sandia National Laboratories, Albuquerque, New Mexico.

Peterson, E. W., P. L. Lagus, and K. Lie. Fluid Flow Measurements of Test Series A and B for the Small Scale Seal Performance Tests, SAND87-7041, Sandia National Laboratories, Albuquerque, New Mexico.

Peterson, E. W., P. L. Lagus, and K. Lie. WIPP Horizon Free Field Fluid Transport Characteristics, SAND87-7164, Sandia National Laboratories, Albuquerque, New Mexico.

Peterson, E. W., P. L. Lagus, and J. C. Stormont, 1986. Permeability Measurements at the Waste Isolation Pilot Plant (WIPP), SAND85-1988A, Sandia National Laboratories, Albuquerque, New Mexico, presented at Waste Management '86 Conference.

Peterson, E. W., and J. C. Stormont, 1984. Test WIPP Horizon In Situ Permeability Measurements, Sandia National Laboratories, Albuquerque, New Mexico.

Peterson, R. D., A. J. Johnson, and S. D. Swanson, 1988. "Application of Microwave Energy for Solidification of TRU Waste," Proceedings of the Symposium on Waste Management at Tuscon, Arizona, February, 1988, University of Arizona, Vol. II.

Pettigrew, R., and Associates, 1977. WIPP Conceptual Design Report (Soils and Foundation Investigation), prepared for Sandia National Laboratories, Albuquerque, New Mexico.

Pfeifle, T. W., 1985. Mechanical Properties and Consolidation of Potential DHLW Backfill Materials: Crushed Salt and 70/30 Bentonite/Sand, SAND85-7208, RE/SPEC, Inc., for

Sandia National Laboratories, Albuquerque, New Mexico.

Pfeifle, T. W., 1987. Backfill Material Specifications and Requirements for the WIPP Simulated DHLW and TRU Waste Technology Experiments, SAND85-7209, RE/SPEC, Inc., for Sandia National Laboratories, Albuquerque, New Mexico.

Pfeifle, T. W., 1987. Mechanical Properties and Consolidation of Potential DHLW Backfill Materials: Crushed Salt and 70/30 Bentonite/Sand, SAND85-7208, RE/SPEC, Inc., for Sandia National Laboratories, Albuquerque, New Mexico.

Pfeifle, T. W., and P. E. Senseny, 1981. Elastic-Plastic Deformation of Anhydrite and Polyhalite as Determined from Quasi-Static Triaxial Compressions Tests, SAND81-7063, RE/SPEC, Inc., for Sandia National Laboratories, Albuquerque, New Mexico.

Pfeifle, T. W., and P. E. Senseny, 1985. Permeability and Consolidation of Crushed Salt from the WIPP Site, RE/SPEC Topical Report RSI-0278.

Pietz, J., 1983. "An Integral Equation Approach to Modeling Radionuclide Migration in the Backfill Barrier," Master's Thesis, University of New Mexico, Albuquerque, New Mexico.

Powers, D. W., 1979. WIPP Statement of Work for Deepening WIPP 13, Sandia National Laboratories, Albuquerque, New Mexico.

Powers, D. W., 1980. Markov Processes and Cycles in the Evaporites of the Upper Salado Formation, Southeastern New Mexico, SAND80-2530A, Sandia National Laboratories, Albuquerque, New Mexico, presented at University of Utah Seminar, University of Utah.

Powers, D. W., 1981. "Geologic Investigation of the WIPP Site: Overview and Issues," SAND81-0201C, in Environmental Geology and Hydrology in New Mexico, New Mexico Geological Society Special Publication 10, pp. 199-123.

Powers, D. W., et al., 1978. Geological Characterization Report, Waste Isolation Pilot Plant (WIPP) Site, Southeastern New Mexico, SAND78-1596, Sandia National Laboratories, Albuquerque, New Mexico.

Powers, D. W., and W. D. Weart, 1979. "Status of Siting for the Waste Isolation Pilot Plant," Trans ANS, Vol. 33, pp. 141-142. Also published as SAND79-1329A, Sandia National Laboratories, Albuquerque, New Mexico.

Preece, D. S., 1987. Pretest 3-D Finite Modeling of the Wedge Pillar Portion of the WIPP Geomechanical Evaluation (Room G) In Situ Experiment, SAND87-0472, Sandia National Laboratories, Albuquerque, New Mexico.

Price, R. H., 1980. Effects of Anhydrite and Pressure on the Mechanical Behavior of Synthetic Rock Salt, SAND80-2005J, Sandia National Laboratories, Albuquerque, New Mexico, presented at the Fall 1980 AGU Meeting, San Francisco, California.

Rack, H. J., L. Abrego, and M. A. Molecke, 1981. Titanium Utilization in Long-Term Nuclear Waste Storage, SAND80-2129A, Sandia National Laboratories, Albuquerque, New Mexico, presented at the Annual AIME Meeting, Chicago, Illinois.

Ratigan, J. L., 1987. A Sensitivity Study of Brine Transport into a Borehole Containing a Commercial High-Level Waste Canister, ONWI-384, prepared for the Office of Nuclear Waste Isolation, Battelle Memorial Institute, Columbus, Ohio.

Ratigan, J. L., and R. A. Wagner, 1978. "Thermomechanical Analysis of Crushed-Salt Backfilled Disposal Rooms in a Conceptual Radioactive Waste Repository in Dome Salt," in Conceptual Design Report, National Waste Terminal Storage Repository for Storing Reprocessing Wastes in a Dome Salt Formation, Special Study No. 2, Vol. XVIII, Stearns-Roger Services, Denver, Colorado.

Rhoderick, J. E., and A. D. Buck, 1981. Borehole Plugging Program (Waste Disposal), Report 2, Petrographic Examination of Several Four-Year-Old Laboratory-Developed Grout Mixtures, USAE Waterways Experiment Station Miscellaneous Paper C-78-1 (also issued as ONWI-342).

Rhoderick, J. E., and A. D. Buck, 1981. Examination of Simulated Borehole Specimens, SL-81-7, SAND81-7109, ONWI-247, Structures Laboratory, U.S. Army Engineers, Waterways Experiment Station, Vicksburg, Mississippi.

Rhoderick, J. E., and A. D. Buck, 1981. Petrographic Examination of Several Four-Year-Old Laboratory-Developed Grant Mixtures, C-78-1 Waterways Experiment Station, ONWI-342, Sandia National Laboratories, Albuquerque, New Mexico.

Rhoderick, J. E., G. W. Wong, and A. D. Buck, 1981. Examination of Samples of Bell Canyon Test 1-FF Grout, SAND81-7110, Sandia National Laboratories, Albuquerque, New Mexico, SL-81-5, ONWI-246, Structures Laboratory, U.S. Army Engineers, Waterways Experiment Station, Vicksburg, Mississippi.

Rigdon, S. D., and P. S. Hughes, 1979. Waste Isolation Pilot Plant (WIPP) Operator Dose Calculations Final Report, SAND79-7028, Sandia National Laboratories, Albuquerque, New Mexico.

Roedder, E., and H. E. Belkin, 1979. "Application of Studies of Fluid Inclusions in Permian Salado Salt, New Mexico, to Probems of Siting the Waste Isolation Pilot Plant," ed. G. J. McCarthy, Scientific Basics for Nuclear Waste Management, Vol. 1, Plenum Press, New York, pp. 313-321.

Roy, D. M., M. W. Grutzeck, K. Mather, and A. D. Buck, 1981. PSU/WES Interlaboratory Comparative Methodology Study of an Experimental Cementitious Repository Seal Material, Report 2, Final Results, USAE Waterways Experiment Station Miscellaneous Paper SL-81-2 (also published as ONWI-324).

Rupp, A. F., 1948. Dilution of Stack Gases in Cross Winds, USAEC Report AECD-1811 (CE-1620), Clinton Laboratories.

Ruppen, J. A., M. A. Molecke, and R. S. Glass, 1981. "Titanium Utilization in Long-Term Nuclear Waste Storage," SAND81-2466J, Sandia National Laboratories, Albuquerque, New Mexico, published in Titanium for Energy and Industrial Applications, ed. D. Eylon, TMS-AIME Publications, pp. 355-69.

Ruppen, J. A., R. B. Diegle, R.S. Glass, and T. J. Headley, 1983. "Some Effects of Microstructure and Chemistry on Corrosion and Hydrogen Embrittlement of TiCode-12," SAND82-1279C, Sandia National Laboratories, Albuquerque, New Mexico, presented at Materials Research Society, Boston, Massachusetts, published in Scientific Basis for Nuclear Waste Management VI, Vol. 15, ed. D. Brookins, North-Holland, New York.

Sandia National Laboratories, 1977. A Study of the Nuclear Regulatory Commission Quality Assurance Program, SAND77-1009, Sandia National Laboratories, Albuquerque, New Mexico.

Sandia National Laboratories, 1977. WIPP Conceptual Design Report, SAND77-0274, Albuquerque, New Mexico.

Sandia National Laboratories, 1978. Review Comments on Geological Characterization Report, Waste Isolation Pilot Plant (WIPP) Site, Southeastern New Mexico, SAND78-1596, Volumes I and II, Albuquerque, New Mexico.

Sandia National Laboratories, 1979. Summary of Research and Development Activities in Support of Waste Acceptance Criteria for WIPP, SAND79-1305, Albuquerque, New Mexico.

Sandia National Laboratories, 1979. Waste Isolation Pilot Plant (WIPP) Program Plan for FY 1979, SAND79-0098, Albuquerque, New Mexico.

Sandia National Laboratories, 1979. Waste Isolation Pilot Plant (WIPP) Program for FY 1980, SAND79-1771, Albuquerque, New Mexico.

Sandia National Laboratories, 1980. Waste Isolation Pilot Plant (WIPP) Program Plan for FY 1981, SAND79-1771, Albuquerque, New Mexico.

Sandia National Laboratories, 1982. Waste Isolation Pilot Plant (WIPP) Program Plan for FY 1982, SAND82-0597, Albuquerque, New Mexico.

Sandia National Laboratories, 1983. Waste Isolation Pilot Plant (WIPP) Program Plan for FY 1983, SAND83-0127, Albuquerque, New Mexico.

Sandia National Laboratories, 1983. Waste Isolation Pilot Plant (WIPP) Program Plan for FY 1984, Sandia National Laboratories, Albuquerque, New Mexico.

Sandia National Laboratories, 1984. Waste Isolation Pilot Plant (WIPP) Program Plan for FY 1985, Albuquerque, New Mexico.

Sandia National Laboratories, 1985. Waste Isolation Pilot Plant (WIPP) Program Plan for FY 1986, Albuquerque, New Mexico.

Sandia National Laboratories, 1987. Waste Isolation Pilot Plant (WIPP) Program Plan for FY 1987, Albuquerque, New Mexico.

Sandia National Laboratories, 1987. Waste Isolation Pilot Plant (WIPP) Program Plan for FY 1988, Albuquerque, New Mexico.

Sandia National Laboratories, 1987. WIPP Conceptual Design Report, SAND77-0274: Part I, II, III, Addendum C-F, Addendum A thru J, L, and M (Addendum K was never published), Albuquerque, New Mexico.

Sarkar, A. K., M. W. Barnes, and D. M. Roy, 1982. Longevity of Borehole and Shaft Sealing Materials: I. Thermodynamic Properties of Cement and Related Phases Applied to Repository Sealing, ONWI-201, Office of Nuclear Waste Isolation, Columbus, Ohio.

Sattler, A. R., and C. L. Christensen, 1980. Measurements of Very Large Deformations in "Potash Salt" in Conjunction with an Ongoing Mining Operation, SAND79-2254, Sandia National Laboratories, Albuquerque, New Mexico.

Sattler, A. R., and T. O. Hunter, 1987. Pre-WIPP In Situ Experiments in Salt Part I, Executive Summary; Part II, Program Description, SAND79-0625, Sandia National Laboratories, Albuquerque, New Mexico.

Sattler, A. R., H. C. Walker, and T. O. Hunter, 1979. In Situ Experiments to Support Development of the Waste Isolation Pilot Plant (WIPP), SAND79-0905C, Sandia National Laboratories, Albuquerque, New Mexico, presented at Materials Research Society, Boston, Massachusetts.

Schutz, P. W., and J. A. Hall, Titanium Metals Corp of America, 1987. Optimization of Mechanical/Corrosion Properties of Ti-Code 12 Plate and Sheet---Part II: Thermomechanical Processing Effects, State II Final Report, SAND87-7171, Sandia National Laboratories, Albuquerque, New Mexico.

Schutz, R. W., and J. A. Hall, 1984. Optimization of Mechanical/Corrosion Properties of TI-code-12 Plate and Sheet, Part I: Compositional Effects, Stage I Final Report, SAND83-7438, TIMET Titanium Metals Corporation of America, Henderson Technical Laboratory, Henderson, Nevada.

Senseny, P. E., 1986. Triaxial Compression Creep Tests on Salt from the Waste Isolation Pilot Plant, SAND85-7261, Sandia National Laboratories, Albuquerque, New Mexico.

Serata, S., and T. E. Cundey, 1978. Rheological Method of Salt County Design for Underground Storage of Solid Liquid and Gaseous Matters, 19th U.S. Symposium on Rock Mechanics, ed. Y. S. Kim, Reno, Nevada.

Serata, S., S. Sakuri, and T. Adachi, 1972. "Theory of Aggregate Rock Behavior Based on Absolute Three-Dimensional Testing (ATT) of Rock Salt," Basic and Applied Rock Mechanics, Proceedings 10th U.S. Symposium Rock Mechanics, ed. K. E. Gray, AIME.

Serne, R. J., et al., 1977. Batch Kd Measurements of Nuclides to Estimate Migration Potential at the Proposed Waste Isolation Pilot Plant in New Mexico, PNL-2448, Battelle Pacific Northwest Laboratories.

Seward, P. O., 1982. Abridged Borehole Histories for the Waste Isolation Pilot Plant (WIPP) Studies, SAND82-0080, Sandia National Laboratories Albuquerque, New Mexico.

Shade, J. W., D. J. Bradley, and M. A. Molecke, 1980. "Joint PNL0WIPP High-Level Total

System Test," PNL-SA-8716, published in Proceedings of the 1980 National Waste Terminal Storage Program Information Meeting, ONWI-212, p. 234.

Shaffer, S. E., 1982. Bibliography of Geology of Southeastern New Mexico, SAND82-0-770, Sandia National Laboratories, Albuquerque, New Mexico.

Shefelbine, H. C., 1979. Test of Thermal Conductivity Probe in the Avery Island Salt Mine, Sandia National Laboratories, Albuquerque, New Mexico.

Shefelbine, H. C., 1979. WIPP Statement of Work for Deepening WIPP 13, Sandia National Laboratories, Albuquerque, New Mexico.

Shefelbine, H. C., 1982. Brine Migration: A Summary Report, SAND82-0152, Sandia National Laboratories, Albuquerque, New Mexico.

Shor, A. J., C. F. Baes, and C. M. Canonico, 1981. Consolidation and Permeability of Salt in Brine, ORNL-5774, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Sisson, C. E., 1978. Predicted Temperatures in a Bedded-Salt Repository Resulting from Burial of DOE High-Level Nuclear Waste Canisters, SAND78-0924, Sandia National Laboratories, Albuquerque, New Mexico.

Sjaardema, G. D., and R. D. Krieg, 1987. A Constitutive Model for the Consolidation of WIPP Crushed Salt and Its Use in Analyses of Backfilled Shaft Drift Configurations, SAND87-1977, Sandia National Laboratories, Albuquerque, New Mexico.

Sorensen, N. R., J. A. Ruppen, and R. W. Schutz, 1984. The Environmental Cracking of TiCode 12 in a Repository Brine, SAND83-2207J, NACE Corrosion 1984 International Forum and Exhibition, New Orleans, Louisiana.

Spiegler, P., 1981. An Approach to Calculating Upper Bounds on Maximum Individual Doses from the Use of Contaminated Well Water Following a WIPP Repository Breach.

St. John, C. M., 1978. "Computer Models and the Design of Underground Radioactive Waste Repositories," M.I.T. Special Summer Session on Rock Engineering.

Statler, R. D., 1978. Field Operations Program for WIPP H5A, B, C, H6A, B and C, Sandia National Laboratories, Albuquerque, New Mexico.

Statler, R. D., 1978. Hydro Test Wells H-4, H-5 and H-6, Vol. II, Sandia National Laboratories, Albuquerque, New Mexico.

Statler, R. D., 1979. Experimental Coreholes into Breccia Pipe from Mississippi Chemical Potash Mine, Sandia National Laboratories, Albuquerque, New Mexico.

Statler, R. D., 1979. Field Engineering Summary for Hydro Test Wells H-4, H-5, and H-6, Vols. 1 and 2, Sandia National Laboratories, Albuquerque, New Mexico.

Statler, R. D., 1979. Field Engineering Summary for Nash Draw Holes WIPP 25, 26, 27, 28, 29, and 30, Vols. 1 and 2, Sandia National Laboratories, Albuquerque, New Mexico.

Statler, R. D., 1979. Field Operations Plan for Hydrologic Study Wells WIPP H-7A, B&C, H-9A, B7c and H-10A, B&C, Sandia National Laboratories, Albuquerque, New Mexico.

Statler, R. D., 1979. Field Operations Program of Sandia Laboratories WIPP Site Investigations: Resistivity Anomaly WIPP 13, Sandia National Laboratories, Albuquerque, New Mexico.

Statler, R. D., 1979. Site Investigations for WIPP 32, Sandia National Laboratories, Albuquerque, New Mexico.

Statler, R. D., 1979. Site Investigations for WIPP 33, Sandia National Laboratories, Albuquerque, New Mexico.

Statler, R. D., 1979. Well Completion of H-4, 5 & 6 for WIPP Tracer Program Revision No. 1, Sandia National Laboratories, Albuquerque, New Mexico.

Statler, R. D., 1980. Field Operation Plan for Hydrologic Testing Nash Draw Wells WIPP-25 Through 30, Sandia National Laboratories, Albuquerque, New Mexico.

Statler, R. D., 1980. Field Operations Program of Sandia National Laboratories WIPP Site Investigations Resistivity Anomaly, WIPP 31, Sandia National Laboratories, Albuquerque, New Mexico.

Statler, R. D., 1980. Field Operations Program of Sandia National Laboratories WIPP Site Investigations WIPP 25-30, Sandia National Laboratories, Albuquerque, New Mexico.

Statler, R. D., 1980. Site Investigations for WIPP 31 Re-Entry and Deepening, Sandia National Laboratories, Albuquerque, New Mexico.

Stein, C. L., 1985. Mineralogy in the Waste Isolation Pilot Plant (WIPP) Facility Stratigraphic Horizon, SAND85-0321, Sandia National Laboratories, Albuquerque, New Mexico.

Stickney, R. G., and A. R. Sattler, 1979. In Situ Performance Evaluation of Stress Gages at Avery Island, Sandia National Laboratories, Albuquerque, New Mexico.

Stinebaugh, R. E., 1979. Compressibility of Granulated Rock Salt, SAND79-1119, Sandia National Laboratories, Albuquerque, New Mexico.

Stinebaugh, R. E., 1979. Retrieval of Canistered Experimental Waste at the Waste Isolation Pilot Plant, SAND79-1239, Sandia National Laboratories, Albuquerque, New Mexico.

Stone, C. M., R. D. Krieg, and Z. E. Beisinger, 1985. SANCHO--A Finite Element Computer Program for the Quasistatic, Large Deformation, Inelastic Response of Two-Dimensional Solids, SAND84-2618, Sandia National Laboratories, Albuquerque, New Mexico.

Stone, L. M., R. D. Krieg, and L. J. Branstetter, 1981. The Effect of Clay-Seam Behavior on WIPP Repository Design, SAND81-0768, Sandia National Laboratories, Albuquerque, New Mexico.

Stormont, J. C., 1983. Field Evaluation of Stress Measurement Techniques at the WIPP Site,

Sandia National Laboratories, Albuquerque, New Mexico.

Stormont, J. C., 1984. Plugging and Sealing Program for the Waste Isolation Pilot Plant (WIPP), SAND84-1057, Sandia National Laboratories, Albuquerque, New Mexico.

Stormont, J. C., 1984. "Recommendation to Reduce Width of Certain Drifts in the WIPP to Facilitate Repository Sealing," Memorandum, Sandia National Laboratories, Albuquerque, New Mexico.

Stormont, J. C., 1985. Operations Plan: Test Series B of the Small Scale Seal Performance Tests, Sandia National Laboratories, Albuquerque, New Mexico.

Stormont, J. C., 1985. Test Plan: Appendix B, Small Scale Seal Performance Tests, Sandia National Laboratories, Albuquerque, New Mexico.

Stormont, J. C., 1986. Development and Implementation: Test Series A of the Small Scale Seal Performance Tests, SAND85-2602, Sandia National Laboratories, Albuquerque, New Mexico.

Stormont, J. C., 1986. Development and Implementation: Test Series B of the Small Scale Seal Performance Tests, SAND86-1329, Sandia National Laboratories, Albuquerque, New Mexico.

Stormont, J. C., 1986. In Situ Seal Tests at the Waste Isolation Pilot Plant (WIPP), SAND85-1989C, Sandia National Laboratories, Albuquerque, New Mexico, presented at the Waste Management 1986, Tucson, Arizona.

Stormont, J. C., 1986. Operations Plan: Test Series C of the Small Scale Seal Performance Tests, Sandia National Laboratories, Albuquerque, New Mexico.

Stormont, J. C., 1986. Permeability Measurements at the Waste Isolation Pilot Plant (WIPP), SAND85-1988C, Sandia National Laboratories, Albuquerque, New Mexico, presented at the Waste Management 1986, Tucson, Arizona.

Stormont, J. C., 1986. Test Plan: Appendix C, Small Scale Seal Performance Tests, Sandia National Laboratories, Albuquerque, New Mexico.

Stormont, J. C., 1987. Small Scale Seal Performance Test Series A Thermal/Structural Data Through the 180th Day, SAND87-0178, Sandia National Laboratories, Albuquerque, New Mexico.

Stormont, J. C., 1988. Preliminary Seal Design Evaluation for the Waste Isolation Pilot Plant, SAND87-3083, Sandia National Laboratories, Albuquerque, New Mexico.

Stormont, J. C. Test Plan for the AEC-7 Re-Entry Test, Sandia National Laboratories, Albuquerque, New Mexico.

Stormont, J. C. Test Plan: Small Scale Seal Performance Tests, Sandia National Laboratories, Albuquerque, New Mexico.

Stormont, J. C., and J. G. Arguello. Model Calculations of Flow Through Shaft Seals in the

Rustler Formation, SAND87-2859 (in preparation), Sandia National Laboratories, Albuquerque, New Mexico.

Stormont, J. C., and C. L. Howard, 1986. Development, Implementation and Early Results: Test Series B of the Small Scale Seal Performance Tests, SAND86-1329, Sandia National Laboratories, Albuquerque, New Mexico.

Stormont, J. C., and C. L. Howard, 1986. Development, Implementation and Early Results: Test Series C of the Small Scale Seal Performance Tests, SAND87-2203, Sandia National Laboratories, Albuquerque, New Mexico.

Stormont, J. C., and C. L. Howard, 1987. Development, Implementation and Early Results: Test Series C of the Small Scale Seal Performance Tests, SAND87-2203, Sandia National Laboratories, Albuquerque, New Mexico.

Stormont, J. C., R. V. Matalucci, and H. S. Morgan, 1984. Field Evaluation of Stress Measurement Techniques in Rock Salt, SAND83-2051A, Sandia National Laboratories, Albuquerque, New Mexico, presented at 30th International Instrumentation Symposium, Denver, Colorado (report entered under SAND83-2507).

Stormont, J. C., R. V. Matalucci, and H. S. Morgan, 1984. Field Tests of Stress Measurement Techniques in Rock Salt, SAND83-2507, Sandia National Laboratories, Albuquerque, New Mexico.

Stormont, J. C., and E. W. Peterson, 1984. Test Plan: WIPP Horizon In Situ Permeability Measurement, Sandia National Laboratories, Albuquerque, New Mexico.

Stormont, J. C., E. W. Peterson, and P. L. Lagus, 1987. Summary of and Observations About WIPP Facility Horizon Flow Measurements Through 1986, SAND87-0176, Sandia National Laboratories, Albuquerque, New Mexico.

Stroup, D. E., and P. E. Senseny, 1987. Influence of Bentonite Content on Consolidation and Permeability of Crushed Salt from the WIPP, Topical Report RSI-0309, RE/SPEC, Inc., Albuquerque, New Mexico.

Sullivan, B. R., 1983. Results of the SPDV Site Validation Experiment Waste Isolation Pilot Plant, TME-3177, ed. S. Black, R. Nausten, and D. Shukla, prepared for U.S. Department of Energy.

Sumlin, H. A., 1981. A Report on the Development of Equipment and Techniques for Drilling and Machining Bedded Salt, SAND80-2603, Sandia National Laboratories, Albuquerque, New Mexico.

Sutherland, H. J., 1978. "Permeability of SENM Rock Salt," Interoffice Memo, Sandia National Laboratories, Albuquerque, New Mexico.

Sutherland, H. J., 1982. Subsidence Prediction of High Extraction Mining Using Complementary Influence Functions, SAND82-2949, Sandia National Laboratories, Albuquerque, New Mexico.

Sutherland, H. J., and S. P. Cave, 1979. Gas Permeability of SENM Rock Salt, SAND78-2287,

Sandia National Laboratories, Albuquerque, New Mexico.

Sutherland, H. J., and S. P. Cave, 1980. "Argon Gas Permeability of New Mexico Rock Salt Under Hydrostatic Compression," International Journal of Rock Mechanic Mineral Science & Geomechanical Abstracts, 17, pp. 281-288.

Sutherland, S. H., and D. E. Bennet, 1979. Defense High-Level Waste and Spent Fuel Characterization for Geologic Waste Repositories, SAND79-0172, Sandia National Laboratories, Albuquerque, New Mexico.

Sweet, J. N., and J. E. McCreight, 1979. Thermal Conductivity of Rock Salt and Other Geologic Materials from the Site of the Proposed Waste Isolation Pilot Plant, SAND79-1665C, Sandia National Laboratories, Albuquerque, New Mexico, presented at Thermal Conductivity Conference, Chicago, Illinois.

Sweet, J. N., and J. E. McCreight, 1980. Thermal Conductivity of Rock Salt and Other Geologic Materials from the Site of the Proposed Waste Isolation Pilot Plant, SAND79-1665, Sandia National Laboratories, Albuquerque, New Mexico.

Sweet, J. N., and J. E. McCreight, 1980. Thermal Properties Measurement on Rock Salt Samples from the Site of the Proposed Waste Isolation Pilot Plant, SAND80-0709, Sandia National Laboratories, Albuquerque, New Mexico.

Sweet, J. N., and J. E. McCreight, 1980. Thermal Properties Measurements on Rock Salt Samples from the Site of the Proposed Waste Isolation Pilot Plant, SAND80-0799, Sandia National Laboratories, Albuquerque, New Mexico.

Szeki, A., 1978. Influence of Volume on Creep Behavior of Rock Salt Pillars, Fifth International Salt Symposium, Hamburg, Federal Republic of Germany.

Terra Tek Inc., 1978. "Gas Permeabilities of SENM Salt," consulting memo to Sandia National Laboratories, Albuquerque, New Mexico.

Teufel, L. W., 1981. Mechanical Properties of Anhydrite and Polyhalite in Quasi-Static Triaxial Compression, SAND81-0858, Sandia National Laboratories, Albuquerque, New Mexico.

Thomas, R. L., C. V. Char, and W. J. Bergeron, 1973. "Finite Element Analysis of Creep in Rock Salt Pillar Models," Proceedings, 14th Symposium Rock Mechanics, eds. H. R. Hardy and R. Stefanko, AIME.

Thorne, B. J., 1980. Drift Response to Elevated Temperatures, SAND80-7086, Sandia National Laboratories, Albuquerque, New Mexico.

Thorne, B. J., 1981. Calculations of the Uplift of an Elastic Model of the Rustler Formation over a 75 K Watt/Acre Repository at the WIPP Site, SAND80-7166, Sandia National Laboratories, Albuquerque, New Mexico.

Thorne, B. J., and D. K. Rudeen, 1981. Regional Effects of TRU Repository Heat, SAND80-7161, Sandia National Laboratories, Albuquerque, New Mexico.

\Tien, P. L., F. B. Nimick, A. B. Muller, P. A. Davis, R. V. Guzowski, L. E. Dida, and R. L. Hunter, 1983. Repository Site Data and Information in Bedded Salt: Palo Duro Basin, Texas, NUREG/CR-3129, SAND82-2223, Sandia National Laboratories, Albuquerque, New Mexico.

Tillerson, J. R., and P. R. Dawson, 1981. "Heated Mine Room and Pillar Secondary Creep Response," SAND79-1199, Sandia National Laboratories, Albuquerque, New Mexico, published in International Journal for Numerical and Analytical Methods in Geomechanics, Vol. 5, pp. 177-193.

Tillerson, J. R., and M. M. Madsen, 1978. Thermoelastic Capabilities for the SAND-BMINES Program, SAND77-0378, Sandia National Laboratories, Albuquerque, New Mexico.

Torres, T. M., 1983. Heated Axisymmetric Pillar, Sandia National Laboratories, Albuquerque, New Mexico.

Torres, T. M., 1985. Scope of Work for Gas Testing, Sandia National Laboratories, Albuquerque, New Mexico.

Torres, T. M., 1986. Preliminary Evaluation of Drift Closure and Closure Rate Response with Respect to the Effects of Gas Occurrences Within the Waste Isolation Pilot Plant, SAND86-1059, Sandia National Laboratories, Albuquerque, New Mexico.

Torres, T. M., 1986. Two-Dimensions Axisymmetric and Plane Strain Scoping Calculations for the Small Scale Seal Performance Test, Series A, SAND86-0488, Sandia National Laboratories, Albuquerque, New Mexico.

Torres, T. M., 1987. Design Evaluation: Structural Calculations for the Construction and Salt Handling Shaft and the Waste Handling Shaft at the Waste Isolation Pilot Plant (WIPP), SAND87-2230, Sandia National Laboratories, Albuquerque, New Mexico.

Trucano, T. G., 1981. Computer Modeling of Stress States Associated with Fluid Migration Experiments, SAND81-2008, Sandia National Laboratories, Albuquerque, New Mexico.

Tyler, L. D., 1984. In Situ Tests and Instrumentation in the WIPP Facility, SAND84-0986C, Sandia National Laboratories, Albuquerque, New Mexico, presented at CEC/NEA Workshop, Brussels, Belgium.

Tyler, L. R., R. V. Matalucci, M. A. Molecke, D. E. Munson, E. J. Nowak, and J. C. Stormont, 1988. Summary Report for the WIPP Technology Development Program for the Isolation of Radioactive Waste, SAND88-0844, Sandia National Laboratories, Albuquerque, New Mexico.

Urai, J. L., C. J. Spiers, J. Zwart, and G. S. Lister, 1986. "Weakening of Rock Salt by Water During Long-Term Creep," Nature, Vol. 324, pp. 554-557.

Van Sambeek, L. L., 1987. Thermal and Thermomechanical Analyses of WIPP Shaft Seals, SAND87-7039, Sandia National Laboratories, Albuquerque, New Mexico.

Van Sambeek, L. L., and J. C. Stormont, 1987. Thermal/Structural Modeling of Test Series A of the Small Scale Seal Performance Tests, SAND87-7037, Sandia National

Laboratories, Albuquerque, New Mexico.

Wagner, R. A., and J. L. Ratigan, 1979. Thermo/Viscoelastic Analysis of a Heated Room with Crushed Salt Backfill in the Pre-WIPP Dedicated Area, RSI-0081.

Wakeley, L. D., and D. M. Roy, 1985. Cementatious Mixtures for Sealing Evaporite and Clastic Rocks in a Radioactive-Waste Repository, MP SL-85-16, prepared for Sandia National Laboratories, Albuquerque, New Mexico.

Wakeley, L. D., M. D. Walley, and A. D. Buck, 1986. Development of Fresh-Grout Subsequent to the Bell Canyon Tests (BCT), SL-86-2, Sandia National Laboratories, Albuquerque, New Mexico.

Wawersik, W. R., 1978. Effects of Pressure, Deviator Stress and Temperature on Transient Creep of Rock Salt at Low Confining Pressure AGU, Vol. 59, No. 4, p. 376.

Wawersik, W. R., 1979. Creep Fracture in Compression, in Chapter 3 in Summary of Research and Development Activities in Support of Waste Acceptance Criteria for WIPP, SAND79-1305, Sandia National Laboratories, Albuquerque, New Mexico.

Wawersik, W. R., 1979. Indirect Deformation (Strain) Measurements and Calibrations in Sandia Triaxial Apparatus for Rock Testing to 250°C, SAND79-0114, Sandia National Laboratories, Albuquerque, New Mexico.

Wawersik, W. R., 1981. Recent Measurements of Power Law Stress Exponents, Activation Energies and Creep Transients, SAND81-1117C, Sandia National Laboratories, Albuquerque, New Mexico, presented at the First Conference on the Mechanical Behavior of Salt, The Pennsylvania State University, University Park, Pennsylvania.

Wawersik, W. R., 1982. Low Temperature Creep Mechanisms in Rock Salt, SAND82-2071A, Sandia National Laboratories, Albuquerque, New Mexico, presented at AGU Fall Meeting, San Francisco, California.

Wawersik, W. R., 1983. Determination of Steady State Creep Rates and Activation Parameters for Rock Salt, SAND82-2731J, Sandia National Laboratories, Albuquerque, New Mexico.

Wawersik, W. R., 1983. Scope of Work for Gas Testing, Sandia National Laboratories, Albuquerque, New Mexico.

Wawersik, W. R., 1984. Comparison of Creep and Creep Models for Rock Salt, SAND84-0888A, Sandia National Laboratories, Albuquerque, New Mexico.

Wawersik, W. R., 1984. Hydraulic Fracturing Tests in Salt, SAND84-0890A, Sandia National Laboratories, Albuquerque, New Mexico, presented at Second Conference on the Mechanical Behavior of Salt, Hannover, Federal Republic of Germany.

Wawersik, W. R., 1984. On the Evolution of Texture and Microstructure in Laboratory Deformed Salt, SAND84-0889A, Sandia National Laboratories, Albuquerque, New Mexico, presented at Second Conference on the Mechanical Behavior of Salt, Hannover, Federal Republic of Germany.

Wawersik, W. R., 1986. Microstructures in Deformed Rock Salt from Around Old Mine Workings, SAND86-2187A, Sandia National Laboratories, Albuquerque, New Mexico, presented at the Fall AGU Meeting, San Francisco, California.

Wawersik, W. R., J. F. Callender, R. Weaver, and R. K. Dropek, 1976. Preliminary Determination of Mechanical Properties on Rock Salt from Southeastern New Mexico, Pre-Proc., 17th U.S. Symposium Rock Mechanics, University of Utah.

Wawersik, W. R., and D. W. Hannum, 1978. Mechanical Behavior of New Mexico Rock Salt in Triaxial Compression Up to 200°C, 5th International Salt Symposium, Hamburg, Federal Republic of Germany.

Wawersik, W. R., and D. W. Hannum, 1979. Interim Summary of Sandia Creep Experiments on Rock Salt from the WIPP Study Area, Southeastern New Mexico, SAND79-0115, Sandia National Laboratories, Albuquerque, New Mexico.

Wawersik, W. R. and D. W. Hannum, 1980. "Mechanical Behavior of New Mexico Rock Salt in Triaxial Compression Up to 200°C," Journal of Geophysical Research, Vol. 85, No. B2, pp. 891-900. Also published as SAND77-1192, Sandia National Laboratories, Albuquerque, New Mexico.

Wawersik, W. R., W. Herrmann, S. T. Montgomery, and H. S. Lauson, 1981. Evacuation Design in Rock Salt: Laboratory Experiments, Material Modeling and Validations, SAND81-1167C, Sandia National Laboratories, Albuquerque, New Mexico, presented at International Symposium on Rock Mechanics Related to Caverns and Pressure Shafts, Aachen, Federal Republic of Germany.

Wawersik, W. R., and H. S. Morgan, 1987. "Evaluating Design Procedures for Structures in Rock Salt," SAND86-2597C, Sandia National Laboratories, Albuquerque, New Mexico, published in the Proceedings of the 28th U.S. Symposium on Rock Mechanics, presented at the University of Arizona, Tucson, Arizona.

Wawersik, W. R., and D. W. Preece, 1982. Creep Testing of Salt: Procedures, Problems and Suggestions, SAND81-1116C, Sandia National Laboratories, Albuquerque, New Mexico, presented at the First Conference on the Mechanical Behavior of Salt, The Pennsylvania State University, University Park, Pennsylvania.

Wawersik, W. R., and C. M. Stone, 1985. Application of Hydraulic Fracturing to Determine Virgin In Situ Stress State Around Waste Isolation Pilot Plant---In Situ Measurements, SAND85-1776, Sandia National Laboratories, Albuquerque, New Mexico.

Wawersik, W. R., and C. M. Stone, 1986. "Experience with Hydraulic Fracturing Tests for Stress Measurements in WIPP," SAND85-2273C, Sandia National Laboratories, Albuquerque, New Mexico, published in the Proceedings of the 27th U.S. Symposium on Rock Mechanics.

Wawersik, W. R., and C. M. Stone, 1988. "Character and Interpretation of Pressure Records in Anelastic Rock with Examples of Hydraulic Fracturing Tests in Salt," SAND87-2569A, Sandia National Laboratories, Albuquerque, New Mexico. Proposed for presentation at the Second International Workshop on Hydraulic Fracturing Stress Measurements, Minneapolis, Minnesota.

Wayland, J. R., and L. Bertholf, 1980. A Comparison of Thermomechanical Calculations for Highly Non-Linear Quasistatic Drift Formations, SAND80-0149, Sandia National Laboratories, Albuquerque, New Mexico.

Wayland, J. R., and S. B. Pahua, 1979. Far Field Thermal Calculations for the WIPP Site in Southeastern New Mexico, SAND79-1013C, Sandia National Laboratories, Albuquerque, New Mexico, presented at the Symposium on the Scientific Basis for Nuclear Waste Management, Boston, Massachusetts.

Weart, W. D., 1977. Beneficial Uses and Disposal of Radioactive Wastes, SAND77-0375C, Sandia National Laboratories, Albuquerque, New Mexico.

Weart, W. D., 1977. New Mexico Waste Isolation Pilot Plant, Mid 1977, Status Report, SAND77-0994, Sandia National Laboratories, Albuquerque, New Mexico.

Weart, W. D., 1978. Evaluation of Salt Beds in New Mexico as a Geologic Repository for Nuclear Waste, SAND78-2086, Sandia National Laboratories, Albuquerque, New Mexico.

Weart, W. D., 1978. Geotechnical Evaluation of the Proposed WIPP Site in Southeast New Mexico, SAND78-2194C, Sandia National Laboratories, Albuquerque, New Mexico.

Weart, W. D., 1979. Meeting Evaluation of the Proposed WIPP Site in Southeast New Mexico, SAND79-1113C, Sandia National Laboratories, Albuquerque, New Mexico, presented at the 87th National American Institute of Chemical Engineers, Boston, Massachusetts.

Weart, W. D., 1979. Status Report on the Waste Isolation Pilot Plant, SAND79-0323A, Sandia National Laboratories, Albuquerque, New Mexico, presented at Trans ANS Special Session on Geologic Repositories, Atlanta, Georgia.

Weart, W. D., 1979. "Status Report on the Waste Isolation Pilot Plant," SAND79-1992A, published in BASALT 79 Annual Report, Second Annual Meeting, Basalt Waste Isolation Program, Richland, Washington.

Weart, W. D., 1979. Status of the WIPP: A Bedded Salt Repository for Defense Radioactive Waste in Southeastern New Mexico, SAND79-1904A, Sandia National Laboratories, Albuquerque, New Mexico, presented to the 1979 Fall Meeting of the Rio Grande Chapter Health Physics Society, Santa Fe, New Mexico.

Weart, W. D., 1979. WIPP: A Bedded Salt Repository for Defense Radioactive Waste in Southeastern New Mexico, SAND78-0934C, Sandia National Laboratories, Albuquerque, New Mexico, presented at the Chemical Society Meeting, Miami, Florida.

Weart, W. D., 1979. WIPP: A Repository for Defense Radioactive Waste in Southeastern New Mexico Salt Beds, SAND79-1207A, Sandia National Laboratories, Albuquerque, New Mexico.

Weart, W. D., 1980. Status of the Waste Isolation Pilot Plant Project, SAND80-0134A, Sandia National Laboratories, Albuquerque, New Mexico, presented at AIME Meeting, Carlsbad Potash Section of the AIME, Carlsbad, New Mexico.

Weart, W. D., 1980. Supplement No. 1 to Field Operations Plan for WIPP 31 Re-entry, Sandia National Laboratories, Albuquerque, New Mexico.

Weart, W. D., 1980. Technical Developments in the WIPP Program, SAND80-1599C, Sandia National Laboratories, Albuquerque, New Mexico, presented at Association of Engineering Geologists 1980 Annual Meeting, Dallas, Texas.

Weart, W. D., 1980. "Technical Issues in the Geologic Disposal of Radioactive Wastes," SAND80-03233C, Sandia National Laboratories, Albuquerque, New Mexico, published in Institute of Environmental Sciences 1980 Proceedings 26th Annual Technical Meeting, Life Cycle Problems and Environmental Technology, Philadelphia, Pennsylvania.

Weart, W. D., 1983. Conclusions Regarding Geotechnical Acceptability of the WIPP Site, SAND83-0226C, Sandia National Laboratories, Albuquerque, New Mexico, presented at the Waste Management 1983 Symposium, Tucson, Arizona.

Weart, W. D., 1983. Summary of Evaluation of the Waste Isolation Pilot Plant (WIPP) Site Suitability, SAND83-0450, Sandia National Laboratories, Albuquerque, New Mexico.

Weart, W. D., 1983. WIPP: the Solution to Disposal of Defense Transuranic Waste, SAND83-0062A, Sandia National Laboratories, Albuquerque, New Mexico, presented at the 1983 Annual Meeting, American Nuclear Society, Detroit, Michigan.

Weart, W. D., 1985. Status of the WIPP, a Defense Transuranic Waste Isolation Facility, SAND85-0736J, Sandia National Laboratories, Albuquerque, New Mexico, Proceedings of the National Council on Radiation Protection and Measurements.

Weart, W. D., and D. W. Powers, 1980. Geologic Issues for the Waste Isolation Pilot Plant (WIPP), A Revised Abstract for the Geological Society of America Symposium, SAND80-1269C, Sandia National Laboratories, Albuquerque, New Mexico.

Weart, W. D., and D. W. Powers, 1981. "Geologic Issues for the Waste Isolation Pilot Plant (WIPP)," Abstract, Geological Society of America, Vol. 12, No. 7, p. 546.

WEC (Westinghouse Electric Corporation), 1988. "Determination of Hydrogen Flow and Diffusion Properties of Selected Graphite Filters," Westinghouse Research and Development Center, Pittsburgh, Pennsylvania.

WEC (Westinghouse Electric Corporation), 1988. "Final Report for the Contact-Handled Transuranic Waste Mock Retrieval Demonstration," prepared for the Department of Energy, January 28, DOE/WIPP 88-006.

WEC (Westinghouse Electric Corporation), 1988. "Plant Work Request 08-1644, Williams Hollow Core Pull Test (4/19/88), Interim Report (6/14/88)," Waste Isolation Pilot Plant, Carlsbad, New Mexico.

WEC (Westinghouse Electric Corporation), 1988. WIPP Radiation Safety Manual, WP-12-5, Waste Isolation Pilot Plant, Carlsbad, New Mexico.

Whelan, G., D. L. Strenge, J. G. Droppe, Jr., B. L. Steelman, and J. W. Burk, 1987. The Remedial Action Priority System (RAPS): Mathematical Formulation, DOE/RL/87-09, PNL-6200, UC-11, Pacific Northwest Laboratory.

Wicks, G. G., 1985. WIPP/SRL In Situ and Laboratory Testing Programs---Part I: MIIT Overview, Nonradioactive Waste Glass Studies, DP-1706, Savannah River Laboratory, Aiken, South Carolina.

Wicks, G. G., M. E. Weinle, and M. A. Molecke, 1987. WIPP/SRL In Situ Tests Part II: Pictorial History of MIIT and Final MIIT Matrices, Assemblies, and Sample Listings, DP-1733 (also SAND86-2720), Savannah River Laboratory, Aiken, South Carolina.

Wilson, W. C., 1980. Field Operations Plan of Sandia National Laboratories Potash Core Test, Sandia National Laboratories, Albuquerque, New Mexico.

Winslow, C. D., 1981. The Sorption of Cesium and Strontium from Concentrated Brines by Backfill Barrier Materials, SAND80-2046, Sandia National Laboratories, Albuquerque, New Mexico.

Woolfolk, S. W., 1982. Radiological Consequences of Brine Release by Human Intrusion into WIPP, TME-3151, U.S. Department of Energy, Albuquerque, New Mexico.

Wowak, W. E., 1979. Early In Situ Measurements Program for the Waste Isolation Pilot Plant, SAND79-0362, Sandia National Laboratories, Albuquerque, New Mexico.

Wowak, W. E., and A. R. Sattler, 1979. Criteria and Preliminary Design for WIPP Exploratory Program Experiments, SAND79-2077, Sandia National Laboratories, Albuquerque, New Mexico.

Yost, F. G., and E. A. Aronson, 1987. Crushed Salt Consolidation Kinetics, SAND87-0264, Sandia National Laboratories, Albuquerque, New Mexico.

Zerwekh, A., S. T. Kosiewicz, and B. Barraclough (Los Alamos Scientific Laboratory), 1977-78. Experimental Studies of the Degradation of RAD-Wastes for the Sandia Laboratories Waste Isolation Pilot Plant, Progress Reports: LA-7246-PR, LA-7259-PR, LA-7478-PR, Los Alamos, New Mexico.

Zeuch, D. H., 1984. Analysis of Consolidation of Crushed Rock Salt Backfill Using a Plastic Flow Model for Isostatic Hot Pressing, SAND84-1599A, Sandia National Laboratories, Albuquerque, New Mexico, presented at American Geophysical Union Fall Meeting, San Francisco, California.

Zeuch, D. H., D. J. Holcomb, and H. S. Lauson, 1985. Analysis of Consolidation of Granulated Rock Salt Using a Plastic Flow Model for Isostatic Hot Pressing, SAND84-1106, Sandia National Laboratories, Albuquerque, New Mexico.

ZoBell, C. E., and M. A. Molecke, 1978. Survey of Microbial Degradation of Asphalts with Notes on Relationship to Nuclear Waste Management, SAND78-1371, Sandia National Laboratories, Albuquerque, New Mexico.

J.8 **RESOURCES**

AIM (Agricultural and Industrial Minerals, Inc.), 1979. Resource Study for the Waste Isolation Pilot Plant Eddy County, New Mexico, AIM, Inc., San Carlos, California.

Biederman, P. L., 1984. National Resources of the Waste Isolation Pilot Plant Site, Energy and Minerals Department, Task Force on National Resources, State of New Mexico.

Braush, L. M., A. K. Kuhn, and J. K. Register, 1982. Natural Resources Study, TME 3156, prepared for U.S. Department of Energy, WIPP Project Office.

Cheeseman, R. J., 1978. "Geology and Oil/Potash Resources of Delaware Basin, Eddy and Lea Counties," in Geology and Mineral Deposits of Ochoan Rocks in Delaware Basin and Adjacent Areas, New Mexico Bureau of Mines and Mineral Resources Circular 159, pp. 7-14.

EEG (Environmental Evaluation Group), 1982. Potential Release Scenario and Radiological Consequence Evaluation of Mineral Resources at WIPP, EEG-12, M. S. Little, Contract No. AF (401)-4423 OROWL, Health and Environment Department, State of New Mexico, Santa Fe, New Mexico.

Foster, R. W., 1974. Oil and Gas Potential of a Proposed Site for Disposal of High-Level Radioactive Waste, New Mexico Bureau of Mines Mineral Resource Open-File Report.

Frick, T. C., and R. W. Taylor, eds., 1962. Petroleum Production Handbook, Vol. II, Society of Petroleum Engineers of AIME, Dallas, Texas.

Hendrickson, G. E., and R. J. Jones, 1952. Geology and Ground Water Resources of Eddy County, New Mexico, Ground Water Report 3, Bureau of Mines and Mineral Resources, New Mexico Institute of Mining and Technology.

John, C. B., R. J. Cheeseman, J. C. Lorenz, and M. L. Millgate, 1978. Potash Ore Reserves in the Proposed Waste Isolation Pilot Plant Area, Eddy County, Southeastern New Mexico, U.S. Geological Survey Open-File Report 78-828.

Jones, C. L., 1954. "The Occurrence and Distribution of Potassium Minerals in Southeastern New Mexico," New Mexico Geological Society Guidebook, 5th Field Conference, Southeastern New Mexico, ed. T. F. Stiff.

Jones, C. L., 1972. "Permian Basin Potash Deposits, Southwestern United States," in Geology of Saline Deposits, UNESCO, Earth Science Series No. 7, pp. 191-101.

Jones, C. L., 1975. Potash Resources in Part of Los Medanos Area of Eddy and Lea Counties, New Mexico, U.S. Geological Survey Open-File Report 75-407.

Jones, C. L., 1978. Test Drilling for Potash Resources, Waste Isolation Pilot Plant Site, Eddy County, New Mexico, U.S. Geological Survey Open-File Report 78-592, Vols. 1 and 2.

Jones, C. L., C. G. Bowles, and K. G. Bell, 1960. Experimental Drill Hole Logging in Potash Deposits of the Carlsbad District, New Mexico, U.S. Geological Survey Open-File Report 60-84.

Keesey, J. J., 1976. Hydrocarbon Evaluation, Proposed Southeastern New Mexico Radioactive Storage Site, Eddy County, New Mexico, Vols. 1 and 2, report to Sandia National Laboratories, Albuquerque, New Mexico.

Keesey, J. J., 1979. Evaluation of Directional Drilling for Oil and Gas Reserves Underlying the WIPP Site Area, Eddy County, New Mexico, Sipes, Williamson, & Associates, Midland, Texas.

Mariah Associates, Inc., 1987. Report of Class II Survey and Testing of Cultural Resources at the WIPP Site at Carlsbad, New Mexico, prepared for the U.S. Army Corps of Engineers, Albuquerque District, New Mexico.

McMillan, C., 1976. Los Medanos Area: Permian Exploration Co., report to Sandia National Laboratories, Albuquerque, New Mexico.

Netherland, Sewell, and Associates, Inc., 1974. Evaluation of Hydrocarbon Potential, AEC Study Area, Southeastern New Mexico, prepared for Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Netherland, Sewell, and Associates, Inc., 1974. Hydrocarbon Distribution Map, ORNL Project 78X 382 84V, Eddy and Lea Counties, New Mexico, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Schaller, W. T., and E. P. Henderson, 1932. "Mineralogy of Drill Cores From the Potash Field of New Mexico and Texas," U.S. Geological Survey Bulletin No. 833.

Siemers, W. T., J. W. Hawley, C. Rautman, and C. Austin, 1978. "Evaluation of the Mineral Potential (Excluding Hydrocarbons, Potash and Water) of the Waste Isolation Pilot Plant Site, Eddy County, New Mexico," New Mexico Bureau of Mines and Mineral Resources, report to Sandia National Laboratories, Albuquerque, New Mexico.

Sipes, Williamson, and Aycock, Inc., 1976. "Hydrocarbon Evaluation, Proposed Southeastern New Mexico Radioactive Material Storage Site, Eddy County, New Mexico," report submitted to Sandia National Laboratories, Albuquerque, New Mexico.

Smith, A. R., 1978. "Sulfur Deposits in Ochoan Rocks of Southeast New Mexico and West Texas," in Geology and Mineral Deposits of Ochoan Rocks in Delaware Basin and Adjacent Areas, New Mexico Bureau of Mines and Mineral Resources Circular 159, pp. 71-76.

U.S. Bureau of Land Management, 1975. Preliminary Regional Environmental Analysis Record--Potash Leasing in Southeastern New Mexico, Department of the Interior, New Mexico State Office.

U.S. Bureau of Land Management, 1977. East Eddy Planning Unit: Unit Resource Analysis, Department of the Interior, Roswell, New Mexico.

U.S. Bureau of Land Management (U.S. Bureau of Mines), 1977. Valuation of Potash Occurrences Within the Waste Isolation Pilot Plant Site in Southeastern New Mexico, report to the Energy Research and Development Administration.

U.S. Geological Survey, 1979. Preliminary Map Showing Distribution of Potash Resources, Carlsbad Mining District, Eddy and Lea Counties, New Mexico, OFR 79-1579.

U.S. Soil Conservation Service, 1971. Soil Survey of Eddy Area, New Mexico, U.S. Department of Agriculture, Government Printing Office, Washington, D.C.

U.S. Soil Conservation Service, 1974. Soil Survey of Lea County, New Mexico, U.S. Department of Agriculture, Government Printing Office, Washington, D.C.

J.9 **SOCIOECONOMICS**

Adcock, L., 1978. An Outline of the Input-Output Model and the Impact Projections Methodology, Tech Document Socioeconomic Portion Waste Isolation Pilot Environmental Impact Report, SAND78-7061, Sandia National Laboratories, Albuquerque, New Mexico.

Adcock & Associates, 1980. Sociocultural Analysis for the Proposed Waste Isolation Pilot Plant, SAND80-7155, Sandia National Laboratories, Albuquerque, New Mexico.

Adcock & Associates, 1980. Socioeconomics Study for Proposed Waste Isolation Pilot Plant, SAND80-7123, Sandia National Laboratories, Albuquerque, New Mexico.

Adcock L., R. R. Lansford, and A. Turpin, 1989. "The Economic Impact of the Waste Isolation Pilot Plant in Southeastern New Mexico," (not available in print).

Adcock L., E. J. Nowak, and J. Myers, 1983. "Socioeconomic Impact Review Waste Isolation Pilot Plant (WIPP) Project," Carlsbad, New Mexico, unpublished draft.

Lansford, R. R., J. A. Diemer, E. M. Jaramillo, A. Turpin, and V. Devers, 1988. "The Economic Impact of Waste Isolation Pilot Plant on Southeastern New Mexico FY 1987," draft, New Mexico State University, Las Cruces, New Mexico.

Lansford, R. R., J. A. Diemer, E. M. Jaramillo, A. Turpin, and V. Devers, 1988. "The Social and Economic Impact of the Department of Energy on the State of New Mexico, FY 1987," New Mexico State University, Las Cruces, New Mexico.

J.10 **TRANSPORTATION**

ANSI (American National Standards Institute), 1987. American National Standard for Radioactive Materials - Leakage Tests on Packages for Shipment, ANSI 14.5 -1987, New York.

Cashwell, J. W., et al., 1986. Transportation Impacts of the Commercial Radioactive Waste Management Program, Appendix 4, Tables 4-4A and 4-4B, SAND 85-2715, TTC-0663, Sandia National Laboratories, Albuquerque, New Mexico.

DOE (U.S. Department of Energy), 1986. Transuranic Waste Transportation Assessment and Guidance Report, DOE/JIO-002, Revision 1, Carlsbad, New Mexico.

DOE (U.S. Department of Energy), 1987. The DOE Evaluation Document for DOT Type 7A, Type A Packaging, MIL 3245/DOE/DP 0053-HI, Mound Applied Technology, Dayton, Ohio.

DOE (U.S. Department of Energy), 1988. Waste Isolation Pilot Plant Emergency Plan, WP 12-7, Revision 4, WIPP Project Office, Carlsbad, New Mexico.

DOE (U.S. Department of Energy), 1989. Emergency Preparedness for Transportation Incidents Involving Radioactive Materials, SAIC 89/1354, Science Application International Corporation, Oak Ridge, Tennessee.

DOE (U.S. Department of Energy), 1989. Radioactive Material Transportation Emergency Response Orientation.

DOE (U.S. Department of Energy), 1989. Safety Analysis Report for the TRUPACT-II Shipping Package, Revision 2, Docket No. 71-9218, prepared for the U.S. Department of Energy by Nuclear Packaging, Inc., Washington, D.C.

DOE (U.S. Department of Energy), 1989. The Quality Assurance Plan for the Transportation and Receipt of Transuranic (TRU) Waste, draft, DOE/WIPP-89-012, Revision 0, Albuquerque, New Mexico.

DOE (U.S. Department of Energy), 1989. TRUPACT-II Content Codes (TRUCON), DOE/WIPP-89004, Revision 1, Albuquerque, New Mexico.

DOT (U.S. Department of Transportation), 1987. Emergency Response Guidebook, DOT p. 5800.4.

EEG (Environmental Evaluation Group), 1983. Potential Problems from Shipment of High-Curie Content Contact-Handled Transuranic, (CH-TRU) Waste to WIPP, EEG-24, R. H. Neill and J. K. Channell, Health and Environment Department, State of New Mexico, Santa Fe, New Mexico.

FEMA (U.S. Federal Emergency Management Agency), 1985. Federal Radiological Emergency Response Plan.

FEMA (U.S. Federal Emergency Management Agency), 1988. Guidance for Developing State and Local Radiological Emergency Response Plans and Preparedness for Transportation Accidents, draft revision, FEMA-REP-5.

Foster, B., 1984. Transportation of Spent Fuel, High-Level Wastes, and Transuranic Wastes, SAND84-7113, TTC-0475, Sandia National Laboratories, prepared for the National Conference of State Legislatures (NCSL), Denver, Colorado.

FRA (Federal Railroad Administration), 1988. Accident/Incident Bulletin No. 156: Calendar Year 1987, Office of Safety, FRA, U.S. Department of Transportation, Washington, D.C.

Huerta, M., et al., 1983. Analysis, Scale Modeling, and Full-Scale Test of Low-Level Nuclear Waste Drum Response to Accident Environments, SAND80-2517, Sandia National Laboratories, Albuquerque, New Mexico.

IAEA (International Atomic Energy Agency), 1985. Explanatory Material for the IAEA Regulations for the Safe Transport of Radioactive Material, second edition, Safety Series No. 7, Vienna, Austria.

Joy, D. S., et al., 1982. Highway, a Transportation Routing Model: Program Description and User's Manual, ORNL/TM 84-19, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Madsen, M. M., et al., 1983. RADTRAN II User Guide, SAND82-2681, TTC-0399, Sandia National Laboratories, Albuquerque, New Mexico.

McClure, D., and E. Luna, 1987. Public Concerns About RAM Transport--Communicating Engineering Data on Risk, SAND87-2656A, Sandia National Laboratories, Albuquerque, New Mexico.

Mishma, J., and Schwendiaman, L.C., 1973. Some Experimental Measurements of Airborne Uranium (Representing Plutonium) in Transportation Accidents, BNWL-1732, Pacific Northwest Laboratories.

Mitter, E. L., R. D. Hume, F. J. Vilardo, E. Felgenbaum, H. Briggs and R. Piercy, 19__. Survey of Current State Radiological Emergency Response Capabilities for Transportation Related Incidents, School of Public and Environmental Affairs, Indiana University, NUREG/CR-1620, U.S. Nuclear Regulatory Commission, Washington, D.C.

NRC (Nuclear Regulatory Commission), 1975. Leakage Tests on Packages for Shipment of Radioactive Materials, Regulatory Guide 7.4., Washington, D.C.

NRC (Nuclear Regulatory Commission), 1987. Shipping Container Response to Severe Highway and Railway Accident Conditions, NUREG/CR-4829, Vols. 1 and 2, Lawrence Livermore National Laboratory, Livermore, California.

Peterson, B. E., 1984. INTERLINE: A Railroad Routing Model, ORNL/TM-8944, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Rand McNally, 1987. Rand McNally Road Atlas, Rand McNally & Company.

Rand McNally, 1988. Rand McNally Handy Railroad Atlas of the United States, Rand McNally & Company.

Rao, R. K., E. L. Wilmont, and R. E. Luna, 1982. Nonradiological Impacts of Transporting Radioactive Material, SAND81-1703, TTC-0236, Sandia National Laboratories, Albuquerque, New Mexico.

Sandia National Laboratories, 1983. Transporting Spent Reactor Fuel--Allegations and Responses, SAND82-2778, Albuquerque, New Mexico.

Shirley, C. G., 1983. "A Simple Risk Analysis Tool for Estimating Release Fractions From Shipments of 55-Gallon Drums Involved in Transportation Accidents," Seventh International Symposium on Packaging and Transportation of Radioactive Materials (PATRAM 83), New Orleans, Louisiana.

Taylor, J. M., and S. L. Daniel, 1977. RADTRAN: A Computer Code to Analyze Transportation of Radioactive Material, SAND76-0243, TTC-0239, Sandia National Laboratories, Albuquerque, New Mexico.

Taylor, J. M., and S. L. Daniel, 1982. RADTRAN II: Revised computer Code to Analyze Transportation of Radioactive Material, SAND80-1943, TTC-0239, Sandia National Laboratories, Albuquerque, New Mexico.

Wilmont, E. L., M. M. Madsen, J. W. Cashwell, and D. S. Joy, 1983. A Preliminary Analysis of the Cost and Risk of Transporting Nuclear Waste to Potential Candidate Repository Sites, SAND83-0867, TTC-0434, Sandia National Laboratories, Albuquerque, New Mexico.

Wolf, T. A. 1984. The Transportation of Nuclear Materials, SAND84-0062, Sandia National Laboratories, Albuquerque, New Mexico.

J.11 **WASTE TREATMENT**

Berglund, G. R., 1989. "PREPP Rotary Kiln Plant Performance Overview," Proceedings of the 1989 Incineration Conference, Knoxville, Tennessee, May 1989.

Bibler, N. E., 1978-81. Gas Production from Alpha Radiolysis of Concrete Containing TRU Incineratory Ash, Progress Reports: DPST-78-150-1, DPST-78-150-2, DPST-80-150-1 and DPST-80-150-2, Savannah River Laboratory, Aiken, South Carolina.

Bibler, N. E., 1979. Radiolytic Gas Generation in Concrete Made with Incinerator Ash Containing Transuranium Nuclides, DP-MS-79-25, Savannah River Laboratory, Aiken, South Carolina, presented at Materials Research Society, Boston, Massachusetts, November, 1979, published in Scientific Basis for Nuclear Waste Management, Vol. 2.

Brunner, H., and B. Christ, 1985. "Recent Experiences with Cement Solidification Systems," Proceedings of the Symposium on Waste Management at Tucson, Arizona, March, Vol. II, pp. 493-496.

DOE (U.S. Department of Energy), undated. "Destroying Toxic Wastes by Incineration," in Waste Management Technology, Oak Ridge, Tennessee.

Dole, L. R., 1985. Overview of the Applications of Cement-Based Immobilization Technologies Developed at U.S. DOE Facilities, Martin Marietta Energy Systems, Inc., CONF-850314-62.

Holton, L. K., R. D. Dierks, R. W. Goles, Y. B. Katayama, J. E. Surma, and N. M. Thomas, 1988. "Operating Experience in a Radioactive Liquid-fed Ceramic Melter Vitrification Facility," Proceedings of the Symposium on Waste Management at Tucson, Arizona, February, Vol. II, pp. 217-227.

Jolley, R. L., and B. R. Rodgers, 1987. "A Survey of Low-Level Radioactive Waste Treatment Methods and Problem Areas Associated with Commercial Nuclear Power Plants," Proceedings of the Symposium on Waste Management at Tucson, Arizona, March, Vol. III, p. 675-680.

Klein, W. J., 1983. "Development of Process Systems for Radioactive Incinerator Ash," Proceedings of the Symposium on Waste Management at Tucson, Arizona, February, Vol. I, pp. 457-463.

Klingler, L. M., and K. M. Armstrong, 1986. Application of a Glass Furnace System to Low Level Radioactive and Mixed Waste Disposal, MLM-3351-OP, U.S. Department of Energy.

Schneider, V. W., and F. W. Lederbrink, 1982. "Treatment of TRU Wastes: Recent Results of Developments Underway in Germany," Proceedings of the American Nuclear Society Topical Meeting on the Treatment and Handling of Radioactive Wastes at Richland, April, pp. 362-366.

Tait, T. D., 1983. Demonstration Test Assessment of the Slagging Pyrolysis Incinerator for Processing INEL Transuranic Waste, EGG-TF-6192, EG&G Idaho, Inc., Idaho Falls, Idaho.

Trigilio, G., 1981. Volume Reduction Techniques in Low Level Radioactive Waste Management, NUREG/CR-2206, U.S. Nuclear Regulatory Commission, Washington, D.C., September.

Vavruska, J. S., L. C. Borduin, D. A. Hutchins, and C. L. Warner, 1989. "Los Alamos Controlled Air Incinerator Upgrade for TRU/Mixed Waste Operations," Proceedings of 1989 Incinceration Conference, Knoxville, Tennessee, May.

Williams, S., and D. L. Charlesworth, 1988. "Development of a Plutonium-239 Recovery Incinerator," Proceedings of the Symposium on Waste Management at Tucson, Arizona, Vol. II, pp. 501-506.

Zerwekh, A., 1979. "Gas Generation for Radiolytic Attack of TRU-Contaminated Hydrogenous Waste," LA-7674-MS, Los Alamos National Laboratory, Los Alamos, New Mexico.

APPENDIX J

BIBLIOGRAPHY

TABLE OF CONTENTS

NOTE TO THE READER:

Appendix J has not been reprinted in this final SEIS. The reader is referred to the draft SEIS for the complete appendix.

Appendix J contains a bibliography related to the WIPP. The list contains various writings about the WIPP, not merely those referenced in the draft and final SEIS. The citations are organized into seven subject areas:

1. Design Development
2. Environmental
3. Geochemistry
4. Geology
5. Hydrology
6. Repository
7. Resources

For additional bibliographic citations, see the reference lists following each section and appendix of this final SEIS.

APPENDIX K

DOE READING ROOMS AND PUBLIC LIBRARIES

TABLE OF CONTENTS

LIST OF TABLES

K.1 **INTRODUCTION**

This appendix contains lists of the DOE public reading rooms and public libraries that will receive copies of this SEIS, including appendices, comment response volumes, and copies of the hearing transcripts, exhibits, and written documents received in response to the draft SEIS (Tables K.1.1 and K.1.2). As noted on the tables, the DOE public reading rooms plus public or university libraries in the cities of Carlsbad, Albuquerque, and Santa Fe, New Mexico and Denver and Boulder, Colorado have available complete sets of the supporting documents referenced in this SEIS.

TABLE K.1.1 Location of DOE public reading rooms receiving SEIS documents and references[a]

U.S. Department of Energy-HQ
Public Reading Room
Room 1E-190 Forrestal Building
1000 Independence Ave., SW
Washington, DC 20585
(202-586-6020)

U.S. Department of Energy-ID
Public Reading Room
University Place
1776 Science Center Drive
Idaho Falls, ID 83402
(208-526-1144)

U.S. Department of Energy-NV
Public Reading Room
2753 South Highland Street
Las Vegas, NV 89109
(702-295-1274)

U.S. Department of Energy-OR
Public Reading Room
Federal Building
200 Administration Road
Oak Ridge, TN 37830
(615-576-1216)

U.S. Department of Energy
National Atomic Museum
Public Reading Room
Wyoming Boulevard South
Kirtland Air Force Base
Albuquerque, NM 87115
(505-844-4378)

U.S. Department of Energy-RL
Public Reading Room
Hanford Science Center
825 Jadwin Avenue
Richland, WA 99352
(509-376-8583)

U.S. Department of Energy-SR
Public Reading Room
University of South Carolina - Aiken
Gregg - Graniteville Library
171 University Parkway
Aiken, SC 29801
(803-648-6851; ext. 3320)

U.S. Department of Energy-SFO
Public Reading Room
1333 Broadway, 7th Floor
Oakland, CA 94612
(415-273-4428)

U.S. Department of Energy-CH
Public Reading Room
9800 South Cass Avenue, Building 201
Argonne, IL 60439
(312-972-2010)

[a] Complete sets of the supporting documents referenced in this SEIS are available at these locations.

TABLE K.1.2 Location of public libraries receiving SEIS documents

Alabama Public Library Service
6030 Monticello Drive
Montgomery, AL 36130
(205-277-7330)

Arkansas State Library
Document Services
1 Capitol Mall
Little Rock, AR 72201
(501-682-1527)

Arizona Library
Federal Documents
Department of Library Archives and
 Public Records
1700 W. Washington
Phoenix, AZ 85007
(602-542-4121)

California State Library
Library and Courts Building
914 Capitol Mall
Sacramento, CA 95814
(916-324-4863)

Government Publications
Norlin Library
University of Colorado/Boulder
Boulder, CO 80309
(303-492-8834)

Denver Public Library[a]
Government Documents Department
Second Floor
1357 Broadway
Denver, CO 80203-2165
(303-571-2000)

Atlanta-Fulton Public Library
Ivan Allen Department
Central Library
1 Margaret Mitchell Square
Atlanta, GA 30303
(404-730-1900)

Idaho State Library
325 W. State Street
Boise, ID 83702
(208-334-5124)

Illinois State Library
350 Centennial Building
Springfield, IL 62756
(271-782-5430)

Indiana State Library
140 N. Senate Avenue
Indianapolis, IN 46204
(317-232-3675)

State Library of Louisiana
760 Riverside North
Baton Rouge, LA 70821
(504-342-4923)

Missouri State Library
Federal Documents Office
2002 Missouri Blvd.
Jefferson City, MO 65109
(314-751-4552)

Mississippi Library Commission
1221 Ellis Avenue
Jackson, MS 39209
(601-359-1036)

Nevada State Library and Archives
Federal Documents
401 N. Carson Street
Carson City, NV 89710
(702-885-5160) (800-922-2880)

Albuquerque Public Library
501 Copper NW
Albuquerque, NM 87102
(505-768-5140)

Zimmerman Library [a]
Government Publications
University of New Mexico
Roma Avenue and Yale Boulevard
Albuquerque, NM 87131
(505-277-5441)

Carlsbad Public Library [a]
Public Document Room
101 South Halagueno Street
Carlsbad, NM 88220
(505-885-6776)

[a] Complete sets of the supporting documents referenced in this SEIS are available at these locations.

TABLE K.1.2 Concluded

El Rito Public Library
PO Box 5
El Rito, NM 87530
(None)

Pannell Library
New Mexico Junior College
5317 Lovington Highway
Hobbs, NM 88240
(505-392-4510)

Thomas Branigan Memory Library
200 East Picacho
Las Cruces, NM 88001
(505-526-1045)

Roswell Public Library
301 N. Pennsylvania
Roswell, NM 88201
(505-622-7101)

New Mexico State Library[a]
325 Don Gaspar
Santa Fe, NM 87503
(505-827-3827)

Santa Fe Public Library [a]
145 Washington Avenue
Santa Fe, NM 87501
(505-984-6780)

New Mexico Tech Library
Campus Station
Socorro, NM 87801
(505-835-5614)

Ohio State Library Board
Documents Department
65 S. Front Street
Columbus, OH 43266
(614-644-7051)

Oklahoma Department of Libraries
200 NE 18th Street
Oklahoma City, OK 73105
(405-521-2502)

Oregon State Library
State Library Building
Court and Summer Streets
Salem, OR 97310
(503-378-4277)

South Carolina State Library
1500 Senate Street
Columbia, SC 29201
(803-734-8666)

Texas State Library
Information Services Division
1201 Brazos Street
Austin, TX 78701
(512-463-5460)

Utah State Library
2150 South 300 West
Suite 16
Salt Lake City, UT 84115
(801-466-5888)

Washington State Library
16th and Water Streets
Olympia, WA 98504
(206-753-5590)

Wyoming State Library
Government Documents
Supreme Court Building
2301 Capitol Avenue
Cheyenne, WY 82002
(307-777-6333)

[a] Complete sets of the supporting documents referenced in this SEIS are available at these locations.

APPENDIX L

CONTAINERS AND CASKS FOR SHIPPING TRU WASTE

TABLE OF CONTENTS

LIST OF FIGURES

LIST OF TABLES

L.1 **INTRODUCTION**

This appendix was prepared in response to comments on the draft SEIS. It provides information that supplements Subsection 3.1.1.3, which discusses the shipping containers and casks to be used for transporting TRU waste to the WIPP. It discusses both the TRUPACT-II container, which will be used to transport contact-handled TRU waste, and the NuPac 72B cask, which will be used to transport remotely handled TRU waste. The discussions include descriptions of the TRUPACT-II and the NuPac 72B designs, but they are mainly directed at the certification of these designs by the U.S. Nuclear Regulatory Commission (NRC) and the analysis and tests necessary to obtain the certification.

The design of the TRUPACT-II was certified by the NRC on August 30, 1989. This appendix presents a detailed discussion of the NRC requirements for the designs to be certified. It further describes how compliance has been demonstrated for the TRUPACT-II container and how it will be demonstrated for the NuPac 72B cask. Also discussed are the NRC's requirements for the fabrication, operation, and maintenance of the shipping containers or casks, including restrictions on the waste to be transported. The last section describes quality assurance for the TRUPACT-II and NuPac 72B programs.

The initial Certificate of Compliance for the TRUPACT-II by the NRC limits shipments to only certain waste forms (see Annex 1 to this appendix). In the future, the DOE will apply to the NRC to amend the Certificate of Compliance to include other TRU waste forms known to exist.

Most of the information in this appendix was obtained from the Safety Analysis Report for the TRUPACT-II container (DOE, 1989a), the TRUPACT-II Operation and Maintenance Manual (DOE, 1989b), and the Quality Assurance Plan for the Transportation and Receipt of Transuranic (TRU) Waste (DOE, 1989c).

L.2 THE TRUPACT-II SHIPPING CONTAINER

The TRUPACT-II container will be used for shipping contact-handled (CH) TRU waste. It has been designed and constructed to meet the regulations issued by the NRC for "Type B packaging"[1] in 10 CFR Part 71. A Type B packaging with double containment is the type of container that must be used for the transport of TRU waste containing more than 20 curies of plutonium per package. A certificate stating that the TRUPACT-II complies with the NRC regulations was issued by the NRC on August 30, 1989. The NRC certificate is reproduced in this appendix as Annex 1.

The TRUPACT-II shipping container has been designed to be rugged and lightweight, because these characteristics enhance the safety of transportation. The use of rugged, yet deformable, packaging features provides capabilities which prevent the release of contents if it were subjected to extreme abuse in an accident. A lightweight design allows the transport of a larger payload per shipment while meeting highway weight limits, thereby reducing the number of waste shipments.

Before proceeding with the fabrication of the TRUPACT-II containers, four full-scale containers were built and tested. One of these served as the engineering prototype; the other three were full-scale containers that were tested in accordance with the NRC's requirements for certification. In addition, a thorough analysis of the CH TRU waste was performed to establish payload-control procedures that meet NRC criteria for transport. These controls have been approved by the NRC as acceptable methods for complying with the applicable regulations for payloads.

L.2.1 DESCRIPTION OF THE TRUPACT-II SHIPPING CONTAINER

As shown in Figure L.2.1, the TRUPACT-II container is a cylinder with a flat bottom and a domed top; it is transported in an upright position. The overall dimensions of the TRUPACT-II are approximately 8 ft in diameter by 10 ft in height; the inner containment vessel is approximately 6 ft in diameter by 8 ft in height.

To provide double containment for the TRU waste, it consists of an inner containment vessel and an outer containment vessel; the latter is part of the outer containment assembly. NRC regulations require the two separate levels of containment to be used for shipments of plutonium in excess of 20 curies per container.

[1] In the NRC regulations governing the transportation of radioactive materials (10 CFR Part 71), the term "packaging" is used to mean the shipping container or cask and the term "package" is used to mean the shipping container together with its radioactive contents.

FIGURE L.2.1
CROSS SECTION OF TRUPACT-II

The inner and the outer containment vessels have removable lids that are held in place by banded lockrings and retaining tabs. The containment vessels are nonvented and are designed for a maximum normal operating pressure of 50 pounds per square inch.

The capacity of each TRUPACT-II shipping container is 7,265 lb of payload, including pallets, slip sheets, and waste, packed in either 55-gal drums or two 67-cubic-ft standard waste boxes. The maximum gross shipping weight of a loaded TRUPACT-II container is 19,250 lb. The weight of the payload is restricted to meet highway weight limits. Up to three TRUPACT-II containers may be transported in each truck shipment. They will be hauled on a custom-designed semitrailer pulled by a conventional tractor.

L.2.1.1 **Inner Containment Vessel**

The inner containment vessel is a stainless-steel pressure vessel that contains the waste payload. The payload is protected by spacers that are made of aluminum honeycomb and are located in each of the two domed heads of the inner vessel (Figure L.2.1). The lower body of the inner containment vessel has a closure ring with two grooves, each containing an O-ring seal. The upper lid of the vessel has a mating flat surface that seals against the two O-rings once the lid and the body are assembled. Compression of the O-rings between the lid and the body form a bore-type seal. As the lid is lowered onto the body, retaining tabs on a lockring slide through recesses in the mating tabs on the body. When the lid is fully engaged, the lockring can be rotated to the closed position; the lockring cannot be rotated unless the lid is correctly mated to the body. The locking mechanism secures the lid to the body, and this maintains leaktight seals under both normal and accident conditions.

L.2.1.2 **Outer Containment Assembly**

The outer containment assembly is made of stainless steel and polyurethane foam. It consists of an exterior stainless-steel shell and a stainless-steel pressure vessel, the outer containment vessel (Figure L.2.1). Between these steel shells there is a layer of fire-retardant polyurethane foam approximately 10 inches thick. The steel walls surrounding the foam layers are lined with a heat-resistant ceramic-fiber paper, which enhances the resistance of the polyurethane foam to fire damage. On the outside of this foam and ceramic fiber, the exterior stainless-steel shell acts as a protective structure and an impact limiter. This multilayered design increases the overall strength of the container and provides the ability to withstand potential accidents associated with transport.

Like the inner containment vessel, the lower body of the outer containment vessel has a seal flange ring with two grooves, each containing an O-ring seal. The upper lid of the vessel seals against the two O-ring seals of the body when assembled. The lockring secures the lid in place and maintains leaktight seals under both normal and accident conditions, providing the same containment capability as the inner vessel (double containment).

L.2.2 **NRC CERTIFICATION**[2]

[2] To be consistent with the NRC regulations, the terms "packaging" and "package" are used in this section to mean the shipping container and the shipping container loaded with radioactive waste, respectively.

The DOE agreed to have the NRC certify the designs of the shipping containers or casks used for the transport of contact-handled or remotely handled TRU waste, respectively. This agreement was stated in the second modification (August 4, 1987) to the consultation and cooperation agreement between the DOE and the State of New Mexico (see Subsection 10.2.5).

The NRC requirements for the certification of shipping containers and casks are included in 10 CFR Part 71, "Packaging and Transportation of Radioactive Materials."
There are two basic types of packagings for radioactive materials: Type A and Type B; the latter is the type that the NRC requires for the transport of the type of waste that will be sent to the WIPP. Type A packages must withstand normal conditions of transport without loss or dispersal of their radioactive contents as demonstrated through tests outlined in regulations issued by the Department of Transportation (49 CFR Part 173). Type B packaging must withstand both normal and accident transport conditions without releasing its radioactive contents. In order to transport TRU waste containing more than 20 curies of plutonium per package, the Type B packaging must have a double containment.

L.2.2.1 **Procedure for NRC Certification**

L.2.2.1.1 General Procedure

In order for the design of a packaging to be certified, the applicant (usually the developer of the packaging) must submit to the NRC a description of the package; an evaluation of the package; and a description of the quality assurance program for the design, fabrication, assembly, testing, maintenance, repairs, modification, and use of the proposed package.

The description of the package must be in sufficient detail to identify it accurately and provide a sufficient basis for evaluation. For the packaging, this description must include a number of specified items, such as the containment system, materials of construction, weights and dimensions, methods of fabrication, and lifting and tiedown devices. In addition, the description must include information about the payload. For example, it must identify the radioactive constituents of the payload and their quantity, identify fissile constituents, describe the chemical and physical form, and state the maximum heat generated by the radioactive payload.

The evaluation of the package is to consist of a demonstration that the packaging complies with the standards specified in 10 CFR Part 71. The standards in Subpart E include general design requirements (e.g., fastening devices for containment vessels, maximum surface temperatures), requirements for lifting and tiedown devices, external radiation limits, and special requirements for packages containing fissile materials or plutonium in excess of 20 curies. Subpart F specifies the evaluations that must be performed to demonstrate that the package can withstand normal and accident conditions without loss of integrity.

The evaluations of response to normal transportation conditions are to include the following: exposure to high and low temperatures, reduced and increased external pressure, vibration, and a water spray simulating a heavy rainfall; a free drop for a specified distance (referred to as a handling drop); and an impact by a vertical steel cylinder, 1-1/4 inches in diameter, dropped from a height of 40 inches onto the most vulnerable surface of the package. It is also necessary to determine and demonstrate the response of the package to accident conditions. The requirements for this evaluation are discussed in detail in the next subsection.

For the quality assurance program, the applicant must identify any established codes and standards proposed for use in the design, fabrication, assembly, testing, maintenance, and use of the package.

After the application is submitted, the NRC may at any time request additional information. The application is reviewed by the NRC's technical staff, who prepare a safety evaluation report for the particular package design. If the staff determines that all pertinent requirements are met, the NRC issues a certificate of compliance. As already mentioned, the NRC certificate of compliance for the TRUPACT-II design was issued on August 30, 1989. This certificate is reproduced in full in Annex 1 to this appendix.

The certificate of compliance specifies procedures for the fabrication, operation, and maintenance of the packaging and defines the payload that may be transported. The certificate is valid for a period of 5 years. At the end of this period, it must be renewed by submitting an application for renewal.

L.2.2.1.2 Demonstration of Ability to Withstand Accident Conditions

To be certified by the NRC as Type B (10 CFR 71.73), a candidate packaging must demonstrate resistance to the worst conditions that can be expected in a transportation accident. To simulate these hypothetical accident conditions, the NRC has specified a series of impact, thermal, and immersion tests that must be performed in a specified sequence. Acceptable packaging performance can be demonstrated by analysis, by testing, or a combination of both. In either case, the most damaging orientation for the packaging must be considered for each accident condition. In other words, the tests must be directed at the weakest part of the package. The hypothetical accident conditions and the sequence in which the tests are to be performed are as follows:

1) Free drop. A drop from a height of 30 ft onto a flat, unyielding surface in a position for which maximum damage is expected.

2) Puncture. A drop from a height of 40 inches onto a metal bar that is 6 inches in diameter and no less than 8 inches long and is mounted on an unyielding surface. This test is also to be performed in a position for which maximum damage is expected. (The DOE conducted the tests with a puncture bar that was 24 to 48 inches long, depending on the orientation of the TRUPACT-II.)

3) Heat. Exposure to a surrounding heat flux with a minimum temperature of 1475^{o}F for 30 minutes. (The TRUPACT-II test units were exposed to a fully engulfing fire to meet and exceed these requirements.)

4) Immersion. Exposure to an external pressure equivalent to immersion under at least 50 ft of water for no less than 8 hours.

On completion of these tests, the packaging must maintain its containment integrity by passing a leakage-rate test (NRC, 1975).

The Order of the Tests. The order of the tests is reasoned to be the order of events threatening the packaging in a real transportation accident: impact and puncture followed by exposure to fire. The test sequence, therefore, starts with mechanical impacts and then continues with the fire test; this sequence is designed to inflict maximum heat damage. The mechanical and heat tests are applied to the same specimen. The immersion test may be conducted on a separate specimen, because immersion in water is not likely to occur together with an impact accident (IAEA, 1987).

The Free-Drop Test Target. The free-drop test requires the package to strike an unyielding flat target after a free drop from a height of 30 ft, striking the target in a position for which maximum damage is expected. With an unyielding target all of the deformation produced by the test is transferred to the packaging. An actual accident would usually involve a target that yields somewhat, allowing much of the impact energy to be absorbed by the deformation of the target. Thus, an unyielding target forces the packaging to sustain more damage in a given set of test conditions than would a yielding target.

Unyielding targets are specially constructed to have a mass at least 10 times the mass of the package being tested. They are usually made of concrete and steel, and the concrete is often tied to bedrock through a system of steel columns, making the target very stiff or essentially immovable. The surface of the unyielding target is a steel plate that is in intimate contact with the surface of the concrete.

Tests have shown that the damage created by realistic hard targets, such as rock outcroppings or bridge abutments, would require velocities on the order of 80 miles per hour (mph) in order to be equivalent to the 30-ft drop (30 mph) on the unyielding target. For softer targets, such as other vehicles, concrete pavements, retaining walls, and earth embankments, the velocity required to produce equivalent damage exceeds 200 mph (Jefferson, 1983).

The difference between a yielding and an unyielding target can be seen in the results of two drop tests conducted for the DOE in a previous testing program. Two packagings of the same design were tested at Sandia National Laboratories. One packaging was dropped from a height of 30 ft onto an unyielding target. The second packaging was subjected to a test not required by the NRC regulations: it was dropped from a helicopter from a height of approximately 2,000 ft onto hard desert soil. This 6,700-lb package reached a terminal velocity of approximately 246 mph and was embedded in a crater approximately 8 ft deep in the desert soil. The packaging suffered no permanent deformation. The 30-ft drop onto an unyielding target caused more damage to the packaging than the 2,000-ft drop onto hard desert soil (McClure et al., 1987). (The packagings in these tests were not TRUPACT-II containers.)

The Puncture Test. Puncture loads can be expected in accidents because the surfaces that may be hit by a packaging are not always flat. The puncture tests are conducted to demonstrate the integrity of the containment even when weak points (e.g., container seals) are struck. Puncture loads can also produce a loss of the thermal insulation that protects against fires by tearing a hole in the wall of the packaging.

In the puncture test, the packaging is dropped from a height of 40 inches in a position for which maximum damage is expected. The target is the upper end of a vertical steel cylinder that is 6 inches in diameter and of a length that would cause maximum damage to the packaging. This puncture bar must be mounted on an essentially unyielding horizontal surface. The areas exposed to the puncture bar tests are subsequently exposed to the fire test (IAEA, 1987).

The Fully Engulfing Fire Test. The effects of fire on a shipping container depend on the time, the temperature, and the surface exposed. The NRC regulations require exposure to a temperature of 1475°F for 30 minutes over the entire surface of the packaging. In order to have the entire surface exposed to the fire, the packaging must be suspended approximately 4 ft above the fire surface (i.e., a burning fuel pool). The orientation of the packaging above the fuel pool is designed to provide exposure to the highest temperature. Elevating the packaging ensures that the flames are well developed at the location of the packaging, with adequate space for the lateral in-flow of air. This total surface exposure requirement encompasses such events as burning with a torch that is directed at one portion of the task. Since under most accident conditions the heavy packaging would end up on the bottom of the debris, the actual accident conditions would not duplicate the total surface exposure of the regulatory fire test (IAEA, 1987; Jefferson, 1983).

Some fires experienced in actual accident conditions burn longer than 30 minutes, but they either burn at lower temperatures (consuming slower burning materials like wood) or are concentrated over small areas, thus being insufficiently large to envelop the entire packaging. An accident that would produce a heat environment exceeding that called for in the regulations is extremely unlikely (Jefferson, 1983).

The Immersion Test. As a result of a potential for transportation accidents near or on a body of water, a packaging could be subjected to an external pressure from submersion under water. To simulate the equivalent damage from this low-probability event, the NRC regulations require that a packaging be able to withstand the external pressures resulting from submersion at reasonable depths. Engineering estimates indicate that water depths near most bridges, roadways, or harbors would be less than 50 ft. Consequently, 50 ft was selected as the immersion depth. While immersion at depths greater than 50 ft is possible, this value was selected to envelop the equivalent damage from most transportation accidents. In addition, the potential consequences of a significant release of radioactive material would be greatest near a coast or in a shallow body of water. The time of exposure was set at 8 hours, which is time enough to allow the package to come to a steady state from the rate-dependent effects of immersion (IAEA, 1987). Since the main purpose of the immersion test is to demonstrate that a packaging can maintain its structural integrity when subjected to an external pressure, a pressure test or calculation may be substituted for the actual immersion.

The Leakage-Rate Test. After these accident condition tests, a very stringent leakage-rate specification must be met by the packaging. In order to demonstrate that there will be no release of contents under normal accident conditions, both containment vessels must remain leaktight, in accordance with standard ANSI 14.5-1987 of the American National Standards Institute. The stringency of the postaccident-leaktightness standard requires the packaging design to be so robust that it would have to be subjected to an accident much more severe than those simulated in the certification tests before a release of its contents could occur.

L.2.3 **COMPLIANCE OF THE TRUPACT-II PACKAGE WITH NRC REGULATIONS**

On March 3, 1989, the developer of the TRUPACT-II shipping container submitted to the NRC, on behalf of the DOE, the documentation required for an application for certification. This documentation consisted of a comprehensive safety analysis report for the TRUPACT-II shipping container (DOE, 1989a, Rev. 2) and a document describing the codes used in the preparation and characterization of CH TRU waste. Four revisions to the Safety Analysis Report were made to supplement the document with additional information requested by the NRC and the final results of TRUPACT-II tests.

The Safety Analysis Report provides a detailed description of the TRUPACT-II design, operation, maintenance, the payload (CH TRU waste) and quality assurance programs. In addition, the report documents the performance of the TRUPACT-II container in the regulatory tests described above. The manner in which the tests were conducted and the results are discussed below.

Compliance with the evaluation requirements of 10 CFR Part 71 was demonstrated by a combination of analyses and testing of the TRUPACT-II package.

The certificate of compliance was issued by the NRC on August 30, 1989. It is reproduced in full in Annex 1 to this appendix.

L.2.3.1 **Evaluation of Performance**

As reported in Section 2.6 of the Safety Analysis Report for the TRUPACT-II Shipping Package (DOE, 1989a), the container meets the performance requirements of Subpart E of 10 CFR Part 71 for normal transportation conditions. The compliance was demonstrated through analysis and by performing the required free-drop test from a height of 3 ft. The analyses covered the response of TRUPACT-II components to heat and cold, reduced and increased external pressures, and vibration. Exposures to a water spray simulating a heavy rainfall and impact by a steel cylinder 1-1/4 inches in diameter (penetration test) were judged to be of negligible consequence because of the TRUPACT-II construction.

For the hypothetical accident conditions specified in Subpart F of 10 CFR Part 71, tests with full-scale TRUPACT-II units were conducted. The only exception was the immersion criterion, for which compliance was demonstrated by analysis, as allowed by the NRC. The tests were first conducted with an engineering prototype container. The results from these tests were used to develop design enhancements for the container. For example, a thin ceramic-fiber paper was added as a liner to the polyurethane foam cavity of the outer containment assembly to provide additional protection from fire. Subsequently, three full-scale certification units were tested during the period from December 1988 to April 1989. The testing was performed at Sandia National Laboratories, Albuquerque, New Mexico. Before being tested, all four full-scale TRUPACT-II containers were loaded with 7,265 lb (maximum allowable payload weight) of concrete in 14 drums.

The full-scale tests consisted of free drops from a height of 30 ft followed by free drops of 40 inches onto a 6-inch-diameter puncture bar. After undergoing multiple free drops and puncture-bar impacts, the prototype and two certification packages were suspended over a pool containing approximately 8,000 gal of jet fuel, which burned for more than 30 minutes. The external skin temperature exceeded 1475°F during the fire. Because of the excellent thermal properties of the package, the maximum O-ring seal temperature (on either the inner or the outer containment vessel) reached only 260°F, well below allowable temperatures for the seal materials used. Also, it was found that at least 5 inches of the original 10-inch-thick polyurethane foam in the outer containment assembly remained unaffected after the fire test, further demonstrating the safety margins that have been built into the TRUPACT-II shipping container.

As shown in Table L.2.1, the number of drop and puncture tests performed on each test unit exceeded the regulatory requirements in many cases; this was done to confirm that the package could sustain impacts in a variety of "worst-case" orientations and remain leaktight. For example, each of the 30-ft drops on test units 1 and 2 were performed with different sections of the TRUPACT-II container package striking the unyielding target (i.e., tiedown locations on the bottom, top knuckle of the head, etc.).

The full-scale testing of the test units under the hypothetical accident conditions was conducted with the first certification test unit at the ambient temperature of Albuquerque, New Mexico, in December 1988 (40 to 70°F). The second and third certification test units were chilled to −20°F before the first drops and again before the final leakage-rate tests to prove the ability of the O-rings to function properly at low temperatures.

The leakage rate of the containment seals was tested before, during, and after the test sequence on each test unit. On the first and the third test units, both the inner and the outer containment vessels were demonstrated to be leaktight. On the second test unit, the outer vessel met the criteria for leaktightness as stated in ANSI 14.5-1987 but the inner vessel did not meet this criteria, because debris resulting from the tests interfered with the upper seal of the inner vessel. A wiper O-ring was added to the inner

TABLE L.2.1 Regulatory testing requirements and the actual TRUPACT-II certification testing program

Test	Required number of tests[a]	Number of tests performed: Unit 1	Unit 2	Unit 3
30-ft drop	1	3	3	3
40-inch puncture drop	1	5	6	5
Fire test	1	1	1	0
Immersion	1	By analysis[b]	By analysis[b]	By analysis[b]

[a] From 10 CFR 71.73; requirements can be met by test or analysis.

[b] Same analysis was applicable to all three test units.

containment vessel on the third test unit, and its effectiveness was demonstrated by repeating the drop-test sequence. It is important to mention that had the payload been TRU waste during the testing of these three test units, no release of contents to the outside environment would have occurred because all of the test units remained leaktight to the outside.

L.2.3.2 **Fabrication Controls**

Each step in the fabrication of the TRUPACT-II containers is controlled to ensure that the containers are built to the standards and specifications of the test units used for certifying the design of the package. For example, the stainless steel that is used for the pressure vessels is traceable to the mill, including the pouring and rolling of the steel. This traceability includes test reports on the chemical and physical properties of the steel. When the steel is received at the TRUPACT Assembly Facility in Carlsbad, New Mexico, it is inspected, and each piece of steel is assigned a unique identification that stays with that piece of steel through machining, welding, and final assembly. This means that the components of any TRUPACT-II can be traced back to their origins.

Every machining operation is inspected to verify that the part is made to the drawing requirements from which it was designed. Welding during fabrication is done in accordance with the applicable standards of the American Society of Mechanical Engineers. Welds are nondestructively examined to ensure that there are no defects. Containment boundary welds are examined by x-ray. Welding procedures and welder qualifications (welders must be certified) will be available for audit or review. After welding and machining, the finished pressure vessel is proof-tested at 150 percent of its design pressure (50 lb per square inch) and then examined once again, using a liquid-dye penetrant. (A liquid-dye penetrant is used to detect cracks that cannot be seen with the naked eye.) Finally, each pressure vessel is tested to the "leaktight" criteria. The leaktightness of the containment boundary is tested on each unit before delivery. In addition to possible failures of the O-ring seals, this procedure inspects for leaks in the weld zones and cracks in the vessels.

L.2.3.3 **Operating Procedures**

L.2.3.3.1 Payload Controls and Restrictions. The initial certificate of compliance issued by the NRC (Annex 1 to this appendix) defines the allowable payload (waste materials) that can be transported. Certification of the TRUPACT-II package requires that the payload be controlled to ensure safe transportation.

Each waste container to be transported in the TRUPACT-II shipping container must comply with specific transportation requirements for physical form, the composition and radioactivity of the waste, the chemical compatibility of the waste, and the like. Unique identification codes for each waste container provide a system for tracking the process and packaging history of the waste. This information (along with process controls on waste generation procedures) provides the basis for evaluating the qualification of the waste as payload for the TRUPACT-II. The payload restrictions are described below.

Strict controls will be used at the waste generation and storage facilities to determine the compliance of a given waste package with the transportation requirements. If a package does not meet any of the limits, it cannot be a part of the payload. The Safety Analysis Report for the TRUPACT-II Shipping Package (DOE, 1989a) and supporting documents describe in detail the basis for evaluating the safety of the payload.

The Waste Acceptance Criteria Certification Committee (WACCC) has been identified to the NRC as the DOE's verification organization. The WACCC will ensure payload compliance with the TRUPACT-II certificate of compliance. To verify payload compliance, the WACCC intends to use a process similar to that used for verifying compliance with the WIPP Waste Acceptance Criteria. Therefore, each shipping facility will be required to submit a TRUPACT-II payload compliance plan and an associated quality assurance plan to the WACCC for review and approval. Detailed compliance procedures will be developed and implemented, and their implementation will be audited by the WACCC.

The individual responsible for every TRUPACT-II shipment from a given facility is the Site Certification Official. This person will ensure that the waste containers in a TRUPACT-II shipping container and the total payload are in compliance with all certification and transportation requirements. (See Appendix A for a description of the Waste Acceptance Criteria and their relationship to transportation requirements.)

Physical Form. The physical form of the TRUPACT-II payload is restricted to solid or solidified material. Examples of solid materials are paper, glass, and metals. Examples of solidified materials are cemented sludges. Liquid waste is prohibited in the payload containers except for residual amounts. Sharp objects that might affect the integrity of the payload containers are prohibited unless they are adequately packaged to prevent damage to the payload containers. Sealed containers are prohibited from being included as a part of the waste, except in volumes of 1 gal or less.

These restrictions on the physical form of the waste are met during the generation of the waste. Verification procedures like visual examination, x-ray examination, and sampling of previously packaged containers are routinely used as some of the additional controls.

Chemical Form and Chemical Properties. The following classes of materials are prohibited from the TRUPACT-II payload unless they have been destroyed, neutralized, or otherwise rendered safe:

- Compressed gases
- Explosive materials
- Nonradioactive pyrophorics
- Corrosive materials

In addition, there are restrictions on specific chemicals and materials that can be present within each waste form. These restrictions on the chemical constituents of the waste are needed in order to limit the amount of gases (flammable as well as nonflammable) that might be generated from materials in the waste on exposure to radiation.

Compliance with these requirements will be achieved through process controls at the waste generator and disposal facilities, including procurement and inventory controls. For example, in the course of being generated, waste will be subjected to neutralization and solidification to remove any corrosives that may be present in the waste. Process-flow analyses yield information on the chemical constituents of each waste form.

Chemical Compatibility. The composition of the waste must preclude adverse chemical processes during transport that might pose a threat to the payload. Specifically, it is necessary to establish the following:

1) The chemical compatibility of the waste form within each individual container of waste.

2) Chemical compatibility between waste containers under hypothetical accident conditions. In analyzing the consequences of hypothetical accidents, no credit is taken for the structural integrity of the individual waste containers. All the waste containers are assumed to be breached, and the contents from all the individual waste containers are assumed to mix together. The contents of a waste container (drum or standard waste box) must be compatible, and the contents of different waste containers in the TRUPACT-II must also be compatible.

3) Chemical compatibility of the waste forms with the inner containment vessel of the TRUPACT-II.

4) Chemical compatibility of the waste forms with the O-ring seals of the TRUPACT-II.

Each waste form to be transported in the TRUPACT-II shipping container is analyzed for the above compatibility criteria, using a method proposed by the U.S. Environmental Protection Agency (Hatayama et al., 1980). Only compatible waste forms will be part of the TRUPACT-II payload. This will ensure that chemicals that might affect the performance of the inner containment vessel or the O-ring seals are not released in any significant amounts into the inner containment vessel during transport. In addition, this will ensure that no adverse chemical reactions will take place within the waste containers or between the waste containers under accident conditions. Sampling programs conducted at the waste generating or disposal facilities provide additional verification for the chemical compatibility analyses.

Operating Pressure and Gas Generation. The acceptable maximum operating pressure in the TRUPACT-II cavity is 50 lb per square inch (gauge). The payload is limited in order not to exceed this design pressure. In addition, the generation of gas from the waste (which could occur primarily through the exposure of certain materials to radiation) is controlled to prevent the occurrence of potentially flammable concentrations of gases in the payload or the shipping containers. Gas generation is controlled by limiting the radioactivity of the waste and by restricting the constituents in the waste that may release gases on exposure to radiation.

Decay Heat and Fissile Materials. Decay-heat limits are imposed on each waste container, as well as on the total TRUPACT-II payload, to keep the potential quantity of gases generated below safe limits. In addition, the quantities of fissile materials in the waste containers and the total payload are restricted, so as to remain below the limits established by the NRC to prevent nuclear criticality under all conditions.

Waste Containers. Two types of waste containers can be shipped in the TRUPACT-II shipping containers: 55-gal drums and standard waste boxes. The latter are large steel vessels that are designed to fit in the TRUPACT-II cavity (see Appendix D). A payload consists of either 14 drums or 2 boxes. The containers must be provided with vents equipped with high-efficiency carbon composite filters that allow gases to be released from the containers while retaining particulates.

The main purpose of restrictions on the waste containers is to prevent the buildup of gases within the waste containers. Verification of compliance with these requirements includes controls on waste generation procedures, visual inspection, records and data bases, and sampling programs.

Weight. Weight limits apply to individual waste containers and to the total payload and are as follows:

Container	Weight limit (lb)
Drum	1,000
Standard waste box	4,000
TRUPACT-II shipping container	7,265

Radiation-Dose Rates. The radiation-dose rates on the external surfaces of individual waste containers and the three loaded TRUPACT-II containers to be transported on a trailer will be 200 millirem per hr or less at the surface and 10 millirem per hr or less at a distance of 2 meters from the surface, in accordance with 10 CFR 71.47.

L.2.3.3.2 Procedures for Loading and Assembling TRUPACT-II Shipping Containers. Assembling a TRUPACT-II shipment will involve three steps: 1) preparing each of the waste containers (14 drums or 2 standard waste boxes) in accordance with the specifications in the payload-control procedures (Subsection L.2.3.3.1), 2) loading the waste container into the TRUPACT-II cavity, and 3) testing the leaktightness of the seals on the outer and inner containment vessels of the TRUPACT-II shipping containers.

Specific instructions for operating the TRUPACT-II container will be given to each facility to ensure that the shipping container is loaded and sealed properly. Once the lids of the outer and the inner containment vessels are removed, the payload is lifted into the cavity of the inner vessel. Specially designed lifting devices will be provided to prevent damage to the inner vessel or the outer containment assembly during loading. Before the lid of the inner vessel is installed, the seals and other components must be visually inspected for damage that could impair their function. If function-impairing damage is present, the damaged components are replaced before further use. Once these steps are completed, the inner vessel is ready to be assembled. This is done by positioning the lid above the body and lowering it into position. The lid is then drawn downward to its fully engaged position. Once the lid is fully engaged, the lockring is rotated, thus engaging the locking lugs and locking the lid in place. Lock bolts are then installed to prevent rotation of the lockring. An assembly-verification leaktightness test is then performed to ensure that the O-ring seals were properly installed and not damaged during assembly.

This assembly procedure ensures containment integrity for the following reasons:

1) The mating surfaces between the body and head of both the inner and outer containment vessels are designed like a double tongue-and-groove joint. The head and body are connected by rotating a lockring, attached to the head, that has tabs that mate with corresponding tabs on the body. If the head and the body are not assembled correctly, it will be impossible to rotate the lockring. Ability to rotate the lockring is one verification that the head-to-body connection is properly assembled.

2) The containment boundary seal is made by an elastomer O-ring that is located at the head-to-body interface and is part of the tongue-and-groove joint. There is also a test O-ring and a wiper O-ring (on the inner vessel only). When properly

assembled, the O-rings are captured between the head and body. Each time the head is installed on the body, it is necessary to perform a leak test to verify that the O-rings are in place and that they were not damaged during assembly.

Once the lid of the inner containment vessel is properly installed, the outer vessel can be assembled. This is done in the sequence used for the inner vessel, the only difference being that the lockring is rotated and held in position by means of a mechanical actuator ring. In the locked position, lock bolts hold the actuator ring in position, which, in turn, holds the lockring in position. As in the case of the inner vessel, an assembly-verification leaktightness test is required.

L.2.3.3.3 TRUPACT-II Transport Trailer. The TRUPACT-II transport trailer is of a gooseneck, dropped bed design which is commonly used in commercial fleet operations. The design has been adapted for the transportation of up to three fully loaded TRUPACT-II shipping packages. The TRUPACT-II transport trailer is 42.2 ft in length, the load bearing bed is 40 inches aboveground and when loaded with TRUPACT-IIs, the overall height is 161.5 inches.

Each trailer is provided with 12 each, special tiedown devices used for securing the TRUPACT-II packagings in a vertical position to the trailer. The tiedowns are cam operated, adjustable length U-bolts that interface with, and clamp down on corresponding brackets on the TRUPACT-II packaging. The tiedown restraint applied to the TRUPACT-II packages has been designed to satisfy the tiedown requirements of the DOT, 49 CFR 393.102, and the NRC requirement, 10 CFR 71.45. The Safety Analysis Report for the TRUPACT-II Shipping Package given to the NRC in March 1989 provides the necessary analyses for showing how the TRUPACT-II tiedown system meets these requirements. The trailer has been through a series of tests which demonstrated it can be safely used without restrictions on the nation's highways.

L.2.3.4 **Maintenance**

A detailed maintenance program has been established by the DOE and approved by the NRC for the TRUPACT-II containers. Maintenance procedures include scheduled inspections and replacement of components, structural and pressure tests, and leaktightness tests for maintenance verification (O-ring seals, vent-port plug seals, etc.). The maintenance procedures are described briefly below.

Structural and Pressure Tests. A structural pressure test must be performed on the inner and the outer containment vessel once every 5 years. This involves pressure testing each vessel to 150 percent of the maximum normal operating pressure.

Leaktightness Tests. Maintenance-verification leaktightness tests must be performed for the main O-ring seals and for each vent-port plug seal annually or on seal replacement.

Maintenance of Components. Maintenance is specified for certain components, such as fasteners, lockrings, and seal areas and grooves. The threaded parts of fasteners are to be annually inspected for deformed or stripped threads. Visual inspections are required before every use for the lockring bolts (inner containment vessel and outer containment assembly), the vent-port plugs, and the seal-test port. Any damaged parts must be replaced before further use. The lockring of the inner vessel and the locking actuator of the outer containment assembly are to be inspected before every use for any motion-impairing components. Corrective actions are to be taken whenever necessary. Before each use, and at the time of seal replacement, sealing surfaces and O-ring seal grooves are to be visually inspected for any damage. An annual inspection of the dimensions and surface finishes of the O-ring seal area is also required. The required measurements include groove widths, tab widths, axial play, and the surface finish of sealing areas.

Maintenance, repairs performed, or components replaced will be documented on the TRUPACT-II Maintenance Record Form WP-1709 (DOE/WIPP 88-026). All records of maintenance activities performed on the TRUPACT-II container will be maintained by WIPP Operations for retention and distribution. The records will be designated as quality assurance records and will be maintained as permanent records. All replacement components procured by user facilities will be verified for compliance with applicable material requirements. The DOE shipping and receiving facilities that perform maintenance on TRUPACT-II containers will have in place a quality assurance program that meets the applicable requirements of the DOE (see Section L.4).

L.3 **THE NUPAC 72B CASK PROGRAM**

L.3.1 **BACKGROUND**

To transport remotely handled (RH) TRU waste, the DOE will use the NuPac 72B shipping cask. The NuPac 72B cask is being designed to meet NRC requirements for Type B packages, and the DOE will apply to the NRC for a certificate of compliance before transporting any waste in the 72B cask. The 72B cask is a scaled-down version of the NuPac 125B cask, whose design has been certified by the NRC as a Type B packaging. The 125B cask is being used to transport debris from the core of the damaged Three Mile Island reactor.

L.3.2 **DESCRIPTION OF THE NUPAC 72B SHIPPING CASK**

The NuPac 72B cask is a cylindrical cask consisting of a separate inner vessel within an outer cask protected by impact limiters at each end. A schematic is shown in Figure L.3.1. The outer cask provides the primary containment boundary for the payload, while the inner vessel provides a secondary containment boundary. Neither containment vessel (the outer cask nor the inner vessel) is vented, and each is capable of withstanding an internal pressure of 150 lb per square inch (gauge). The capacity of each cask is 8,000 lb of payload. The payload consists of RH TRU waste in 30- or 55-gal drums contained in a canister. The 72B cask is designed to transport a single canister per shipment. A single 72B cask will be loaded onto a custom-designed semitrailer pulled by a conventional tractor.

The inner containment vessel is made of stainless steel and provides a cavity for the payload canister that is approximately 26.5 inches in diameter and 123 inches long. The lid is secured to the body of the vessel by means of eight closure bolts. Internal spacers are provided at the top, bottom, and at two locations near the middle of the inner vessel to center the canister and facilitate the insertion and removal of the canister.

The outer cask is a stainless-steel vessel constructed of two concentric shells enclosing a cast-lead shield. The shield is for gamma radiation and is approximately 1.9 inches thick. The outer cask is approximately 142 inches long and has an outer diameter of 42 inches. It is protected at each end by energy-absorbing impact limiters, which are stainless-steel shells filled with polyurethane foam. The impact limiters also act as thermal insulators to protect seal areas from fire during an accident.

The payload canister, or RH waste canister, is a DOT 7A Type A carbon steel single shell container measuring approximately 26 inches in diameter with an overall length of 121 inches. The canister is vented using a carbon composite HEPA filter and is capable of transporting three 55-gallon waste drums. The allowable gross weight of the canister and contents is 8,000 pounds.

FIGURE L.3.1
CROSS SECTION OF THE NUPAC 72B CASK

L.3.3 **COMPLIANCE WITH NRC REQUIREMENTS**

In order for the design of the NuPac 72B cask to be certified by the NRC, it will be necessary to demonstrate compliance with the NRC requirements in 10 CFR Part 71 for Type B packages (see Subsection L.2.2). Compliance with these requirements may be demonstrated by analysis or by a combination of analysis and testing. Since the 72B cask is a scaled-down version of the 125B cask, whose design has been certified by the NRC, analysis will be the primary method of demonstrating compliance with the NRC regulations.

L.3.4 **OPERATING PROCEDURES**

L.3.4.1 **Payload Controls and Restrictions**

As in the case of the TRUPACT-II shipping container, the NRC's certificate of compliance for the 72B cask will specify the allowable payload. The restrictions on the payload will be similar to those discussed in Subsection L.2.3.3.1 for CH TRU waste.

Physical and Chemical Form. The restrictions on the physical and chemical form of the payload to be carried by the 72B cask and the necessary payload controls are expected to be similar to those specified for the CH TRU waste in the TRUPACT-II payload. These restrictions are described in Subsection L.2.3.3.1 of this appendix.

Chemical Compatibility. The payload for the 72B cask will be evaluated to ensure chemical compatibility within itself and with the cask. The criteria for evaluating and ensuring chemical compatibility are discussed in Subsection L.2.3.3.1.

Operating Pressure and Gas Generation. The pressure in both containment levels of the cask is 150 lb per square inch (gauge). The payload is restricted in order to not exceed this design pressure. The generation of gas from the waste is controlled to prevent the occurrence of potentially flammable concentrations of gases.

Weight. The maximum weight of the loaded canister in the 72B cask is limited to 8,000 lb. The cask may carry no more than one canister of RH TRU waste.

Decay Heat. The thermal design rating of the package is 300 watts internal decay heat maximum.

Radiation-Dose Rates. The radiation-dose rates on the external surface of the 72B cask will be below the levels specified in 10 CFR 71.47 and must comply with 49 CFR 173.441.

L.3.4.2 **Procedures for Loading the NuPac 72B Cask**

Loading a 72B cask for transport will consist of the following steps: 1) determining that the payload (the canisters of RH TRU waste) has been verified to meet the payload restrictions specified in the certificate of compliance, 2) loading the prepared payload canister into the 72B cask, 3) testing the leaktightness of the seals on the containment vessels of the cask, and 4) securing external impact limiters on the cask.

Specific procedures for operating the 72B cask will be provided to each waste generating or storage facility to ensure that the cask is loaded and sealed properly. The loading procedures include removing the lids from the containment vessels, loading the waste canister into the vessel, installing the lids, and performing the leaktightness tests.

L.3.5 **MAINTENANCE OF THE NUPAC 72B CASK**

As in the case of the TRUPACT-II shipping container, a strict maintenance program will be developed and implemented for the 72B cask. The procedures will be submitted to the NRC as part of the Safety Analysis Report (SAR). The NRC must approve these procedures before the design of the NuPac 72B cask is certified and the cask can be used to transport waste.

The maintenance program will include periodic inspections and replacement of components, structural and pressure tests, leaktightness tests, and routine maintenance of all necessary parts of the cask. A comprehensive quality assurance program will also be developed, as discussed in Section L.4.

L.4 **QUALITY ASSURANCE PROGRAM**

The NRC regulations in 10 CFR Part 71 include requirements for implementing a quality assurance program that is used in the design, purchase, fabrication, handling, shipping, storing, cleaning, assembly, inspection, testing, operation, maintenance, repair, and modification of those components of the TRUPACT-II container and NuPac 72B cask that are important to safety. The quality assurance requirements are not optional; they are mandatory.

The quality assurance program provides a systematic approach to ensuring that a design, and the resulting product or service, are safe and satisfactory for the intended use. The program is aimed at preventing problems, not only at detecting and solving them.

The quality assurance program is developed and implemented by specially trained full-time employees. They report to the highest level of management in their organizations in order to maintain their independence from concerns about costs or schedules. Their primary function is to make sure that the quality assurance program meets the requirements of the NRC and is effective in producing a product that meets required standards and that will maintain its integrity during operation. This requires ascertaining that all workers are trained and qualified to perform their assigned tasks, all workers are trained to understand the program, and work is properly controlled.

Design Control. Quality assurance begins with the design of an item or the description of a service. Large safety margins are established for each item (i.e., if a TRUPACT-II will be operating at a pressure of 50 lb per square inch, it is designed to be strong enough for a pressure of 75 lb per square inch). All of the mathematical calculations and analyses used in making design decisions are reviewed and verified by independent qualified personnel.

Procurement Control. Quality assurance requires that the materials used in constructing a shipping container or cask be tested, both chemically and physically, to make sure that they have the properties needed for the TRUPACT-II or NuPac 72B design. Further, the suppliers who manufacture the materials are evaluated to ensure that they have an acceptable program for ensuring that the materials they are furnishing are properly analyzed, chemically and physically, and that the analysis reports match the material shipped.

Marking and Control of Materials. Once the material arrives, it is inspected by a quality inspector and stored properly for use. The material is placed in an environment that will not damage it and marked or tagged so that its identity is not lost. The materials used in the TRUPACT-II container or the NuPac 72B cask must be traceable from the production unit in which it is used, back to the purchase order used to buy it and the material test report verifying that the material is suitable. Thus, if a problem arises in a particular batch of material, the company must identify every production unit in which the material was used.

Instructions, Procedures, and Inspection. Work on the TRUPACT-II or the NuPac 72B units is performed in accordance with formal instructions, procedures, or drawings that have been reviewed by engineering and quality assurance personnel. Part of this formal system for controlling the work includes setting points during the fabrication for inspection. If one of these predetermined points is ignored and the inspection cannot be performed at a later time, the unit faces rework. These inspection points are a part of every work plan and ensure that the final unit is acceptable. Those same similar instructions, procedures, and drawings are later used to perform preventive maintenance during the operation of the TRUPACT-II container or the NuPac 72B cask.

Control of Processes. Some types of processes require more control than others because special techniques like x-ray examination are needed to determine that they were performed properly. An example of such a process is welding. The quality assurance program makes special provisions for such processes and for ensuring that the special inspection techniques required for these processes are used successfully. These special provisions include testing the skills of the personnel performing the processes, qualifying the procedure being used, and verifying that the materials and equipment for the process are appropriate. In addition, quality assurance personnel perform in-process inspections to make sure that the controls are being used during the actual work. Records of these activities are kept.

Test Control. Any type of testing requires very tight control and careful monitoring by quality assurance personnel. For example, pressure and leaktightness tests on the containment vessels of the TRUPACT-II container are performed in accordance with formal procedures that have been reviewed by both engineering and quality assurance personnel. Tests are witnessed by quality assurance personnel, and test results are formally documented and reviewed for adequacy. Any reworking on the containment boundary of a TRUPACT-II unit requires previous tests to be performed again.

Control of Measuring and Test Equipment. Results from inspections and tests are only as good as the equipment used to measure the results. The quality assurance program requires that the equipment used to measure or test a TRUPACT-II shipping container be calibrated. This means that all measuring and test equipment has to be checked against a national standard for the particular measurement being taken and has to be accurate within a given range. Not only does the equipment have to be checked and adjusted if necessary, it also has to be rechecked periodically. If a piece of equipment is found not to agree with the national standard, the manufacturer has to evaluate each item that was inspected or tested with that piece of equipment.

Acceptability of Components. The acceptability of parts of a TRUPACT-II container or a NuPac 72B cask must be apparent at all stages of fabrication. The quality assurance program provides a method of doing this by using inspection hold points, tagging, etc. If an item is found to be unacceptable, the quality assurance personnel document the problem on what is called a nonconformance report. The item is then marked or tagged and segregated from the rest of production until a decision is reached on what to do with the item. This decision is made by engineering and quality assurance personnel. Sometimes an item can be reworked and made acceptable; sometimes an item must be scrapped. The provisions of the quality assurance program, however, prevent unacceptable items from being unintentionally used in the production process and provide a method for deciding how to handle unacceptable items.

Surveillance. In addition to inspections, quality assurance personnel perform scheduled and unscheduled surveillance of various activities to make sure that employees are operating to the same rules and are performing their jobs well. The activities selected for surveillance are those in progress that are most important to the operation at the time.

Corrective Action. The quality assurance program specifies a method for identifying recurring problems and serious problems that might affect the performance of the product. A formal report, called a "corrective action report," is issued by quality assurance personnel when such problems surface. This report must be answered by production or engineering personnel and must include an explanation of what is causing the problem, a description of what is being done to correct the problem, and a description of what is being done to keep it from happening again. Quality assurance then makes sure that the proper actions have been completed and that they are, in fact, solving the problem. These reports are reviewed by the highest level of management, who make sure that all departments respond quickly.

Document Control. The different parts of the quality assurance program are formally documented to make sure that personnel understand the rules and controls that are necessary to produce a good product. These documents are themselves controlled to make sure that all personnel are working to the same guidelines and that only the latest documents are in use. If a document is changed, the old document must be returned or destroyed and personnel must be trained to ensure that they understand the new rules. This is true of every document that affects work, including work plans, procedures and drawings, and inspection plans.

Quality Assurance Records. The final step before releasing a TRUPACT-II or NuPac 72B unit for use is the review of related quality records. These records tell the production story of a unit. They start with the pedigree of the materials used and proceed through fabrication, inspection, and testing to final acceptance. This final review by quality assurance ensures that the records are complete, inspections have been performed, and the requirements have been met. This same record package, which is several hundred pages, is then retained in duplicate in protected storage for the life of the TRUPACT-II or NuPac 72B unit.

Audits. An important mechanism for ascertaining that the quality assurance program is correctly implemented is the audit. Quality assurance personnel audit their facility and operations to see whether all the established rules and regulations are complied with. If deficiencies are found, they are documented, corrected, and verified as effective. The quality assurance personnel who perform these audits are specially qualified through classroom and on-the-job training to spot problems in the system and get them fixed. Auditors from outside the organization also perform this function. For example, the Westinghouse Electric Corporation (the operating contractor for the WIPP) audits Nuclear Packaging (the manufacturer of the TRUPACT-II container), and the DOE audits Westinghouse, as well as the waste generator and storage facilities. The NRC has also audited Nuclear Packaging as part of the certification process for the TRUPACT-II design and has the prerogative to audit any activities associated with the use of a TRUPACT-II container.

Summary. As overlapping as all of the described quality assurance controls may seem, the checks and balances built into the program are necessary to provide the highest assurance possible that the TRUPACT-II container and the NuPac 72B cask will safely perform its intended function. This program will remain in effect as long as TRUPACT-II or NuPac 72B units are being used.

REFERENCES FOR APPENDIX L

DOE (U.S. Department of Energy), 1989a. Safety Analysis Report for the TRUPACT-II Shipping Package, Rev. 2, Docket No. 71-9218, prepared for the U.S. Department of Energy by Nuclear Packaging, Inc., Federal Way, Washington, D.C.

DOE (U.S. Department of Energy), 1989b. TRUPACT-II Operation and Maintenance Manual, Rev. 0, OM-0134-NP, prepared by Nuclear Packaging, Inc., Federal Way, Washington, D.C.

DOE (U.S. Department of Energy), 1989c. Quality Assurance Plan for the Transportation and Receipt of Transuranic (TRU) Waste, draft, DOE/WIPP-89-012, Rev. 0, Albuquerque, New Mexico.

DOE (U.S. Department of Energy), 1989d. TRUPACT-II Content Codes (TRUCON), DOE/WIPP-89-004, Rev. 1, Albuquerque, New Mexico.

Hatayama et al. (H. K. Hatayama, J. J. Chen, E. R. deVera, R. D. Stephens, and D. L. Storm), 1980. A Method for Determining the Compatibility of Hazardous Wastes, EPA-600/2-80-076, prepared by the California Department of Health Services, Berkeley, California, for the Municipal Environmental Research Laboratory, Cincinnati, Ohio.

IAEA (International Atomic Energy Agency), 1987. Explanatory Material for the IAEA Regulations for the Safe Transport of Radioactive Material, second edition, Safety Series No. 7, Vienna, Austria.

Jefferson, R. M., 1983. Transporting Spent Reactor Fuel: Allegations and Responses, SAND82-2778, TTC-0403, Sandia National Laboratories, Albuquerque, New Mexico.

McClure et al. (J. D. McClure, R. E. Luna, and F. P. Falci, Jr.), 1987. Public Concerns About RAM Transport--Communicating Engineering Data on Risk, SAND87-2656A, TTC-0762, Sandia National Laboratories, Albuquerque, New Mexico.

NRC (U.S. Nuclear Regulatory Commission), 1975. Regulatory Guide 7.4.: Leakage Tests on Packages for Shipment of Radioactive Materials, Washington, D.C.

ANNEX 1

NRC CERTIFICATE OF COMPLIANCE
FOR THE TRUPACT-II SHIPPING CONTAINER

APPENDIX M

SUMMARY OF THE MANAGEMENT PLAN FOR THE TRUCKING CONTRACTOR

TABLE OF CONTENTS

LIST OF FIGURES

LIST OF TABLES

M.1 **INTRODUCTION**

This appendix has been prepared in response to comments on the draft SEIS. Representative comments include concerns about the trucking contractor's experience and safety programs, drivers' rights and training, tractor-trailer requirements, and general safety issues. This appendix addresses these concerns by describing the provisions that will be made and the procedures that will be followed to ensure that the transportation of waste to the WIPP is conducted safely.

This appendix summarizes the management plan developed by the contractor selected by the U.S. Department of Energy (DOE) for transporting transuranic (TRU) waste to the WIPP. The selected contractor is the Dawn Trucking Company of Farmington, New Mexico. The transportation operations will be conducted by truck, using a fleet of tractors provided by the contractor and trailers and shipping containers provided by the DOE. The contractor will conduct the transportation operations from a facility to be developed at Hobbs, New Mexico. The transportation project will be both managed and coordinated from the Hobbs facility, but management and support personnel at the contractor's offices in Farmington will be available to assist if needed.

As described in this appendix, the trucking contractor has developed detailed procedures related to safety, equipment maintenance, quality assurance, driver qualification and training, the duties and responsibilities of drivers, dispatching, the reporting of incidents and accidents, and communications procedures associated with shipment tracking. Many of these procedures are based on the regulations issued by the Department of Transportation (DOT) for the transport of hazardous materials, RCRA (40 CFR Part 263) requirements for the transport of mixed waste, and on the experience of the Federal Government in transporting radioactive materials for several decades, particularly the experience of the DOE in transporting weapons.

In reviewing the WIPP program activities, the National Academy of Sciences (NAS) concluded that the "system proposed for transportation of TRU waste to the WIPP is safer than that employed for any other hazardous material in the United States today and will reduce risk to very low levels."

The DOE and the trucking contractor have tried in this plan to reduce as much as possible the potential for human error or mechanical failure. Extensive driver-training requirements, dry-run readiness experience (see Appendix D.2.3.2), emphasis on safety, inspections that exceed many DOT regulatory requirements, and the use of tractor-trailers equipped with governors that limit speed are a few examples of ways in which transportation risk has been minimized. In addition, this plan will be evaluated for improvements at least annually.

M.2 **SAFETY**

M.2.1 **POLICY**

Safety is of primary importance in planning and conducting all activities related to the transportation of the TRU waste. The objective is to protect the safety of the public and to protect the employees of the trucking contractor from occupational injuries and illnesses. In order to achieve this objective, the trucking contractor will rely on a variety of mechanisms and measures, including the following:

- Compliance with all applicable health and safety requirements of the Federal Government, States, and local jurisdictions
- Provision of vehicles and equipment with the best available mechanical safeguards, including governors that limit speed, and personal protective equipment
- Provision of a facility for equipment maintenance and inspection
- Implementation of a safety program, including personnel training in safe work practices
- Stringent driver training program and penalty provisions
- Accident and emergency training
- Provision of a constant-surveillance service for all loaded shipments
- Provision of communications equipment and services.

M.2.2 **REQUIREMENTS FOR PROTECTING HEALTH AND SAFETY**

All activities related to the transportation of TRU waste will be conducted in accordance with the applicable health and safety requirements of the Federal Government, States, and local jurisdictions, including the requirements promulgated by the U.S. Department of Transportation in Title 49 of the Code of Federal Regulations (49 CFR).

The maintenance facility (see Section M.4) will meet all applicable requirements of the U.S. Occupational Safety and Health Administration and the State of New Mexico. All trucks and drivers will meet the applicable requirements of the U.S. Department of Transportation. To ensure that these requirements are met, the trucking contractor will implement a maintenance and inspection program that will be regularly and continually monitored by contractor and DOE management. Another mechanism for ensuring regulatory compliance will be a safety program, which is discussed in the next subsection.

When waste shipments are under way, all applicable regulations pertaining to the shipment of hazardous waste will be followed.

As described in Section M.6, constant-surveillance service will be provided for all loaded shipments. In addition, a satellite-based tracking system will be used to determine the location and progress of all shipments. Such a tracking system is not a Federal requirement but is a voluntary DOE program decision for WIPP shipments.

M.2.3 **SAFETY PROGRAM**

The transportation contractor will establish and maintain a safety program that will consist of both a safety orientation for new employees and a continuing education program for all employees. To ensure that the safety program is successful, each employee will be made aware of his or her responsibilities in the program. All employees will be required, as a condition of employment, to observe established safety regulations and practices and to use the safety equipment provided.

Every new employee will receive safety instructions, a personnel safety handbook, and any protective equipment deemed necessary. The orientation program for new employees will consist of verbal and written information on job safety, accident-prevention measures, and the responsibilities of the new employee in the safety program. In addition, each driver will be given special training as described in Section M.5.

The continuing education program will include training in applicable safety requirements and regulations, the use of equipment, and safe operating procedures. In addition, safety meetings will be held each week to train and inform employees. All employees will be required to participate in these meetings and to sign an attendance list. The immediate supervisor will be responsible for conducting the meeting. A brief report on the subjects to be discussed will be prepared for each meeting.

M.2.4 **OCCUPATIONAL SAFETY**

As a matter of policy, no employees will work in surroundings that are unsanitary, hazardous, or dangerous to their health or safety. All employees will be required to maintain their project or work areas. Adequate medical and first aid supplies will be available at all work locations.

When needed, the employer will furnish tools, vehicles, and equipment with the best available mechanical safeguards and personal protective equipment. Employees using tools, vehicles, and equipment will be responsible for inspecting them before use to determine that they are in a safe, operable condition.

Each member of the management team will be responsible for not only protecting the safety and health of all employees who report to or are assigned to him or her but also for the safe work conduct of those employees.

M.3 **EQUIPMENT**

The tractors used for hauling TRU waste to the WIPP will be provided by the trucking contractor. The trailers and the shipping containers (TRUPACTs) will be provided by the DOE. It is estimated that the tractor fleet will consist of 10 units domiciled in Hobbs, New Mexico. All vehicles will be 1989 and later models, and will be replaced as needed throughout the program.

All equipment used by the trucking contractor to transport TRU waste will conform to applicable Federal regulations (e.g., the requirements for placarding in 49 CFR Part 172); will meet the needs of the DOE; will meet all functional requirements for TRU waste shipments, such as being equipped with special tiedowns for the TRUPACT-II containers; and will have special equipment related to safety. For example, to prevent speed limits from being exceeded, the vehicles will be equipped with governors that will limit the speed to 65 miles per hour. In addition, the tractors will have a Tripmaster, which will automatically record all the speeds the vehicle reached in traveling. The tractors will also be equipped with radiation detection instruments for use by drivers who will be properly trained in their use, in the event of an accident.

The specifications for the tractors are given in Table M.3.1. These specifications are based in part on the DOE's experience over the last 12 years in the transport of nuclear materials.

The dimensions and weights of the tractors and trailers are given in Table M.3.2. These dimensions and weights are in compliance with applicable Federal and State safety requirements.

Table M.3.1 Specifications for the tractors to be used in hauling TRU waste to the WIPP

Make and model:	FLD-12064ST Freightliner
Wheel-base length:	219 inches
Weight (dry):	15,915 pounds
Engine:	NCT 444 Cummins B/C4 @2100 rpm
Power steering:	Ross TAS-65 by TRW, Inc.
Brakes	
steering axle:	15 x 2 CAM centrifuge drums
driving axles:	16-1/2 x 7 CAM centrifuge drums
emergency brakes:	MGM dual brakes
Engine brake:	Cummins Brake Retarder
Transmission:	Road Ranger 18-speed transmission
Axles	
steering axle:	12000# FF 921
driving axles:	3800# SQ 100 A
Tires	
steering:	Michelin PXZA-1
driving:	Michelin XDHT
Tire chains:	Laclede
Fenders	
steering wheels:	Molded fenders
rear wheels:	Aluminum full fenders
Fifth wheel:	18-inch Holland FW-2535
Air-ride suspension:	Freightliner air-ride suspension, 40,000 pounds
Mobile telephone:	Motorola Dynatac 6000x
Citizens band radio:	40-channel COBRA 29+
Other specifications:	Front leaf springs, 64 inch Aluminum wheels, frame, and fuel tanks Radiation detection meters alpha-beta-gamma meter beta-gamma meter Rockwell tripmaster Heated rear-view mirrors Heated and air-conditioned cab and sleeper Spray guards and mud flaps for the rear and front wheels Locking fuel caps Externally mounted fire extinguisher Tamper-proof fifth wheel locking device

Table M.3.2 Overall dimensions of the tractor-trailer unit

Length

Tractor, total length: 26 feet 6 inches
Trailer, total length: 42 feet 2 inches
Total length: 62 feet 10 inches (with overlap of 5 feet 10 inches)

Width

Trailer: 8 feet 6 inches
Tractor: 8 feet 11 inches (includes side mirrors)

Height

Tractor: 12 feet
Trailer with load (maximum): 13 feet 5 inches

Weight

Tractor	Weight (pounds)
Weight dry	15,915
Fuel	1,100
Tire chains	91
Drivers and equipment	500
Spare tire	190
Tractor weight	17,796
Trailer (includes tools and spare tire)	8,500
Three loaded TRUPACT-II containers (maximum allowable)	53,299
(Maximum loaded shipping weight of any single TRUPACT-II is 19,250 lbs)	
Total weight	79,595

M.4 **EQUIPMENT MAINTENANCE**

M.4.1 **MAINTENANCE FACILITY**

A facility for the maintenance, storage, and dispatching of tractors and trailers will be provided when required by the DOE. Until such time as a facility is required by the DOE, the tractors and trailers will be stored at the WIPP site. The proposed maintenance facility, to be located at a 6-acre site in Hobbs, New Mexico, will be designed to provide most of the facilities needed for fleet maintenance and operation as a truck terminal. It will contain a three-bay maintenance shop with an area of 6,500 square feet and an office building with an area of 1,550 square feet. If the proposed site is unavailable when the WIPP opens, an equivalent facility will be used.

M.4.2 **MAINTENANCE PERSONNEL AND EQUIPMENT**

Initially, the maintenance facility will be staffed by one mechanic, a shop helper, and security guards (see Subsection M.4.6). A second mechanic will be added when needed.

All mechanics will have a minimum of 5 years of qualified experience related to diesel engines, air pressure, brake systems, electrical systems, and arc and gas welding. Certification of training in a 2-year technical school specializing in diesels and heavy equipment will be required. The mechanics will receive special training from the manufacturers of the tractors.

The equipment and tools to be provided in the maintenance facility include the following:

- Overhead crane
- Grease pit
- Two 20-ton jacks
- Transmission floor jack
- Jack stands
- Engine stands
- Cutting torch
- Welder
- Drill press
- Hydraulic press
- Battery charger
- Air compressor with hoses

M.4.3 **MAINTENANCE SCHEDULE**

The schedule to be used for the maintenance of tractors and trailers is given in Table M.4.1. If the manufacturers recommend more frequent maintenance, the manufacturers' recommendations will be followed. Miscellaneous maintenance to repair broken wheels, flat tires, air fittings, air lines, and other similar items will be performed as required.

All in-use tractors and trailers will be inspected monthly, with the inspection recorded on special forms. These forms, which are shown in Figures M.4.1 and M.4.2, specify the items to be inspected. In addition, the trailers will be inspected semiannually and annually (or after driving 10,000 or 20,000 miles, whichever comes first); these inspections will be recorded on the form shown in Figure M.4.3. Furthermore, as described in Section M.6, the tractors and trailers will be inspected by the drivers before each trip, every 2 hours or 100 miles during the trip, and after the trip.

Table M.4.1 Maintenance schedule for tractors and trailers to be used to transport TRU waste to the WIPP

Grease every 5000 miles.

Oil and filter change every 15,000 miles or as specified by manufacturer[a].

New brakes and wheel seals every 100,000 miles or when needed, whichever is first.

New tires every 100,000 miles or when needed, whichever is first.

Miscellaneous maintenance to include universal joints, broken wheels, flats, air fittings, air lines, etc., as required.

[a] For tractors only.

If it is necessary to test welds by a nondestructive examination method, arrangements will be made with a subcontractor. If difficulty in scheduling this procedure is encountered, the weld testing will be performed as directed by the DOE.

For the trailers, which will be furnished by the DOE, no maintenance beyond that considered routine or preventative will be permitted. Also prohibited for the trailers will be any modifications, cutting, welding, or drilling, unless authorized by the DOE.

FIGURE M.4.1
EXAMPLE OF MONTHLY TRACTOR INSPECTION FORM

FIGURE M.4.2
EXAMPLE OF MONTHLY TRAILER INSPECTION FORM

FIGURE M.4.3
EXAMPLE OF ANNUAL AND SEMIANNUAL TRAILER INSPECTION FORM

FIGURE M.4.3
(CONCLUDED)

M.4.4 **QUALITY ASSURANCE AND CONTROL**

The trucking contractor will implement a quality assurance (QA) program that meets the QA requirements of the DOE. Procedures for the QA program will be developed, and personnel will be trained in their implementation. In addition, quality control procedures will be implemented.

The trucking contractor will be responsible for ensuring the accuracy and reliability of measurements, tests, and maintenance procedures performed at the maintenance facility through the use of inspection, measuring, and test equipment of the range, accuracy, and type necessary to determine conformance with established requirements. To the extent required by established procedures, test equipment, gauges, and tooling will be calibrated by an approved standards laboratory. Items requiring calibration will carry readily visible labels showing their calibration status and will be recalibrated as necessary. Items with an expired calibration date will be segregated to ensure that they will not be used for maintenance or inspection.

All replacement parts must conform to manufacturer's specifications for replacement parts and warranted by the maker. The supplier of parts will be required to provide a copy of the warranty at the time a part is delivered for the first time. For subsequent deliveries, the supplier will be required to submit a statement that the part conforms to the original warranty. The warranty and the subsequent quality assurance statement will be kept on file at the maintenance facility. Before it is placed in inventory or installed, each part will be inspected by the mechanic. The mechanic will be responsible for ensuring that all parts received conform to the warranty requirements. The packing slip or other document that accompanies the part will be stamped "Accepted by" and initialed by the mechanic and given to the dispatcher for review.

Material or equipment that does not meet established requirements will be withheld from use until it has been appropriately repaired or reworked. All nonconforming items will be segregated and properly tagged to ensure that they will not be used.

All providers of services will be required to supply documentation that the service meets accepted or required standards applicable to the service being rendered. They will be given a notice of requirements and will be required to certify that their work will be, and has been, conducted according to required standards by qualified personnel. Before authorizing any work, the trucking contractor will verify that the service provider can meet all requirements.

The trucking contractor will verify compliance of the QA program by conducting audits at least every 6 months. The audited organization will verify and document the actions taken to satisfy any recommendations made by the auditors. The results of the audits will be documented and a copy sent to the DOE Transportation Representative.

The QA program will include the requirement that records furnishing evidence of quality assurance be prepared and maintained; examples of such records are reports on audits, inspections, maintenance, and training. The detailed requirements for the control of the QA records will be included in the trucking contractor's QA procedures.

At a minimum, these procedures will address legibility, retention, distribution, maintenance, transmittal to the WIPP, and protection against damage or loss.

At least once a month, the maintenance records and the certification of parts and services provided by other firms will be reviewed by the dispatcher to determine that all standards are being met. If the dispatcher finds that a part or service was not properly certified, the use of that part or service will cease immediately. The provider of the part or service will be notified in writing and required to furnish certification. If certification is not immediately furnished, the provider will be removed from the list of acceptable providers.

If noncertified parts have been installed, the dispatcher will order an immediate inspection of the part to determine whether the part is adequate. If adequacy cannot be ascertained, the part will be replaced. In the event of a noncertified service, the dispatcher will order an immediate review, and the service will be repeated if necessary.

The dispatcher will conduct random inspection to verify the adequacy of repairs performed by employees and by providers.

M.4.5 **RECORDS**

In addition to the QA records discussed above, a record file will be maintained for the inspection sheets and shop tickets for each tractor and trailer. Parts-inventory cost sheets will be attached to each shop ticket (see Figure M.4.4).

All records will be prepared in triplicate. One sheet will be placed in the file mentioned above, one sheet will be forwarded to the contractor's home office, and one sheet will be filed at an off-site location.

M.4.6 **SECURITY**

Security for the maintenance facility will be provided by the following physical features and by personnel procedures. The site will be surrounded by a 6-foot-high chain-link fence with barbed wire at the top. Access will be allowed only for authorized personnel, who will be admitted through a single gate controlled by personnel inside the facility. Floodlights will be used to illuminate the shop, office, fueling, and truck storage area. The site will be occupied at all times (24 hours a day, 365 days a year) by maintenance or dispatching personnel or by a security guard.

All deliveries will be accepted at the gate. If a maintenance service is to be provided by a subcontractor, the service provider will be accompanied by an authorized employee of the maintenance facility. No unauthorized access by the public will be allowed at any time.

FIGURE M.4.4
EXAMPLE OF DAWN TRUCKING SHOP TICKET

M.5 **DRIVERS**

It is estimated that 30 drivers will be needed for the trucking program, and the trucking contractor will ensure that only qualified drivers are hired. The contractor, who is an equal opportunity employer, will locate qualified drivers by posting job openings in Job Service centers in all communities near the WIPP site, including Hobbs, Carlsbad, and Roswell, as well as major cities in New Mexico and western Texas. In addition, the contractor may place advertisements in trucking publications. Drivers will be selected on the basis of ability and experience.

M.5.1 DRIVER QUALIFICATIONS

To qualify initially, applicants will have to meet the following requirements: they must be citizens of the United States and at least 25 years of age; they must have logged at least 100,000 miles in driving semi-tractor trailers, must have at least 2 years of uninterrupted experience in driving commercial semi-tractor trailers during the last 5 years, and may not have any moving violations (including chargeable accidents) in the past 3 years.

The driver-qualifying process will consist of the following:

- Completing an application for employment
- Initial interview
- Verification of employment -- including years of service and mileage logged
- Check of driving record, including possession of a Commercial Driver's License
- A test, given by qualified personnel, that examines performance in the following:
    - -- Pretrip inspection
    - -- Coupling and uncoupling of tractor and trailer
    - -- Placing tractor in operation
    - -- Use of tractor controls and emergency equipment
    - -- Operating the tractor in traffic and while passing other vehicles
    - -- Turning the tractor
    - -- Braking and slowing the tractor by means other than braking (shifting gears)
    - -- Backing and parking the tractor
- Drug screening
- Physical examination
- Written test on Federal motor-carrier safety regulations and hazardous materials regulations in accordance with 49 CFR 391.35
- Driver-profile evaluation.

When a driver has successfully completed this qualification process, a written report on the driver will be sent to the DOE for approval (see Figure M.5.1). If approved, the driver will be trained as described in the next subsection.

M.5.2 **DRIVER TRAINING PROGRAM**

Every driver hired by the trucking contractor will have to complete a training program in accordance with the requirements of 49 CFR 177.825. In addition, every driver will receive training to meet the requirements of 49 CFR Part 397. The training to meet the requirements of 49 CFR will be conducted by the Colorado Safety Institute in Denver. However, if necessary to meet scheduling requirements, an alternative qualified source of training may be used. In addition, every driver will be trained to meet special DOE requirements pertaining to the specific characteristics of the TRUPACT-II shipping containers, the transportation of radioactive materials, monitoring equipment, emergency response, and public relations.

In addition, the drivers will be required to attend a training class conducted by the Transportation Safeguards Division of the DOE's Albuquerque Operations Office. This training will be comprehensive, requiring approximately 68 hours. One instructor will be provided for each two drivers. The training will include driving a WIPP tractor-trailer unit carrying TRUPACT-II containers with simulated loads.

Before the actual shipment of any waste, multiple dry runs from each waste site will be conducted as part of a series of preoperational checks designed to provide experience and hands-on training to the drivers (see Appendix D.2.3.2).

FIGURE M.5.1
DRIVER QUALIFICATION FORM

M.6 **PROCEDURES USED IN WASTE TRANSPORTATION**

M.6.1 **RESPONSIBILITY FOR DAILY OPERATIONS**

The manager/dispatcher at the Hobbs maintenance facility will be responsible for the daily operations of the trucking contractor. The dispatcher will receive and review trip schedules furnished by the WIPP. These schedules will be furnished for intervals of no less than 6 weeks. If there are problems about the schedules, the dispatcher will immediately communicate with the WIPP to resolve the problems.

The dispatcher will prepare and distribute a 30-day schedule to all drivers. If a driver notifies the dispatcher that there are problems with the schedule, the dispatcher will resolve the problem.

The dispatcher will be reachable by beeper or telephone at all times when not in the dispatch facility.

M.6.2 **NUMBER OF DRIVERS**

Two qualified drivers will be used for each shipment of TRU waste. If a driver becomes incapacitated along the way, the alternative driver will ask and receive appropriate instructions from the dispatcher before proceeding.

M.6.3 **SECURITY**

Standard security requirements for materials in transit, as specified in DOE Order 1540.1, will be applied to the TRUPACT-II shipping containers in both the loaded and unloaded condition. Constant surveillance will be provided for each shipment (Subsection M.6.7), and the drivers will know the procedures to be followed in the event of a deliberate obstruction of a shipment. In addition, the location of each TRU waste shipment will be known at all times, via the TRANSCOM satellite-based tracking system (Section M.8).

M.6.4 **PROCEDURES TO BE FOLLOWED BEFORE THE START OF THE TRIP**

The drivers will report to the dispatch center in Hobbs 1 hour before the scheduled time departure. The driver will check in and receive trip routing instructions. The dispatcher will verify that the drivers have arrived to review the route to be taken for the trip. The routes to be taken are the routes defined as "preferred" in Federal regulations. The two drivers assigned to the trip will review the trip route together. If they have any questions, they will discuss them with the dispatcher.

The drivers will obtain a copy of the pretrip inspection form (Figure M.6.1) and the trip report form from the previous trip. They will inspect the truck and the trailer, paying particular attention to any items mentioned as possibly defective in the post-trip report. The drivers will sign the pretrip report if the tractor and the trailer meet requirements. The inspection will include all extra equipment.

If their inspection of the tractor and trailer shows that an item or items do not meet the required standards, the drivers will notify the dispatcher. The dispatcher will decide whether the tractor and trailer are to be dispatched in their current condition or whether further maintenance is required.

If the dispatcher decides to dispatch the tractor and trailer without further maintenance, the drivers have the option of noting their concurrence or nonconcurrence with the decision of the dispatcher. If the dispatcher decides to use another tractor or trailer, the drivers will carry out the same inspection routine.

M.6.5 **PROCEDURES TO BE FOLLOWED AT THE WIPP SITE**

At the WIPP site there will be two trailer-parking areas. Parking Area A will be for trailers incoming with loaded TRUPACT-II shipping containers and trailers that have been inspected by the trucking contractor and are ready to be loaded. Parking Area B will be for empty trailers that require inspection or maintenance and for trailers that are ready for shipment and are loaded with empty TRUPACT-II containers.

At the WIPP site the drivers will present the necessary identification and documentation and receive the shipment documentation, including a manifest which, for mixed waste shipments, conforms to the requirements of 40 CFR Part 263. They will then proceed to the trailer-storage area. At the trailer-storage area, the drivers will leave their tagged empty trailer in Parking Area A and verify that the trailer (from Parking Area B) loaded with empty TRUPACT-II containers has been tagged as ready for service. The drivers will then inspect the trailer, using the trailer-inspection form. As part of the pretrip inspection, the drivers must ensure that the permanently affixed flip-type placards properly signify whether the trailer is carrying a load containing radioactive material or is empty.

If the trailer meets all inspection requirements, the drivers will sign the trailer-inspection sheet and depart from the WIPP site. The departure will follow the correct procedures for notification and departure.

If the trailer does not meet the required standards, the drivers will notify the WIPP and the dispatcher. The drivers will then await a decision by the WIPP and the dispatcher concerning the departure of the trailer.

FIGURE M.6.1
EXAMPLE OF DRIVER'S VEHICLE INSPECTION FORM

M.6.6 **GENERAL PROCEDURES TO BE FOLLOWED DURING THE TRIP**

The drivers must use the preferred route for shipments unless a deviation is permitted under the provisions of 49 CFR 177.825. A deviation is permitted by 49 CFR 177.825 under the following circumstances:

1) Emergency conditions that would make continued use of the preferred route unsafe

2) To make necessary rest, fuel, and vehicle-repair stops (stops will be along the preferred route)

3) To the extent necessary to pick up, deliver, or transfer a highway route controlled quantity package of radioactive materials.

Any required deviation will be reported to the DOE's representative at the WIPP before the deviation occurs. Any unauthorized deviation from the preferred route will result in penalties, as discussed at the end of this section.

Drivers may alternate driving shifts of approximately 5 hours. Thus, the vehicle will be constantly moving unless stopped for inspection, fueling, or weather. When circumstances require an extended stop, the driver will ensure that the shipment is parked in a safe manner.

M.6.7 **CONSTANT SURVEILLANCE**

One driver will keep the tractor and trailer under constant surveillance at all times. Constant surveillance is defined to mean that when the vehicle is not being driven, it must be attended at all times by a driver or a qualified representative of the trucking contractor. A vehicle is "attended" when at least one driver is in the tractor, awake, not in a sleeper berth, or within 100 feet of the vehicle and has the vehicle within his or her constant unobstructed view.

If an extended stop is necessary, a driver must keep the shipment in full view and stay within 100 feet of the shipment at all times.

The trailer with the TRUPACT-II containers must always be connected to the designated tractor during shipment except when stopped at a DOE facility for loading, unloading, or en route to maintenance.

M.6.8 **INSPECTIONS DURING THE TRIP**

The drivers will park the vehicle in a safe place every 2 hours of travel time or 100 miles, whichever is less, and inspect the vehicle.

Deficiencies will be corrected at this time or at the next available repair area. The items to be inspected include the tires, tiedowns, labeling and placarding required for the transportation of radioactive materials, and the antenna used for the TRANSCOM vehicle-tracking equipment (see Section M.8). Items found to be nonconforming will either be corrected at this time or at the next available repair area. If a tire is found to be flat, leaking, or improperly inflated, the tire will be changed or properly inflated. The drivers will also inspect the vehicle lights if lights will be needed before the next stop. Hose connections will be checked, and a visual inspection of the entire vehicle will be made.

The DOT regulations in 49 CFR 397.17 ("Transportation of Hazardous Materials: Driving and Parking Rules") require only tire inspections every 2 hours on vehicles carrying hazardous materials. The DOE has expanded this inspection requirement to include other components and to include unloaded vehicles.

M.6.9 **PROCEDURES AT THE WASTE SITE**

On arrival at the waste site, the drivers will stop at an inspection point where the driver and shipment documentation will be checked by site security before the tractor and trailer are permitted entrance. Specific items to be verified are the bill of lading, tamper-indicating devices, and the serial numbers of the TRUPACT-II shipping containers. The drivers will then proceed to the trailer-parking area and drop off the trailer with the empty TRUPACT-II containers. The drivers will undertake an after-trip inspection of the trailer. They will then proceed to the location of the trailer with loaded TRUPACT-II containers, or, if at a low-volume site, find out when they should return to pick up the trailer after it has been loaded.

The drivers will receive trip documentation and inspect the trailer, using the trailer-inspection form. The drivers will also inspect the tractor before departing from the waste site. The drivers will follow the approved departure procedure when leaving the site. The drivers will then proceed to the WIPP site, using the same routes and procedures used with the empty TRUPACT-II shipping containers.

M.6.10 **PROBLEMS DURING THE TRIP**

If the dispatcher is notified by the driver of a problem during the trip, the dispatcher will notify the DOE's representative at the WIPP.

If the WIPP notifies the dispatcher that a problem exists, the dispatcher will immediately contact the drivers to ensure that procedures are being followed and to obtain firsthand information on the situation. The dispatcher will decide on the best course of action and notify the WIPP of the decision. If the WIPP concurs, the decision will be implemented. If the WIPP does not concur, further discussions will take place.

When notified of a mechanical problem that prevents the tractor or trailer from moving, the dispatcher will immediately make arrangements to rectify the situation after consultation with the WIPP. If a leased tractor is to be used, the dispatcher will consult the list of locations where tractors are available for leasing from a qualified leaser and determine the most convenient location. The leaser will be called and asked to dispatch a tractor that will allow the shipment not to exceed a total weight of 80,000 pounds. The WIPP and the drivers will be notified of the expected time of arrival.

All drivers will carry full instructions for actions to be taken in the event of an accident. The procedures to be followed after an accident are discussed in Subsection M.7.

M.6.11 **DELIVERY OF WASTE AT THE WIPP SITE**

On arrival at the WIPP site, the driver will stop at an inspection point where the driver and shipment documentation must be checked by site security before the shipment is permitted into the secured area. Specific items to be verified are the bill of lading, tamper-indicating devices, and the serial numbers of the TRUPACT-II shipping containers. Shipments will have a radiation survey performed in the designated secure area before entry into the site. The drivers will be badged and proceed to a receiving-inspection position in the radioactive-materials area.

When a shipment arrives at the WIPP site, one driver will remain with the vehicle at all times. The driver will position the trailer as required for further processing in one of the parking areas. After the trailer has been removed, the tractor and drivers will be released. If an empty trailer is available, the drivers will pick up the empty trailer from Parking Area B for delivery to the maintenance facility. The drivers will then return to the maintenance facility with the tractor or tractor and trailer.

M.6.12 **AFTER-TRIP REPORT**

At the conclusion of each round trip, the drivers will complete the driver's vehicle-condition report for the tractor and trailer. They will review the report with the maintenance supervisor. The drivers will be encouraged to present their observations on the performance of the vehicle (tractor and trailer).

M.6.13 **PENALTIES FOR DRIVERS**

If the drivers fail to follow the prescribed procedures, they will be subject to penalties. For an unauthorized deviation from the preferred route, the penalties will be as follows:

- First time -- written warning and 2 weeks' leave without pay
- Second time -- termination of the driver's employment.

A failure to maintain adequate records will result in the same penalties as deviating from the route.

The failure to maintain constant surveillance of the vehicle will result in a termination of the driver's employment.

A chargeable accident will result in a termination of the driver's employment.

A moving violation will result in a termination of the driver's employment.

M.7 **PROCEDURES FOR ACCIDENTS AND INCIDENTS**

All drivers will carry full instructions for actions to be taken in the event of an accident. These instructions will include the procedures for obtaining local, State, or Federal assistance if technical advice or emergency assistance is needed. The TRANSCOM equipment (Section M.8) will provide a communications capacity that can be used in any emergency.

The accidents to be reported are those specified in the applicable Federal regulations, 49 CFR 171.15 and 171.16, the general requirements of 49 CFR Part 394, and the requirements of DOE Order 1540.1.

All accidents, no matter how minor, will be reported to the traffic manager of the waste site, the WIPP, and the dispatcher. Accident reporting will follow normal procedures (49 CFR Part 394) for minor accidents that involve no obvious or suspected damage to the TRUPACT-II shipping containers. In the event of a Type A accident (as defined in DOE Order 5484.1), it will be necessary to notify the DOE Headquarters Emergency Operations Center, and this notification will be made through the Albuquerque Operations Office. The trucking contractor will notify the DOE's Albuquerque Operations Office, the U.S. Department of Transportation, the WIPP, and the shipper in the event of fire and damage in excess of $5,000, breakage, spillage, or suspected contamination with radioactive material, as required by 49 CFR 171.5 and 171.861.

When notified of an emergency situation, the dispatcher will immediately contact the WIPP. If action is needed by the dispatcher, such action will be taken with the concurrence of the WIPP. These actions may include, but are not limited to, the following:

- Having the vehicle repaired
- Dispatching a replacement tractor
- Sending replacement drivers
- Coordinating a route deviation
- Authorizing shipment of replacement parts.

The dispatcher will maintain a log of actions taken during the emergency, including the time of each action. A copy of the record will be sent to the WIPP.

If the drivers perceive a potential obstruction because of a public demonstration, the drivers will immediately notify the local law enforcement agency and the WIPP and describe the situation. The WIPP will advise the drivers as to what action to take. If it is determined by the drivers that the trip should not continue, the drivers will move the tractor to the most secure nearby location, if feasible, and remain with the vehicle.

If it is determined by the drivers that the tractor and trailer cannot be moved because of a deliberately placed obstruction or public demonstration, the drivers will do the following:

1) Notify the WIPP immediately
2) Notify the local law enforcement agency or the State highway patrol
3) Remain in the tractor with the doors secured.

M.8 SHIPMENT TRACKING AND COMMUNICATIONS

M.8.1 SHIPMENT TRACKING

The location of each TRU waste shipment will be monitored in order to maintain shipping and receiving schedules and to learn of any unplanned deviation from the schedule or preferred route. This monitoring will include the status of the shipment at the WIPP site or at the waste site as well as location during transit.

The primary method for monitoring or tracking TRU waste shipments will be the TRANSCOM locating system. TRANSCOM will use a land-based Loran C positioning system to obtain exact data on the longitude and latitude. It will have a transmitter to transmit the Loran C data via satellite to the TRANSCOM Control Center at Oak Ridge, Tennessee, which will be linked to the Central Communications Center at the WIPP (see Appendix D). The transmissions will be converted to location data by the TRANSCOM central computer.

TRANSCOM will provide a two-way digital means of communication. However, with the TRANSCOM system providing routine data, communication by the driver will be required only in the event of significant schedule impacts, such as accidents or delays that affect the delivery schedule by 2 hours or more.

M.8.2 BACKUP COMMUNICATIONS

In the event that the TRANSCOM location system is not available, telephone communications will be used, and the drivers will use the mobile telephone provided. Telephone communications will also be used by the dispatcher and by the waste site to report to the WIPP. To facilitate telephone communications, 800 numbers will be available. The required reports will be as follows:

- The drivers will be required to make a telephone call to the WIPP every 2 hours and when crossing State borders, or as soon thereafter as practical, to report their location.

- Any delays and the reason for delays in transit longer than 2 hours will be reported by the trucking contractor to the WIPP, who will in turn relay the information to the waste site.

- The waste site will notify the WIPP at the time the shipment leaves the site. The notification will include the tractor and trailer numbers, the serial numbers of the TRUPACT-II containers, the drivers' names, the bill-of-lading number, the shipment weight, the route, the date and time the vehicle departed, and the expected arrival time.

APPENDIX N

RE-EVALUATION OF RADIATION RISKS FROM WIPP OPERATIONS

TABLE OF CONTENTS

LIST OF TABLES

N.1 **INTRODUCTION**

Since the supplemental risk assessment process was initiated, there have been two new evaluations of the risks posed by radiation exposure published (BEIR, 1988 and UNSCEAR, 1988).[1] In response to comments made by the DOE during its internal review of the draft SEIS, this appendix has been prepared to evaluate the extent to which these recent studies may affect the estimation of risks reported in this SEIS.

The selection of a risk estimator to evaluate the radiation-induced human health effects of WIPP operations is discussed in Subsection 5.2.2.1. These estimated health risks are summarized in Table 5.14 for transportation-related exposures and in Tables 5.29 and 5.30 for WIPP routine and accident-related exposures, respectively. To establish that the risk estimators utilized provide a conservative estimation of health risk, a comparison is made between certain reported health risks and those which would be predicted by a rigorous application of data provided by the newly available studies.

Based upon data from the BEIR-III report (BEIR, 1980), risk estimators for both cancer incidence and genetic effects have been developed to estimate health effects associated with the calculated doses to the population and individuals. For cancer incidence, a risk estimator of 280 fatal cancers per million person-rem of radiation (external dose plus committed effective dose equivalent) received by the affected population has been used. For genetic effects, a risk estimator of 257 genetic effects per million live-born offspring for each additional rem of radiation received by the gonads of the affected population has been used.

---

[1] On December 20, 1989, the National Research Council's Committee on the BEIR issued a report on the health effects of exposure to low levels of ionizing radiation (BEIR, 1989). This report includes information and analyses from the BEIR-IV report (BEIR, 1988) that are appropriate for cancer and genetic risk assessment along with the delayed health effects that are induced by low linear energy transfer (LET) radiations such as x-rays and gamma radiation. These health effects include fatal cancer induction (carcinogenesis), genetic effects, and retardation from in utero exposure. Quantitative risk estimates based on statistical analyses of the results of human epidemiological studies and animal experiments are presented in the BEIR-V report. A significant portion of the BEIR-V report deals with carcinogenesis in humans because of the extended follow-up in major epidemiological studies (e.g., Japanese atomic-bomb survivors and radiotherapy patients) and the revision of the dosimetric system for the Japanese atomic-bomb survivors.

The report presents risk factors that are higher than proposed in the BEIR-III report (BEIR, 1980). The BEIR-V report estimates that 800 extra cancer deaths would be expected to occur during the exposed population's remaining lifetimes if 100,000 people of all ages were exposed to a whole body dose of 10 rad (or 10 rem) of gamma radiation in a single brief exposure. These 800 excess cancer deaths are in addition to the nearly 20,000 cancer deaths that would occur in the absence of the radiation. This corresponds to a risk factor of 8.0×10^{-4} excess fatal cancers per person-rem (this SEIS used 2.8×10^{-4} excess fatal cancers per person-rem). The 90 percent confidence limits, based solely on sampling variation, for increased cancer mortality due to an acute whole body dose of 10 rem range from about 500 to 1,200 (mean 760) for 100,000 males of all ages and from about 600 to 1,200 (mean 810) for 100,000 females of all ages. The report also recommends using the relative risk model (as used in Subsection N.3) instead of the constant absolute or additive risk model.

The report recognizes that the assessment of carcinogenic risks that may be associated with low doses of radiation requires extrapolation from effects observed for doses exceeding 10 rad and is derived from assumptions about dose-effect relationships and the mechanisms of carcinogenesis. In the analysis of the epidemiological data for the atomic-bomb survivors, the survivors receiving less than 0.5 rad serve as a control group for the survivors receiving more than 0.5 rad. The report also recognizes that its risk estimates become more uncertain when applied to very low doses; however, the risk estimates could either increase or decrease. For low-LET radiations such as gamma rays, the consensus is that cell survival is enhanced by a decrease in dose rate or separation of the dose into several fractions. To apply the models derived from the data on acute exposures, the dose rate effectiveness factor must be considered. The BEIR-V report indicates that it may be desirable to reduce the estimates given above by a factor of 2 for application to populations exposed to small doses at low dose rates because of the dose rate effectiveness factor.

The report recognizes many uncertainties in its analyses. These include the application of results from a Japanese population (with different naturally occurring cancer rates) to a United States population, the certification of the cause of death, time- and age-related effects, and the shape of the dose-response curve. It also recognizes that direct estimates of the lifetime risk can be obtained only after the exposed population has been followed for a lifetime; however, the Japanese survivors (one of the populations followed for the longest time) have been followed for only 40 years. The report also states that studies of populations chronically exposed to low-level radiation (e.g., those residing in regions with elevated natural background radiation) have not shown consistent or conclusive evidence of an associated increase in the risk of cancer.

The risk factors presented in BEIR-V, which became available as this SEIS was in the final stages of completion, are not incorporated in the risk estimates. The DOE will have to study the report thoroughly to determine any warranted changes in risk estimation methods for the generally low dose/low dose rate circumstances analyzed in this SEIS. The purpose of this SEIS, however, is to provide environmental impact information for deciding whether to proceed to the Test Phase (Proposed Action or Alternative Action). In this context, BEIR-V is not significant because 1) the likely increases in risk estimates are relatively small; 2) they affect all alternatives, including No Action; and 3) the DOE will issue another SEIS--using the then current risk assessment methods--before a decision to enter the Disposal Phase, during which most of the radiological impacts associated with the WIPP are predicted to occur.

N.2 REVIEW OF RECENTLY PUBLISHED RADIATION RISK EVALUATIONS

Two recently published evaluations of the risks posed by exposure to ionizing radiation contain data relevant to the radionuclide distribution for the WIPP. These studies are reviewed in terms of determinations and recommendations associated with predicting human health risk from exposure to alpha-emitting radionuclides.

N.2.1 **BEIR-IV**

In January, 1988, the National Research Council's Committee on the Biological Effects of Ionizing Radiations (BEIR) issued a report reviewing available information on the health risks of alpha-emitting radioactivity which has deposited inside the human body (BEIR, 1988). This information is directly relevant to the WIPP, since virtually all of the radionuclides present in TRU waste are alpha-emitters.

In their review, the BEIR Committee determined that the effects of internally-deposited TRU radionuclides occur predominantly in three organs: the bone, the liver, and the lung. Based on data from animal studies as well as limited human exposure data, the BEIR Committee recommended latency periods (i.e., the time between exposure to radiation and the onset of cancer) and risk factors for these organs as follows:

Organ	Latency Period (years)	Fatal Cancer Risk (deaths/million person-rad)
Bone	5	300
Liver	20	300
Lung	5	700

For the bone risk factor, the absorbed dose used is the mean bone dose.

N.2.2 **UNSCEAR**

In 1988, the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) issued the latest in a series of reports to the General Assembly, providing a comprehensive assessment of the sources, effects, and risks of ionizing radiation (UNSCEAR, 1988). In this report, the Committee reviewed available data on radiation exposures and risk estimates.

The Committee recommended a range of risks for radiation-induced fatal cancer. Adjusting for the effects of low doses/dose rates as prescribed by UNSCEAR, the absolute lifetime risk of radiation is 200 to 250 fatal cancers per million person-rad. Latency periods were given as a minimum of 2 to 5 years between exposure and the onset of either leukemia or bone cancer and 10 years for all other types of cancer.

These values are similar to those proposed in the BEIR-III report and used in this SEIS.

N.3 **REASSESSMENT OF RISKS FROM WIPP OPERATIONS**

Based on the information in the reports discussed in Subsection N.2, a reassessment of the risks posed by WIPP operations was performed. The approach used is patterned after the RADRISK computer code (ORNL, 1980), and could be applied to any aspect of the WIPP where radiological dose assessments are performed, including the transportation risk assessment. To establish that the risk estimators used in this SEIS remain conservative, facility operational impacts were selected for reassessment.

N.3.1 <u>**METHODOLOGY SELECTED**</u>

The methodology selected for this assessment uses a life table approach to predict the estimated lifetime risk of fatal cancer from exposure to radiation/radioactivity emitted during the operation of the WIPP.

The reassessment calculates the effects of exposure to two types of radiation:

- Low Linear Energy Transfer (LET) radiation (such as gamma and beta radiation), because of its penetrating nature, can cause damage from either outside the body, from external sources, or inside the body, once ingested or inhaled

- High-LET radiation (such as alpha particles or neutrons) is primarily made up of less penetrating alpha radiation, which can cause damage once inside the body.

Low-LET radiation exposure risk at the WIPP during normal operations is associated almost completely with WIPP occupational workers who are subject to external exposure to gamma radiation while handling the waste containers (primarily the CH TRU shipping containers and containers of TRU waste). WIPP employees and the off-site population can also be exposed to gamma and beta radiation from a plume of radioactivity released in the event of a postulated accident. Radiation doses to low-LET radiation are described in Subsections 5.2.3.3 and 5.2.3.4. The prediction of fatal cancers associated with low-LET radiation exposure uses relationships between absorbed dose and risk developed in the BEIR-III report (BEIR, 1980). The relationships selected use a linear quadratic form to express the relationship between absorbed doses and the risk of cancer:

1) Leukemia and bone cancer (BEIR-III, Table V-16)

2) All other types of cancer (BEIR-III, Table V-19).

The relationships were combined to generate the formulae used in the lifetable.

In accordance with BEIR-III, a 10-year latency period is assumed for low-LET radiation prior to the onset of cancer. Any radiation-induced cancer will not begin to develop until the end of this latency period. In the eleventh year, the risk would be related to the exposure in the first year; the risk in the twelfth year would be related to the exposure in the first and second years; risk in subsequent years would be evaluated in the same manner.

Once the latency period had passed, an exposed individual would have a risk of radiation-induced cancer for the remainder of his/her lifetime. If the exposure is continued, the risk would continue to increase. When the exposure is stopped (e.g., by termination of WIPP operations), the risk would continue to increase for the length of the latency period and thereafter would remain constant. Specifically for low-LET radiation and 25 years of operation, the risk of radiation-induced cancer would begin in the eleventh year and continue to increase until the thirty-sixth year, when it would become constant for the duration of the individual's lifetime. The risk, in the thirty-sixth and following years, would be dependent on the total exposure during the 25 years of operation.

The dose equivalents caused by high-LET radiation exposure to WIPP waste are the result of inhaling, and to a lesser extent, ingesting alpha-emitting radioactivity. They are expressed in terms of committed effective dose equivalents (CEDE's), which provide a measure of the damage done to the body over a 50-year period due to an intake in a single year. These CEDE's are described in Subsections 5.2.3.3 and 5.2.3.4. To assess the impact of these CEDE's on human health, they are converted into organ doses to the bone, the liver, and the lung as identified by the BEIR-IV report (BEIR, 1988).

The prediction of fatal cancers associated with high-LET radiation is accomplished through a series of steps:

1) The conversion of CEDE's to annual effective dose equivalents

2) The conversion of annual effective dose equivalents to annual organ dose equivalents

3) The prediction of fatal cancers for each organ

4) The summation of the organ fatal cancer risks to predict the total risk of cancer.

The waste going to the WIPP will contain a variety of radionuclides which emit high-LET radiation. In an attempt to simplify the evaluation of the various types of radionuclides, the SEIS uses the concept of the "Plutonium-239 Equivalent Curie (PE-Ci)." This concept, described in Appendix F.2, uses the ratio of effective dose equivalent conversion factors between a radionuclide and plutonium-239 (Inhalation Class W) to convert each radionuclide's concentration into an equivalent concentration of plutonium-239(W). All analyses then treat the waste as though plutonium-239(W) were the only radionuclide present. The dose conversion factors used are for the inhalation pathway, using a 1.0 micron aerodynamic median activity diameter (AMAD) and a 50-year commitment period (Dunning, 1986).

Since the retention time for plutonium-239 in the human body is so long (ICRP, 1979), this methodology assumes that the radioactivity remains in the organ of interest for an indefinite period. Thus, the 50-year CEDE's are converted to annual effective dose equivalents simply by dividing by 50. Further, the annual effective dose equivalents are assumed to continue throughout the population's lifetime (i.e., they do not stop at the end of the 50-year period).

To obtain the dose equivalent to the three specific organs of interest (bone, liver, and lung), each annual effective dose equivalent is multiplied by the ratio of the organ CEDE dose conversion factor to the effective dose conversion factor for plutonium-239(W) (Dunning, 1986).

To ensure that this approach was conservative, the conversion factors from effective dose equivalent to organ dose equivalent were calculated for all organs of interest. For the liver and the bone, the assumption that all the activity was plutonium-239(W) was found to be conservative. For the lung, however, there were two radionuclides (uranium-233 and californium-252) which have higher conversion factors. To account for this difference, the conversion factor from effective to organ dose equivalent for the lung was adjusted based on the anticipated concentrations of these two radionuclides in the waste.

One additional adjustment had to be made. The risks of bone cancer are expressed in terms of the mean bone dose. The organ dose equivalent conversion factor used for bone in this SEIS considers the endosteal cells only. To calculate risks, the mean bone dose risk estimator has to be converted to an endosteal dose risk estimator. The conversion was accomplished using the bone dosimetry model published by the International Commission on Radiological Protection (ICRP, 1979).

Once these conversions are made, the number of excess fatal cancers can be predicted using the risk factors and latency periods contained in the BEIR-IV report (see Subsection N.2.1).

The reassessment evaluated risks from both routine WIPP emissions and postulated accidental releases. For routine emissions, the reassessment follows a cohort of people (evenly distributed between the two sexes) through a 109-year lifetime. All people in this cohort are assumed to be simultaneously liveborn at the time the WIPP goes operational. The cohort is exposed to radioactivity/radiation for the 25 years of WIPP operations. The first 5 years are associated with the WIPP's Test Phase. The remaining 20 years are associated with the WIPP's Disposal Phase.

For each year of the cohort's lifetime, the lifetable takes the following steps:

1) Given the population existing at the beginning of the year, the total background mortality, the total background cancer mortality, and the background mortalities for bone, liver, and lung cancer are calculated.

2) The high-LET annual effective dose equivalents associated with the WIPP are converted into bone, liver, and lung dose equivalents, and the number of predicted excess fatal cancers is calculated based on those dose equivalents and the starting population. Latency periods are built into the calculation for each type of cancer.

3) The low-LET annual effective dose equivalent associated with the WIPP is converted into an annual predicted number of excess fatal cancers using the starting population and the dose equivalent (if any). The risk in subsequent years due to a given year's detriment (the actual external plus the CEDE) is corrected to reflect the decrease in the cohort population over time. A latency period is also built into this calculation.

4) The population surviving at the end of the year is calculated by subtracting the background mortality and the predicted numbers of excess fatal bone, liver, lung, and low-LET cancer from the population living at the beginning of the year.

At the end of the 109-year lifetime, the excess number of fatal cancers was totalled.

The reassessment also calculated predicted excess fatal cancers from effective dose equivalents received by individuals during postulated accidental WIPP releases. The reassessment follows a cohort of people (evenly distributed between the two sexes) through a 109-year lifetime. All people in this cohort are assumed to be simultaneously liveborn at the time of the postulated accident and exposed to radioactivity/radiation from the accident event. Deaths are calculated as described above for routine operations. At the end of the 109-year lifetime, the excess number of cancer deaths was totalled and divided by the number of people assumed for the cohort to arrive at the excess fatal cancer risk to an individual.

N.3.2 **SCENARIOS SELECTED**

In order to make health effects comparisons between results obtained utilizing the SEIS methodology and those calculated using the more rigorous approach described above, four dose consequence calculations were selected. These four calculations are not all inclusive but are representative of the full range of exposure pathways, radiation types, and individual and population assessments addressed by the SEIS.

1) The collective CEDE received by the off-site population during normal operations (see Table 5.23)

2) The collective CEDE received by the WIPP's employee population (waste handling crew) during normal operations (see Table 5.24)

3) The highest predicted CEDE to a member of the public, that is associated with postulated accident C-10 (see Table 5.28)

4) The highest predicted CEDE to a WIPP employee, that associated with postulated accident C-3 (see Table 5.28).

For each of these scenarios, the total number of predicted fatal cancers was calculated. Similar values for excess fatal cancers were calculated based upon the SEIS health effects estimates of 280 fatal cancers per million person-rem of population detriment.

N.3.3 **PUBLIC HEALTH EFFECTS**

The total numbers of predicted excess fatal cancers using the two assessment methodologies are shown in Table N.3.1. The table shows that the estimated health effects associated with WIPP operations as reported in this SEIS overstate estimates obtainable from the latest available recommendations for assessing human health effects associated with radiation exposure.

An example of the lifetable analysis is presented in Table N.3.2 for the population risk resulting from routine WIPP emissions.

TABLE N.3.1 Estimated excess fatal cancers caused by WIPP operations during the Test and Disposal Phases[a]

	SEIS	BEIR-IV

Scenario	methodology	methodology
Off-site population due to routine WIPP emissions[b]	6.8×10^{-6}	3.2×10^{-6}
WIPP employee population during routine WIPP operations[c]	1.0×10^{-1}	4.1×10^{-2}
Maximum off-site individual due to postulated WIPP accident C-10	4.8×10^{-4}	1.6×10^{-4}
Maximum worker due to postulated WIPP accident C-3	1.7×10^{-3}	5.6×10^{-4}

[a] Population risks are expressed as the total number of excess fatal cancers in the entire population. Individual risks are most easily interpreted as the excess risk of an individual contracting a fatal cancer (e.g., 4.8×10^{-4} represents 48 chances in 100,000).

[b] Off-site population is 112,966 people living within 50 miles of the WIPP.

[c] Employee population is 18 radiation workers.

N.3.4 **GENETIC EFFECTS**

The references mentioned in Subsection N.2 also discuss the genetic effects of radiation exposure. Based on the data currently available, the following genetic risk factors for subsequent generations apply to WIPP radiation doses:

Type of Radiation	Genetic Risk Factor (per million live offspring per rad)
Low-LET (UNSCEAR, 1988)	120
High-LET (BEIR, 1988)	600

Using these risk factors, the genetic risk caused by WIPP emissions (both routine and accidental) were calculated. For high-LET radiation, the calculation involved three steps:

1) Converting the CEDE for each scenario into a committed dose equivalent (CDE) to reproductive organs (testes and ovaries)

2) Dividing the CDE by the quality factor for alpha radiation (20) to convert dose equivalent to absorbed dose (rem to rad)

3) Multiplying the committed dose by the genetic risk factor to obtain the risk to subsequent generations.

For low-LET radiation, the first two steps were not necessary since the dose equivalent is uniform over the whole body and the quality factor for low-LET radiation is 1. The results of these calculations are shown in Table N.3.3.

These risks were then compared with the risk of fatal cancer associated with the particular scenario. The ratio of the genetic risk to the excess fatal cancer risk is also shown in Table N.3.3. In all cases, the risk of genetic effects was less than 93% of the cancer risk. The major factor affecting the magnitude of the risk was the low-LET contribution. For the transuranic elements present in the waste at the WIPP, the CDE to reproductive organs is a fraction of the CEDE. This fact and the large quality factor for alpha radiation were the principal reasons for the lower contribution of high-LET radiation.

These results support the conclusion made in this SEIS that the risk of fatal cancer provides the most conservative measure of the health effects caused by WIPP operations.

TABLE N.3.3 Estimated excess genetic effects caused by WIPP operations

Scenario	Excess genetic effects[a]	Ratio of excess genetic effects to excess fatal cancers[b]
Off-site population due to routine WIPP emissions[c]	5.6×10^{-8}	0.02
WIPP employee population during routine WIPP operations[d]	3.8×10^{-2}	0.93
Maximum off-site individual due to postulated WIPP accident C-10	7.1×10^{-6}	0.04
Maximum worker due to postulated WIPP accident C-3	2.6×10^{-5}	0.05

[a] Population risks are expressed as the total number of excess genetic effects appearing in live-born offspring in all future generations of the exposed population. Individual risks are most easily interpreted as the excess risk of a genetic effect appearing in the live-born offspring in all future generations of the exposed individual.

[b] Excess fatal cancers taken from Table N.3.1, BEIR-IV methodology. The ratios presented compare to the 0.918 risk estimator used in this SEIS.

[c] Off-site population is 112,966 people living within 50 miles of the WIPP.

[d] Employee population is 18 radiation workers.

REFERENCES FOR APPENDIX N

BEIR, 1989. Health Effects of Exposure to Low Levels of Ionizing Radiation, National Academy of Sciences, Committee on the Biological Effects of Ionizing Radiations.

BEIR, 1988. Health Risks of Radon and Other Internally Deposited Alpha Emitters, National Academy of Sciences, Committee on the Biological Effects of Ionizing Radiations.

BEIR, 1980. The Effects on Populations of Exposure to Low Levels of Ionizing Radiation: 1980, National Academy of Sciences, Committee on the Biological Effects of Ionizing Radiations.

Dunning Jr., D.E., 1986. Estimates of Internal Dose Equivalent from Inhalation and Ingestion of Selected Radionuclides, WIPP-DOE-176, Rev. 1, prepared for the U.S. Department of Energy.

ICRP (International Commission on Radiological Protection), 1979. Annals of the ICRP, ICRP Publication 30, Limits for Intake of Radionuclides by Workers, Pergamon Press, Washington, D.C.

ORNL (Oak Ridge National Laboratory), 1980. Dunning Jr. D.E. et al, A Combined Methodology for Estimating Dose Rates and Health Effects from Exposure to Radioactive Effluents, ORNL/TM-7105, December, 1980.

UNSCEAR, 1988. United Nations Scientific Committee on the Effects of Atomic Radiation, Sources, Effects, and Risks of Ionizing Radiation, 1988 Report to the General Assembly, with Annexes.

TABLE N.3.2 Lifetable for population dose and risk resulting from routine emissions

	Year of Operation										
	1	2	3	4	5	6	7	8	9	10 - 24	25
Collective CEDE (person-rem)	4.7×10^{-4}	4.7×10^{-4}	4.7×10^{-4}	4.7×10^{-4}	4.7×10^{-4}	1.1×10^{-3}	1.1×10^{-3}	1.1×10^{-3}	1.1×10^{-3}	a	1.1×10^{-3}
External EDE (person-rem)	0.0×10^{0}	0.0×10^{0}	0.0×10^{0}	0.0×10^{0}	0.0×10^{0}	0.0×10^{0}	0.0×10^{0}	0.0×10^{0}	0.0×10^{0}	a	0.0×10^{0}
Committed DE (person-rem)	4.7×10^{-4}	4.7×10^{-4}	4.7×10^{-4}	4.7×10^{-4}	4.7×10^{-4}	1.1×10^{-3}	1.1×10^{-3}	1.1×10^{-3}	1.1×10^{-3}	a	1.1×10^{-3}

Age of individual	Test Phase			Disposal Phase						lung	liver	bone	Excess	Excess	Excess	Excess	Total	
	1	2	3	4	5	6	7	8	9	10 - 24	25	Summed dose	Population	cancer deaths	cancer deaths	cancer deaths	Cancer deaths	Natural deaths
0	9.4×10^{-6}	0.0×10^{0}	0.0×10^{0}	0.0×10^{0}	0.0×10^{0}	0.0×10^{0}	0.0×10^{0}	0.0×10^{0}	0.0×10^{0}	a	0.0×10^{0}	9.4×10^{-6}	112,966	0.0×10^{0}	0.0×10^{0}	0.0×10^{0}	0.0×10^{0}	2.3×10^{3}
1	9.4×10^{-6}	9.4×10^{-6}	0.0×10^{0}	0.0×10^{0}	0.0×10^{0}	0.0×10^{0}	0.0×10^{0}	0.0×10^{0}	0.0×10^{0}	a	0.0×10^{0}	1.9×10^{-5}	110,704	0.0×10^{0}	0.0×10^{0}	0.0×10^{0}	0.0×10^{0}	1.4×10^{2}
2	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	0.0×10^{0}	0.0×10^{0}	0.0×10^{0}	0.0×10^{0}	0.0×10^{0}	0.0×10^{0}	a	0.0×10^{0}	2.8×10^{-5}	110,566	0.0×10^{0}	0.0×10^{0}	0.0×10^{0}	0.0×10^{0}	9.5×10^{1}
3	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	0.0×10^{0}	0.0×10^{0}	0.0×10^{0}	0.0×10^{0}	0.0×10^{0}	a	0.0×10^{0}	3.8×10^{-5}	110,471	0.0×10^{0}	0.0×10^{0}	0.0×10^{0}	0.0×10^{0}	7.6×10^{1}
4	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	0.0×10^{0}	0.0×10^{0}	0.0×10^{0}	0.0×10^{0}	a	0.0×10^{0}	4.7×10^{-5}	110,395	0.0×10^{0}	0.0×10^{0}	0.0×10^{0}	0.0×10^{0}	6.3×10^{1}
5	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	0.0×10^{0}	0.0×10^{0}	0.0×10^{0}	a	0.0×10^{0}	6.9×10^{-5}	110,332	5.4×10^{-11}	0.0×10^{0}	1.7×10^{-10}	2.2×10^{-10}	5.6×10^{1}
6	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	0.0×10^{0}	0.0×10^{0}	a	0.0×10^{0}	9.1×10^{-5}	110,276	1.1×10^{-10}	0.0×10^{0}	3.4×10^{-10}	4.5×10^{-10}	5.1×10^{1}
7	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	0.0×10^{0}	a	0.0×10^{0}	1.1×10^{-4}	110,225	1.6×10^{-10}	0.0×10^{0}	5.1×10^{-10}	6.7×10^{-10}	4.7×10^{1}
8	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	0.0×10^{0}	1.4×10^{-4}	110,177	2.1×10^{-10}	0.0×10^{0}	6.8×10^{-10}	8.9×10^{-10}	4.3×10^{1}
9	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	0.0×10^{0}	1.6×10^{-4}	110,134	2.7×10^{-10}	0.0×10^{0}	8.5×10^{-10}	1.1×10^{-9}	3.7×10^{1}

10	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	0.0×10^{0}	1.8×10^{-4}	110,097	3.9×10^{-10}	0.0×10^{0}	1.2×10^{-9}	1.6×10^{-9}	3.4×10^{1}
11	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	0.0×10^{0}	2.0×10^{-4}	110,063	5.2×10^{-10}	0.0×10^{0}	1.6×10^{-9}	2.2×10^{-9}	3.3×10^{1}
12	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	0.0×10^{0}	2.2×10^{-4}	110,030	6.4×10^{-10}	0.0×10^{0}	2.0×10^{-9}	2.7×10^{-9}	3.9×10^{1}
13	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	0.0×10^{0}	2.5×10^{-4}	109,991	7.7×10^{-10}	0.0×10^{0}	2.4×10^{-9}	3.2×10^{-9}	5.1×10^{1}

TABLE N.3.2 Continued

Age of individual	Test Phase 1	2	3	4	5	6	7	8	Disposal Phase 9	lung 10 - 24	liver 25	bone Summed dose	excess Population	Excess cancer deaths	Excess cancer deaths	Excess cancer deaths	Total cancer deaths	Natural deaths
14	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	0.0×10^{0}	2.7×10^{-4}	109,941	9.0×10^{-10}	0.0×10^{0}	2.8×10^{-9}	3.7×10^{-9}	6.9×10^{1}
15	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	0.0×10^{0}	2.9×10^{-4}	109,871	1.0×10^{-9}	0.0×10^{0}	3.2×10^{-9}	4.2×10^{-9}	9.0×10^{1}
16	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	0.0×10^{0}	3.1×10^{-4}	109,781	1.1×10^{-9}	0.0×10^{0}	3.6×10^{-9}	4.8×10^{-9}	1.1×10^{2}
17	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	0.0×10^{0}	3.3×10^{-4}	109,671	1.3×10^{-9}	0.0×10^{0}	4.0×10^{-9}	5.3×10^{-9}	1.3×10^{2}
18	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	0.0×10^{0}	3.6×10^{-4}	109,542	1.4×10^{-9}	0.0×10^{0}	4.4×10^{-9}	5.8×10^{-9}	1.4×10^{2}
19	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	0.0×10^{0}	3.8×10^{-4}	109,402	1.5×10^{-9}	0.0×10^{0}	4.8×10^{-9}	6.3×10^{-9}	1.5×10^{2}
20	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	0.0×10^{0}	4.0×10^{-4}	109,255	1.6×10^{-9}	5.4×10^{-10}	5.2×10^{-9}	7.4×10^{-9}	1.5×10^{2}
21	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	0.0×10^{0}	4.2×10^{-4}	109,102	1.8×10^{-9}	1.1×10^{-9}	5.6×10^{-9}	8.5×10^{-9}	1.6×10^{2}
22	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	0.0×10^{0}	4.4×10^{-4}	108,942	1.9×10^{-9}	1.6×10^{-9}	6.0×10^{-9}	9.5×10^{-9}	1.7×10^{2}
23	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	0.0×10^{0}	4.7×10^{-4}	108,776	2.0×10^{-9}	2.2×10^{-9}	6.4×10^{-9}	1.1×10^{-8}	1.7×10^{2}
24	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	108,610	2.2×10^{-9}	2.7×10^{-9}	6.8×10^{-9}	1.2×10^{-8}	1.6×10^{2}
25	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	108,446	2.3×10^{-9}	4.0×10^{-9}	7.2×10^{-9}	1.3×10^{-8}	1.6×10^{2}
26	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	108,287	2.4×10^{-9}	5.3×10^{-9}	7.6×10^{-9}	1.5×10^{-8}	1.5×10^{2}
27	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	108,132	2.5×10^{-9}	6.5×10^{-9}	8.0×10^{-9}	1.7×10^{-8}	1.5×10^{2}
28	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	107,978	2.7×10^{-9}	7.8×10^{-9}	8.4×10^{-9}	1.9×10^{-8}	1.6×10^{2}
29	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	107,823	2.8×10^{-9}	9.1×10^{-9}	8.8×10^{-9}	2.1×10^{-8}	1.6×10^{2}
30	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	107,662	2.8×10^{-9}	1.0×10^{-8}	8.8×10^{-9}	2.2×10^{-8}	1.7×10^{2}
31	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	107,495	2.8×10^{-9}	1.2×10^{-8}	8.8×10^{-9}	2.3×10^{-8}	1.8×10^{2}
32	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	107,320	2.8×10^{-9}	1.3×10^{-8}	8.8×10^{-9}	2.4×10^{-8}	1.8×10^{2}

33	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	107,135	2.8×10^{-9}	1.4×10^{-8}	8.8×10^{-9}	2.6×10^{-8}	2.0×10^{2}
34	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	106,939	2.8×10^{-9}	1.5×10^{-8}	8.8×10^{-9}	2.7×10^{-8}	2.1×10^{2}
35	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	106,731	2.8×10^{-9}	1.7×10^{-8}	8.8×10^{-9}	2.8×10^{-8}	2.2×10^{2}
36	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	106,508	2.8×10^{-9}	1.8×10^{-8}	8.8×10^{-9}	3.0×10^{-8}	2.4×10^{2}

TABLE N.3.2 Continued

Age of individual	Test Phase 1	2	3	4	5	6	7	Disposal Phase 8	9	lung 10 - 24	liver 25	bone Summed dose	excess Population	Excess cancer deaths	Excess cancer deaths	Excess cancer deaths	Total cancer deaths	Natural deaths
37	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	106,268	2.8×10^{-9}	1.9×10^{-8}	8.8×10^{-9}	3.1×10^{-8}	2.6×10^{2}
38	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	106,009	2.8×10^{-9}	2.1×10^{-8}	8.8×10^{-9}	3.2×10^{-8}	2.8×10^{2}
39	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	105,727	2.8×10^{-9}	2.2×10^{-8}	8.8×10^{-9}	3.3×10^{-8}	3.1×10^{2}
40	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	105,420	2.8×10^{-9}	2.3×10^{-8}	8.8×10^{-9}	3.5×10^{-8}	3.3×10^{2}
41	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	105,089	2.8×10^{-9}	2.4×10^{-8}	8.8×10^{-9}	3.6×10^{-8}	3.6×10^{2}
42	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	104,731	2.8×10^{-9}	2.6×10^{-8}	8.8×10^{-9}	3.7×10^{-8}	3.9×10^{2}
43	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	104,343	2.8×10^{-9}	2.7×10^{-8}	8.8×10^{-9}	3.8×10^{-8}	4.2×10^{2}
44	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	103,922	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	4.6×10^{2}
45	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	103,461	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	5.0×10^{2}
46	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	102,961	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	5.4×10^{2}
47	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	102,417	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	5.9×10^{2}
48	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	101,829	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	6.4×10^{2}
49	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	101,194	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	6.9×10^{2}
50	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	100,508	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	7.4×10^{2}
51	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	99,766	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	8.0×10^{2}
52	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	98,964	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	8.7×10^{2}
53	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	98,097	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	9.4×10^{2}
54	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	97,158	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	1.0×10^{3}
55	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	96,145	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	1.1×10^{3}

56	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	95,052	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	1.2×10^{3}
57	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	93,878	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	1.3×10^{3}
58	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	92,619	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	1.3×10^{3}
59	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	91,274	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	1.4×10^{3}

TABLE N.3.2 Continued

Age of individual	Test Phase 1	2	3	4	5	6	7	8	Disposal Phase 9	lung 10 - 24	liver 25	bone Summed dose	excess Population	Excess cancer deaths	Excess cancer deaths	Excess cancer deaths	Total cancer deaths	Natural deaths
60	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	89,841	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	1.5×10^{3}
61	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	88,318	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	1.6×10^{3}
62	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	86,703	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	1.7×10^{3}
63	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	84,991	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	1.8×10^{3}
64	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	83,178	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	1.9×10^{3}
65	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	81,260	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	2.0×10^{3}
66	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	79,233	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	2.1×10^{3}
67	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	77,094	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	2.2×10^{3}
68	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	74,845	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	2.4×10^{3}
69	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	72,486	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	2.5×10^{3}
70	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	70,021	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	2.6×10^{3}
71	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	67,458	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	2.7×10^{3}
72	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	64,798	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	2.8×10^{3}
73	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	62,034	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	2.9×10^{3}
74	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	59,153	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	3.0×10^{3}
75	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	56,151	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	3.1×10^{3}
76	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	53,033	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	3.2×10^{3}
77	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	49,819	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	3.3×10^{3}
78	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	46,533	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	3.3×10^{3}

79	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	43,205	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	3.3×10^{3}
80	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	39,860	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	3.3×10^{3}
81	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	36,514	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	3.3×10^{3}
82	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	33,184	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	3.3×10^{3}

TABLE N.3.2 Continued

Age of individual	Test Phase			Disposal Phase						lung	liver	bone	excess	Excess	Excess	Excess	Total	
	1	2	3	4	5	6	7	8	9	10 - 24	25	Summed dose	Population	cancer deaths	cancer deaths	cancer deaths	cancer deaths	Natural deaths
83	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	29,901	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	3.2×10^{3}
84	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	26,703	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	3.1×10^{3}
85	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	23,619	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	3.0×10^{3}
86	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	20,653	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	2.8×10^{3}
87	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	17,813	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	2.7×10^{3}
88	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	15,145	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	2.4×10^{3}
89	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	12,697	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	2.2×10^{3}
90	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	10,502	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	1.9×10^{3}
91	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	8,559	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	1.7×10^{3}
92	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	6,857	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	1.5×10^{3}
93	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	5,392	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	1.2×10^{3}
94	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	4,159	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	1.0×10^{3}
95	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	3,147	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	8.1×10^{2}
96	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	2,337	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	6.3×10^{2}
97	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	1,706	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	4.8×10^{2}
98	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	1,228	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	3.6×10^{2}
99	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	872	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	2.6×10^{2}
100	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	612	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	1.9×10^{2}
101	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	424	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	1.3×10^{2}

102	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	291	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	9.3×10^{1}
103	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	197	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	6.5×10^{1}
104	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	132	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	4.4×10^{1}
105	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	88	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	3.0×10^{1}

TABLE N.3.2 Concluded

		Test Phase				Disposal Phase							Excess	Excess	Excess	Total	
Age									lung	liver	bone	excess					
of											Summed		cancer	cancer	cancer	cancer	Natural
individual	1	2	3	4	5	6	7	8	9	10 - 24	25	dose	Population	deaths	deaths	deaths	deaths	deaths
106	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	58	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	2.0×10^{1}
107	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	38	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	1.3×10^{1}
108	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	25	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	8.8×10^{0}
109	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}	a	2.2×10^{-5}	4.9×10^{-4}	16	2.8×10^{-9}	2.8×10^{-8}	8.8×10^{-9}	4.0×10^{-8}	5.7×10^{0}
												Totals =		2.5×10^{-7}	2.2×10^{-6}	8.1×10^{-7}	3.2×10^{-6}	1.1×10^{5}

Age of individual = Age of individual members of public subject to exposure. Lifetable is extended through 109 years.

Year of operation = 25 year operating lifetime of WIPP showing annual effective dose equivalent for test and disposal phases.

Summed dose = summation of annual effective dose equivalents in a given year.

Population = Shows decrease in total population over time as a result of deaths from all causes. Initial population within 50 miles of WIPP is 112,966.

Natural death rate = Natural death rate for each age group.

Excess lung cancer deaths - Excess lung cancer deaths within remaining population resulting from WIPP-related exposure incurred 5 years ago (latency period for lung cancer).

Excess liver cancer deaths - Excess liver cancer deaths within remaining population resulting from WIPP-related exposure incurred 20 years ago (latency period for liver cancer).

Excess bone cancer deaths - Excess bone cancer deaths within remaining population resulting from WIPP-related exposure incurred 5 years ago (latency period for bone cancer).

Natural deaths = deaths from all natural causes in that year.

Totals = Total of indicated column.

Total Excess deaths = Total of all excess lung, liver, and bone cancer deaths in the population of 112,966.

a Columns for years 10 through 24 not shown for ease of illustration.

APPENDIX O

TEST PLAN SUMMARY

TABLE OF CONTENTS

LIST OF FIGURES

LIST OF TABLES

O.1 **INTRODUCTION**

This appendix describes the underground tests using TRU waste proposed at the WIPP during the Test Phase. This appendix has been prepared in response to comments that requested additional details on the proposed Test Plan, especially as to how the Test Plan relates to the Proposed Action. As noted in Subsection 3.1.1.4, the initial step of the Proposed Action is to conduct a Test Phase of approximately 5 years. The Test Phase has two distinct elements: 1) the Performance Assessment and 2) the Integrated Operations Demonstration. These elements continue to evolve. At this time, the Performance Assessment tests using TRU waste would be composed of laboratory, bin-scale, and alcove tests, and plans on such issues as waste source, type, and volumes for the initial phase of tests are nearing finalization (DOE, 1989a). Waste requirements for the integrated operations demonstration remain uncertain. The DOE, in December 1989, published a detailed phased plan for the Test Phase (DOE, 1989a) that focused on the methods and activities required to demonstrate compliance with the long-term performance standard of 40 CFR 191, Subpart B. In addition, several of the tests planned for the Test Phase would provide data that would be used to support WIPP's demonstration that there would be no migration of hazardous constituents of the waste, as required under the RCRA Land Disposal Restrictions (40 CFR 268). A separate, detailed plan would be developed to describe in detail the Integrated Operations Demonstration. As discussed below, the DOE believes that the analyses in this SEIS bound the potential impacts that would be estimated to arise from any such waste requirements decision.

During the Test Phase, the DOE proposes to transport to and emplace in the WIPP limited quantities of waste; the specific quantities of waste emplaced would be limited to that deemed necessary to achieve the objectives of the Test Phase. For purposes of bounding the potential impacts of the Test Phase in this SEIS, the DOE assumes that up to 10 percent of the volume of TRU waste that could ultimately be permanently emplaced at the WIPP would be emplaced during the Test Phase. The actual amount of waste proposed for the Test Phase would likely be less than that assumed for purposes of analysis in this SEIS. It is also assumed for purposes of bounding the impacts that waste would be shipped from all 10 facilities, although it is now likely that only waste from Rocky Flats Plant and the Idaho National Engineering Laboratory would be used during the initial phases of the proposed Test Phase.

Subsets of the Proposed Action include conducting the Test Phase with bin-scale and/or alcove tests without the Integrated Operations Demonstration and the conduct of these tests with lesser volumes of waste than assumed in the SEIS. The impacts of these subsets would be bounded by the analysis of the Proposed Action in this SEIS.

O.1.1 **EXECUTIVE SUMMARY**

The following has been derived with modification from the Executive Summary of the proposed Test Plan (DOE, 1989a).

O.1.1.1 **Objectives of the WIPP Test Phase**

The purpose of the Test Phase is to further the intent of Congress to demonstrate safe and environmentally acceptable disposal of defense wastes and thereby establish a permanent disposal facility for TRU wastes. The activities that will provide the needed information include experiments, analyses, and operations at the WIPP facility. Although the initial part of the Test Phase is well defined, experimental programs will evolve with increasing understanding of the systems under test. The nature, scope, waste quantities, and timing of experiments and full-scale rooms recommended by various groups remain flexible. The sum total of waste for these tests would initially require approximately 2 percent by volume of the design capacity.

The initial plans for the Test Phase described in this document call for the emplacement of approximately 0.5 percent by volume of the design capacity for Phases 1 and 2 of the alcove tests and Phases 1 and 2 of the bin-scale tests. These bin-scale and alcove tests will support assessment of compliance with the EPA Standard, 40 CFR 191, Subpart B, Sections 13 and 15, and the RCRA Land Disposal Restrictions, 40 CFR 268, Section 6. Additional tests will be defined based on the data acquired during the first two phases of the bin-scale and alcove tests and to incorporate potential engineered alternatives.

In addition, the EPA has requested that the Project monitor the performance of the facility by emplacing waste in 2 full-scale, instrumented, backfilled, sealed rooms after an appropriate demonstration of retrieval using simulated waste. Waste requirements for these 2 full-scale room tests would be approximately 1.5 percent by volume of design capacity. The DOE will conduct a feasibility evaluation to determine the best technical approach, scope, and timing of such monitoring. The DOE will consult the NAS/NAE WIPP Panel, the EPA, the State of New Mexico, and the EEG prior to initiation of such tests.

Also, waste requirements for an Operations Demonstration have not yet been determined. As suggested by several reviewers, the DOE will evaluate the operational experience to be gained through the conduct of all of the test activities and will factor this into future decisions on the scope and timing of an Operations Demonstration. Waste emplaced in the WIPP during the Test Phase would be retrievable until the DOE decides whether the WIPP should become a disposal facility. During the Test Phase, per agreement with the State of New Mexico, the WIPP would meet the applicable requirements of the EPA Standard, 40 CFR Part 191, Subpart A.

The two primary objectives of the Test Phase are to demonstrate the following:

1) Reasonable assurance of compliance of the WIPP disposal system with the long-term disposal standards of the EPA Standard, 40 CFR Part 191, Subpart B, Sections 13 and 15. Compliance of the disposal system would be determined based on a performance assessment, which would include an analysis of the WIPP disposal system design and an evaluation of potential engineered alternatives.

2) The ability of the DOE TRU waste management system to safely and effectively certify, package, transport, and emplace waste at the WIPP in accordance with all applicable regulatory requirements. Acceptability of the waste management system would be evaluated by operations testing and monitoring, both individually and collectively, of the elements of the TRU waste management system. The Operations Demonstration program will be presented in greater detail in a separate document.

These objectives are consistent with the Congressional guidance to demonstrate the safe and environmentally acceptable disposal of TRU waste. In addition, several of the tests planned for the Test Phase would provide data that may also be used to verify the WIPP's demonstration that there would be no migration of hazardous constituents of the waste, as required under the RCRA Land Disposal Restrictions, 40 CFR Part 268, Section 6.

O.1.1.2 **Description of Test Phase Activities**

The objectives would be accomplished by completion of two important programs: a Performance Assessment and an Operations Demonstration. These two programs would provide the necessary information to determine compliance of the disposal system with applicable environmental requirements and to evaluate the safety and effectiveness of the TRU waste management system operations.

Although Subpart B of 40 CFR Part 191 was vacated and remanded to the EPA by the U.S. Court of Appeals for the First Circuit, this Plan (DOE, 1989a) addresses the Standard as first promulgated. The 1987 Second Modification to the Agreement for Consultation and Cooperation between the DOE and the State of New Mexico (1981) commits the WIPP project to continue the performance assessment planning as though the 1985 Standard remained in effect. Compliance plans for the WIPP would be revised as necessary in response to any changes in the Standard.

O.1.1.2.1 Performance Assessment. The performance objective for the WIPP disposal system is to adequately isolate TRU waste from the accessible environment; the performance requirements are reasonable assurance of compliance with the 10,000-year release limits and the 1,000-year dose limits of the EPA Standard, 40 CFR Part 191, Subpart B, Sections 13 and 15. The 10,000-year performance assessment would predict cumulative releases of radionuclides to the accessible environment resulting from both disturbed and undisturbed performance of the disposal system. The 1,000-year assessment would predict annual doses to members of the public in the accessible environment resulting from undisturbed disposal system performance. It would not address the concentration limits established by Subpart B for special sources of groundwater, because no such sources exist at the WIPP. In evaluating compliance with Subpart B, the guidance provided in Appendix B of the Standard would be followed. To ensure that all plausible responses are identified, scenarios would be developed by coupling the individual events and processes that occur. These scenarios would be screened on the basis of probability, consequence, physical reasonableness, and regulatory interest.

Consequence analysis would be used to calculate a performance measure for each of the remaining significant scenarios. The performance measures for the scenarios would be normalized, summed, and reported as a "complementary cumulative distribution function" of release probabilities. Uncertainties in the data would be included in calculations of the performance measure for each scenario. To show that the WIPP can meet the annual dose limits set for 1,000-year performance, the Standard requires that releases from the undisturbed scenarios be analyzed. If any release to the accessible environment is predicted, transport along biological pathways would be modeled, and doses would be estimated. Uncertainties in the data would be included in the dose calculations.

The performance assessment process would be divided into five elements: scenario screening, repository/shaft system behavior and performance modeling, controlled area behavior characteristics and performance modeling, computational system development, and consequence analysis. The combined repository/shaft system and controlled area represent the disposal system that would be assessed.

O.1.1.2.2 Disposal System Characterization Activities. Accurately simulating behavior of the disposal system requires data derived from experiments conducted in the laboratory as well as in the WIPP underground. Such scientific investigations have been conducted since 1975. These studies have resolved many technical issues and have focused attention on aspects still requiring investigation.

There are four major areas of scientific investigation integral to the assessment of disposal system performance. These areas examine the behavior of the disposal room and drift system, the sealing system, structural and fluid-flow behavior of the Salado Formation, and non-Salado hydrology and radionuclide migration. Investigation of these areas involves both laboratory and large-scale underground tests.

Disposal room and drift system activities would examine the interaction of TRU waste and backfill in a waste room. The combined interactions of the source term, waste containers, emplaced backfill and admixtures, brine inflow, and gas generation would be studied through laboratory testing, modeling, and in situ testing. The behavior and performance of possible backfills and additives to be emplaced in access drifts as part of facility decommissioning would also be investigated.

An important parameter of the disposal room and drift system is gas generation. Gaseous products would be generated by microbial and radiolytic decomposition of the TRU waste and corrosion of the waste and waste containers. Gas generation tests with actual TRU waste would be required to characterize the behavior of the disposal system under realistic conditions. These tests would consist of laboratory tests using radioactive and nonradioactive simulated waste, three phases of bin-scale tests with CH TRU waste, and two phases of alcove tests with CH TRU waste. These tests would provide the data needed to evaluate the effects of gas generated by the waste in realistic environments for both the operational (short-term) period and the postoperational (long-term) period. The information collected in these tests would aid the performance assessment in establishing a sufficient level of confidence in the consequence analysis to demonstrate compliance with the EPA Standard. The waste quantities required for these tests represent approximately 0.5 percent by volume of the WIPP disposal area design capacity. In addition to supporting the Performance Assessment Program, the gas generation tests would provide information to be used to verify the RCRA No-Migration Variance Petition's demonstration that the hazardous constituents will not migrate.

Sealing system activities would examine seal design, system behavior, and overall performance evaluation. Seals would be developed for use in drifts to isolate waste panels, in access shafts to isolate the repository from the accessible environment, and in exploratory boreholes. Laboratory and in situ tests would evaluate behavior of potential seal materials such as crushed salt, salt/clay mixtures, and concretes. The effect of hazardous constituents of the waste on seal components would also be tested.

Studies of structural and fluid-flow behavior of the Salado Formation would improve the capability to model fluid flow, hydrologic transport, waste room and drift response, and shaft closure. Healing of fractures in the disturbed zone outside excavations and around seals in shafts and access drifts would be evaluated by modeling. Effects of brine on salt creep would be examined. Laboratory and in situ tests would provide data for improving models of excavation closure, fracture behavior, permeability, and fluid-flow characteristics of the Salado Formation, and brine inflow to excavated rooms. A wide range of studies would address the behavior of penetrations through the Salado Formation, openings at the repository level, and fluid flow to and through these disturbances in the host rock.

The non-Salado hydrology and radionuclide migration activities would address transport of waste to the Rustler Formation and in the Rustler Formation under present and future conditions. Laboratory studies of sorption and retardation in the Rustler Formation would be included, as well as in situ geophysical and hydrological tests from the surface.

In conjunction with the performance assessment, potential engineered alternatives to the current waste disposal system design would be examined. This examination would prepare the DOE to implement any necessary changes to the design in a timely manner as a contingency if performance assessment results have a high degree of uncertainty or are unsatisfactory, or if changes are required to enhance the demonstration of no migration as required under RCRA. Examples of alternatives under consideration are waste processing, changes in the waste disposal room or panel configuration, and passive markers. Engineered alternatives would be screened for relative effectiveness using a design analysis model, and would be screened for feasibility with respect to cost, state of technology, regulatory concerns, and worker exposure. The bin-scale tests, which would use actual radioactive waste underground at the WIPP, would be scheduled in three phases. Engineered alternatives that pass initial screening would be tested in Phase 3, and if identified early enough, in Phases 1 and 2. Alternatives that seem effective and feasible would then be evaluated using the formal performance assessment process to quantify the improvement in disposal system performance.

O.1.1.2.3 Operations Demonstration. The purpose of the Operations Demonstration Program is to demonstrate safe and effective emplacement of certified waste at the WIPP facility. A separate document would be developed to describe the Operations Demonstration following the Secretary of Energy's decision as to the scope and timing of the program. Key elements of the Operations Demonstration would be waste certification and packaging at the generating/storage facilities, the operation of the transportation system, and operation of the WIPP. This demonstration would be integrated to include all elements of the TRU waste management system and would require both CH and RH TRU waste operations. Operational data needs include results from the evaluation of the safety, environmental adequacy, and effectiveness of operations that would certify, transport, and emplace waste at the WIPP. In addition, operational data would be derived from the experience gained during mock demonstrations of bin and drum emplacement and retrieval, and the emplacement of actual TRU waste for bin-scale and alcove experiments underground at the WIPP. The goal of the Operations Demonstration is to provide assurance that operations can be conducted within the limits of all applicable regulatory, technical, industrial, and managerial criteria.

O.1.2 **BACKGROUND**

TRU waste proposed to be disposed of at the WIPP is contained in a mixture of standard 55-gal (208 L) drums and standard waste boxes (SWB). The waste results from nuclear weapons research and production. It consists of laboratory hardware (such as ring stands and other metal structures, and glassware); other laboratory waste (such as Kimwipes, tissues, and towels); protective gloves and clothing; chemicals and inorganic process sludges (generally stabilized with cement); plastic, rubber, and resin; worn-out engineered equipment and tools; and residual organic compounds.

The processes by which gas may be generated include microbial action, corrosion, and radiolysis. In the short-term, these gases are generated predominantly from radiolytic degradation of the waste, and include hydrogen, oxygen (rapidly depleted in most cases), carbon oxides, and low-molecular-weight organic compounds (Zerwekh, 1979; Kosiewicz, et al., 1979; Kosiewicz, 1981; Molecke, 1979). Radiolysis of water and potentially intruding brines could also generate appreciable quantities of hydrogen (and oxygen) in the postoperational and long-term time periods. Microbial degradation mechanisms may be a major concern in both the short- and long-term time periods (Caldwell, et al., 1987; Molecke, 1979). Microbially generated gases include carbon dioxide or methane (Caldwell, et al., 1987; Molecke, 1979), potentially nitrogen from denitrification of nitrates, and hydrogen sulfide from sulfate-reducing bacteria (Brush and Anderson, 1988). Anaerobic (anoxic) metal corrosion in the postoperational and long-term periods could also generate signification quantities of hydrogen (Brush and Anderson, 1988; Molecke, 1979). No radioactive gases would be generated, with the exception of radon ($T_{1/2}$ = 3.8 days) from the decay of transuranic isotopes in the wastes. No radioactive particulates would be released, because the drums would be vented through HEPA filters.

The potential for gas generation in the WIPP and its effect on the long-term performance of the repository is a primary focus of the gas generation test program. WIPP waste emplacement operations for permanent disposal would include placement in rooms and entries within the eight panels; the rooms would be backfilled with an appropriately designed material. After being filled with containers of waste and backfilled, the panels would be sealed from the rest of the underground facility. Any net gas generated by the waste after a panel is sealed must be considered in the long-term performance assessment calculations. The performance of the WIPP disposal system includes not only the room behavior, but also the individual and coupled behavior of the panel seals, access drifts, shaft seals, disturbed zones in the rock around the excavation, and potential transport of radionuclides and hazardous waste through the upper water-bearing units to the accessible environment.

Since the 1980 FEIS, changes in the understanding of factors that affect long-term performance have occurred. These are described below.

- The Salado Formation is probably hydraulically saturated, with very low effective permeability in undisturbed regions. At the time of the FEIS, it was thought to be hydraulically unsaturated, with sufficient gas permeability to dissipate any gases that might be generated by emplaced waste. Thus, the estimated far-field permeability of the Salado Formation has decreased since 1980.

- Current estimates of total gas generation from degradation of emplaced waste and containers are smaller than similar estimates in the FEIS, although uncertainties exist in gas-generation rates, total volumes of gas generated, and the time periods over which gas generation might occur.

- Decreased far-field permeability suggests that the WIPP repository following closure may be dominated by gas at elevated pressure, with little or no free brine within the workings.

- The volumes of gas potentially generated, even in the absence of free brine, may exceed the gas-storage capacity of the waste emplacement rooms at their final state of closure under lithostatic pressure. Gas storage (or relief of pressures) is possible through 1) an expansion of the rooms, after closure, to something less than their original volume; 2) generation of a secondary zone of increased porosity from fracturing around the waste emplacement rooms, or in an incompletely removed disturbed rock zone; 3) migration of gas along open fractures within Marker Bed 139, within or around panel seals, and perhaps within stratigraphic contacts at and near the repository horizon; and 4) following transport from the panels, migration of gas into the shafts and adjacent marker beds.

Thus, laboratory, bin-scale, and alcove tests are proposed to evaluate the effects of gas generation and consumption. These tests are intended to collect, interpret, and refine data necessary for performance assessment. The data resulting from the tests would reduce uncertainty in the performance assessment by verifying assumptions and providing input data on gas generation, gas depletion, and aqueous radiochemistry.

O.1.3 **PROPOSED TESTING**

The laboratory tests would use only simulated waste (nonradioactive) or spiked waste containing a single radionuclide to assess radiolysis and effects of compaction. This appendix addresses only underground tests using actual TRU waste; a brief description of the laboratory tests is presented in the Draft Final Plan for the Waste Isolation Pilot Plant Test Phase: Performance Assessment (DOE, 1989a).

The bin-scale tests would use CH TRU waste specially prepared and modified to provide both repository relevant gas and brine-leachate radiochemical data. (Bins are specially produced, instrumented containers that will hold the equivalent of about 6 drums of CH waste.) The bin-scale tests would confirm and extend similar past and current laboratory test results. Bin-scale tests would provide the results of a scaled verification and evaluation of the impacts of synergistic waste degradation, gas-generation modes, and the effectiveness of backfill additives designed to consume gases ("gas getters"). These tests would include a range of environments: wet, dry, with oxygen, without oxygen, backfilled with gas getters, and backfilled without gas getters.

The alcove tests would use a mix of unmodified (as received) and specially prepared CH TRU waste to obtain information on the operational phase conditions and on the long-term, postoperational phase conditions. Alcove tests are the only experiments planned that can incorporate the impacts of the actual repository environment on the degradation behavior of the waste. The repository impacts are expected to include gases released from the host rock salt (e.g., nitrogen) intermixing with or influencing waste degradation modes; brine influx and consequent humidity effects; long-term waste compaction; and total encapsulation of the waste containers by backfill containing gas getter materials.

The gas generation experiments would not include RH TRU waste. Experiments with CH TRU waste are expected to bound any effects of RH TRU waste, for two reasons. First, the repository would contain 4,000 to 5,000 RH canisters with an average radionuclide content of 37 curies per canister (DOE, 1989c; Table 3.3 in this SEIS). Thus, the maximum RH loading is expected to be 185,000 Ci, only 2 percent of the initial CH loading. Half of the RH radionuclides are short-lived, with half lives of less than 30 years. Second, RH TRU waste would be emplaced in individually drilled and sealed boreholes in the pillars, not in the waste panels proper. Preliminary calculations suggest that these boreholes will creep closed in about 10 years, making waste inaccessible to brine intrusion and degradation (Lappin et al., 1989).

Underground testing would provide data necessary to conduct the performance assessment with a sufficient degree of confidence. Previously, gas generation was not considered a critical factor in the long-term performance of the WIPP. Calculations of gas transport out of the repository into the surrounding Salado Formation (DOE, 1980b; Sandia, 1979) suggested that permeability of the Salado Formation was high enough to allow gas to dissipate without a significant increase in repository pressure, even if the high gas production rates estimated by Molecke (1979) as upper bounds were applicable. Recent, more definitive far-field permeability calculations (Tyler et al., 1988), indicate that permeability of the Salado Formation is low enough that the anticipated high gas production rates may significantly pressurize the repository. Thus, improved understanding of parameters such as gas generation and the repository system (backfill and host rock) behavior have become necessary to establish a realistic range of gas production rates for WIPP. Available estimates of the rates of gas production by CH TRU waste are based on laboratory studies of processes such as radiolysis, microbial activity, corrosion, thermal degradation (Molecke, 1979), and field studies of gases accumulated in the tops of drums (headspace gases) (Clements and Kudera, 1985).

Another investigation of gas generation processes was reported by Brush and Anderson (1988). It was concluded that processes such as drum corrosion, microbial decomposition of cellulosic materials, and reactions between drum corrosion products and microbially-generated gases could affect the gas and water budget of the repository. These processes could consume or produce quantities of water similar to the quantities of brine that are expected to seep into the repository from the Salado Formation.

The Performance Assessment must address the gas and water content of the disposal rooms because these factors could affect long-term performance calculations, especially in the human intrusion scenarios. However, obtaining gas production data representative of the total waste mix is difficult due to the extreme heterogeneity of CH TRU waste, which is the result of the wide variety of generating waste streams. A test program that will be representative would require a large number of experiments and, in large-scale tests, a significant and representative sample of the total waste inventory.

Bin-scale and alcove tests are thus necessary to acquire the data for predictions of long-term gas and water content of WIPP disposal rooms and to assess their impact on repository performance. It is evident, based on all previous investigations, that a proper understanding of gas generation rates and quantities is critical to predicting the behavior and ultimate state of the repository. The TRU waste tests described in this appendix are designed to provide that understanding and help establish an acceptable level of confidence in the prediction of repository performance.

These tests would also help in establishing whether modifications to the design of the disposal system are needed. Rates of gas consumption, normally controlled by radiolysis, microbial degradation, and corrosion, can presumably be increased by including gas getter materials as a backfill component. In addition, anoxic corrosion reactions that generate hydrogen require and consume water in the process. Thus, modifying the disposal room design to minimize brine inflow may limit hydrogen generation.

In addition, the testing program would collect data to support WIPP's RCRA No Migration Variance Petition. Key aspects of the gas testing program related to RCRA compliance are:

- To identify any hazardous components (such as volatile organic compounds) that may be released from waste.

- To gain greater understanding of potential chemical interactions that may occur between various waste types and between waste and repository host rock, brine, and alternative backfill and gas getter materials.

- To observe and report on waste and repository behavior to meet monitoring requirements related to the granting by the EPA of a No-Migration Variance for the WIPP. Air monitoring of all potential releases from the bin and alcove experiments would be conducted throughout the Test Phase.

- To evaluate through a combination of modeling and experimental studies, the expected structural and fluid-flow response of WIPP to internal gas pressurization.

- To evaluate the potential for degradation of the seals and plugs (final design, not temporary inflatable seals) due to exposure to the volatile organic compounds in the waste.

In conjunction with the performance assessment activities, the Project will examine engineered alternatives to the current waste disposal system design. It will prepare the Project to implement any necessary changes to the design in a timely manner as a contingency if performance assessment results have a high degree of uncertainty or are unsatisfactory, or if changes are required to enhance the demonstration of no migration as required under RCRA. Examples of types of alternatives under consideration include waste processing and changes in the storage room or panel configuration. Engineered alternatives will be screened for relative effectiveness using a design analysis model and will be screened for feasibility with respect to cost, state of technology, regulatory concerns, and worker exposure; they will then be tested in laboratory or larger scale experiments where possible. Phase 3 bin-scale tests will incorporate appropriate alternatives, and it is possible that some alternatives will be identified early enough to include them in Phases 1 and 2. Potentially effective and feasible alternatives will be evaluated using the formal performance assessment process to quantify the improvement in disposal system performance.

O.2 **APPROACH**

An assessment of gas issues must consider three elements: gas production, gas consumption, and gas transport.

Gas production is a function of radiolysis and chemical and biological interactions between the waste, waste containers, engineered backfill, brine, and salt. Gas consumption is controlled by the processes of radiolytic and microbial degradation and corrosion. Gas transport depends on the ability of the formation to accept the gas and allow it to disperse. The primary parameter controlling gas transport (in the absence of seal failure) is the Salado Formation gas permeability, which differs for different gases. The gas transport element can be addressed by investigations without waste, but gas production and consumption are largely functions of the waste itself; therefore, radioactive waste is needed in the testing.

The approach of the bin-scale tests is to use test bins that will be large enough to contain a mixture of up to 6 drum volumes of actual CH TRU waste, drum metals, backfill materials, brine, and salt. Sources of gas generation would be introduced into the various environments created in each bin (wet, dry, with oxygen, without oxygen, with gas getters, and without gas getters). For microbial gas generation the sources would be halophilic and nonhalophilic bacteria. Drum and metallic waste materials would provide the corrosion gas source, and the radioactive component of the waste would be the source of radiolytic gas generation. The bin-scale tests would also provide an environment in which various types of gas generation may occur simultaneously. Therefore, these tests would provide a realistic, credible, and synergistic test for the gas generation rates and interactions with backfill and gas getters.

Alcove tests would confirm the results of the laboratory and bin-scale tests. These tests would allow a larger, synergistic test of gas generation, waste compaction impacts, and effectiveness of gas getter material. The tests would consist of waste emplaced in five sealed, atmosphere-controlled test alcoves (each about one-quarter the volume of a waste disposal room). This testing arrangement allows lesser quantities of waste per test alcove, so that more types of test conditions can be accommodated. The waste emplaced in the alcoves would include a typical, representative quantity and mixture of waste types and waste loadings. The volume of waste required is based on both statistical evaluations and practical considerations, and is subject to change based on oversight agency concerns, initial results of the tests, modification of the tests to accommodate treated waste, and other factors.

To accurately measure gas production and consumption, actual radioactive waste must be used. Data needed for the performance assessment models could be obtained from the combination of laboratory tests using small-scale, simulated waste (Brush, 1989; Zerwekh, 1979; Kosiewicz et al., 1979; Kosiewicz 1980, 1981; Caldwell et al., 1987; Molecke, 1979), intermediate, bin-scale tests (Molecke, 1989a), and large, alcove (field) tests (Molecke, 1989b). Resultant data from all of these experimental programs, when coupled with model development, would be used to assess the importance of gas to the performance of the repository. The laboratory-scale tests have been described in more detail by DOE (1989a) and Brush (1989). The strong interrelationship of the bin- and alcove-scale experimental programs, and the perceived benefits and disadvantages of each program are detailed below.

The bin-scale tests may be viewed as larger-scale laboratory experiments, except that they would have the following advantages:

1) They would incorporate actual radioactive TRU waste, and also contain minor chemical components, organic compounds and solvents, and microbial contaminants which could have a very significant impact on overall gas generation and source-term radiochemistry;

2) There would be very few test simulations or required assumptions;

3) All test components, waste forms, contaminants, and possibly engineered alternative materials would be interacting in a synergistic, repository relevant environment, in which various modes of gas generation are occurring simultaneously;

4) The larger scale of the test bins, incorporating about 6 drum-volumes of waste each, would help smooth out the known nonhomogeneities among supposedly similar waste types;

5) The total test matrix could be expanded as necessary, to incorporate new waste forms, backfill and getter materials, and engineered alternatives as they are developed and are ready for testing; and

6) These tests could provide for the rapid collection of data, as compared to the alcove tests, consistent with present Performance Assessment schedules.

The disadvantages of the bin-scale test program are

1) The inability to test at high gas pressures;

2) The inability to fully incorporate all repository environmental effects -- as in the alcove tests;

3) The performance of bin-scale tests at the WIPP is linked to first receipt of waste; and

4) Tests can only examine limited interactions between waste types.

The in situ alcove tests would be conducted under credible, expected-case repository conditions. The major advantages of the alcove tests are

1) Tests would provide "real-world" data, with the fewest simulations or restraints of any of the test programs that could potentially bias the end results;

2) They would be the only tests which actually incorporate the environmental, possibly synergistic effects of the repository itself, i.e., gases and fluids released from the host rock, mine geochemistry and biochemistry, etc., on waste degradation rates and modes;

3) Assessments would determine the gas generation rates for the times of interest, and incorporate how the gases will either be produced or consumed;

4) There would be no significant scaling effects due to the size of the test alcoves; and

5) Many waste forms would be emplaced together in the same test alcove, as would be the case in an operating repository.

The major disadvantages of the alcove tests are

1) The inability to test at high gas pressures because of underground facility safety concerns;

2) The limited number of test alcoves available, resulting in a limitation on test variables and test replicates that can be incorporated;

3) The combination of many waste types within each test alcove makes interpretation of the effects from each type or degradation mechanism almost impossible without comparison to other program data;

4) The large volume of each test alcove, plus the initial trapped gas (air or nitrogen), decreases the analytical sensitivity for gases of interest being produced -- small changes in the quantity of produced gases may be masked;

5) The expected rates of production for individual gases, and changes in those rates, may not be clearly evident for an appreciable period of time -- when compared to gases generated and analyzed in the smaller test bins; and

6) There is no human access to the alcoves after test initiation; potential engineered modifications cannot be added after the test begins.

The added degrees of experimental control, assumed increased sensitivity and selectivity for gas analyses, and the increased number of test conditions for variables to be used in the bin-scale tests -- relative to the alcove tests -- allows the interpretation of obtained data to be simpler and more straightforward than that from the alcove tests. As such, the bin-scale tests provide a technically more satisfying and rapid means of obtaining data.

Collecting test data from any of the test types must not be simply a monitoring or confirmatory activity. Data must be used for both analytical and predictive performance assessment modeling calculations and for comparison with smaller-scale laboratory data on simulated waste. It must be emphasized that it is the combined suite of CH TRU waste test programs (laboratory, bin-scale, and alcove) that are required to provide the full spectrum of information and expertise needed for the performance assessment program. Each test program has its own advantages and disadvantages. None of the three test programs alone can credibly produce the required information.

O.3 **TEST DESCRIPTIONS**

A description of the proposed bin-scale and alcove tests is presented in the following subsections. The test description includes the objectives of the tests, a summary of the tests, and the transportation and emplacement operations. These descriptions are summarized from Brush (1989) and Molecke (1989a and 1989b).

O.3.1 **BIN-SCALE TESTS**

The primary purpose of the bin-scale test program is to provide relevant data and technical support to the WIPP Performance Assessment program for both predictive modeling studies and for the assessment of hazardous component release, and consequent impacts on the WIPP, in relation to EPA concerns and regulations. Specific data to be obtained include the quantities, compositions, and kinetic rate data on gas production and consumption resulting from various CH TRU waste degradation mechanisms.

Similar data on potentially hazardous volatile organic compounds released by the waste and waste-brine leachate or source-term radiochemistry would also be provided. Actual CH TRU waste would be used in these tests.

The degradation and interaction behavior of several representative classifications and types of waste would be tested under aerobic and anaerobic conditions representative of the Disposal Phase and the long-term, postoperational phase of the repository. Tests are intended to allow evaluation of impacts of several types and quantities of intruding brine; impacts on gas production and consumption of waste interactions with salt, container materials, backfill, and gas getter materials; and gas production resulting from synergism among various degradation modes. The tests would be controlled so that safety of personnel is maintained by the use of leak-tight bins, venting through HEPA filters, and close monitoring.

In total, the first two phases of the bin-scale test program would include 116 waste-filled bins and 8 empty test bins (representing background conditions), and a contingency of 8 additional waste-filled bins. This represents a total of 608 drum-volume-equivalents (55-gal, 208 L) of actual CH TRU waste. A later phase of the test program is also defined but cannot be described in adequate detail at this time; all future test additions and contingencies would be included in this "Phase 3." The DOE has formed an Engineering Alternatives Task Force to evaluate potential waste form treatments, facility design modifications, and regulatory compliance approaches that may be evaluated during Phase 3 of the Test Phase. Phase 3 test bins would include any other alternate or processed waste forms, backfill materials, and/or getter materials that may be defined and developed in the future. These materials may be tested in Phase 1 or 2 if they are identified early enough. As indicated in Subsection O.2, the volume of waste to ultimately be used in the Test Phase is subject to modification (a maximum volume of 10 percent of the total waste destined for the WIPP, as analyzed in this SEIS).

O.3.1.1 **Bin-Scale Test Objectives**

The objectives of the bin-scale tests are to

1) Quantify gas quantities, composition, generation, and depletion rates from TRU waste as a function of waste type, time, and interactions with brines and other repository natural and engineered barrier materials with a high degree of control; the experimental conditions would be primarily representative of the long-term, postoperational phase of the repository and the operational phase.

2) Provide a larger-scale evaluation and extension of the laboratory-scale test results, using actual CH TRU waste under repository relevant, expected conditions.

3) Evaluate the synergistic impacts of microbial action, potential saturation, waste compaction, degradation-product contamination, etc., on the gas-generation capacity and radiochemical environment of TRU waste.

4) Incorporate long-term room closure and waste compaction impacts on gas generation by including supercompacted waste.

5) Evaluate effectiveness for minimizing overall gas generation by incorporating getter materials, waste form modifications, and/or engineered alternatives into the CH TRU waste test system.

6) Measure solution leachate radiochemistry and hazardous constituent chemistry from saturated TRU waste interactions as a function of many credible environmental variables.

7) Determine the amount of volatile organic compounds/hazardous gases released from the TRU waste under realistic repository conditions in order to quantify releases of hazardous constituents and adequately address RCRA requirements. Reactive carbon composite filters will not be used because they could affect the behavior of these gases.

8) Provide necessary gas-generation and depletion data and source-term information in direct support of WIPP performance assessment analyses, predictive modeling, and related evaluation, and to justify pertinent assumptions used in modeling.

9) Help establish an acceptable level of confidence in the performance assessment calculations. Help evaluate pertinent modeling assumptions. Help eliminate most "what if" questions and concerns.

O.3.1.2 **Bin-Scale Test Summary**

The bin-scale tests involve testing in multiple large, instrumented metal "bins" with specially prepared TRU waste and appropriate material additives. The "prepared" waste includes up to 6 drum-volume-equivalents of a specific type of actual CH TRU waste with added backfill materials (including salt), metal corrodants (mild steel wire mesh), and brine (to be injected at WIPP). Within each individual test bin there would be a specific type of TRU waste, either noncompacted or compacted. Any plastic bags encapsulating this waste would be "prebreached;" that is, the bags would be sliced or slashed, or the waste itself would be shredded. Special preparation of the waste would occur at the generator/preparer facility. This "prebreaching" permits contact between, and interactions of, the waste with other added components within the bin, and within a time frame shorter than expected in the repository.

Each WIPP test bin, after special waste preparation and filling, would be shipped to WIPP for emplacement and monitoring during the test period. These test bins are specifically designed to fit within a SWB (which is transported within a TRUPACT-2) for transportation to the WIPP and eventual post-test disposal. The test bin alone would not be used for transportation or as a terminal disposal container; the bin is for testing purposes only.

Each bin would function as a nominally independent, isolated and controlled system. All of the test bins for Phases 1 and 2 would be isolated within one underground test room, Room 1 of Panel 1 (Figure O.3.1). In Phase 3, bins may also be placed in Room 2 of Panel 1. The leak-tight bins would have a closely controlled and sealed test environment (internal atmosphere) similar to an isolated, waste-filled repository room. Each bin would be equipped with remote-reading thermocouples, pressure gages, pressure relief valves, gas flow/volume monitors, redundant gas sampling valves, and oxygen-specific detectors. Each test bin and associated instruments would be periodically and closely controlled and monitored by a computerized data acquisition system. Each bin would also be equipped with integral, non-gas-sorbing high efficiency particulate air (HEPA) filters. As such, any gases sampled or released would not contain particulate radioactive contamination.

The bin-scale test matrix includes combinations of the following parameters:

4 representative TRU waste materials classifications (waste types)
2 levels of waste compaction
4 types of backfill material
4 brine moistness conditions.

The four waste types that have been selected for testing are

High-organic/newly generated (HONG) (compacted and noncompacted)
Low-organic/newly generated (LONG) (compacted and noncompacted)
High-organic/old waste (HOOW)
Inorganic processing sludges (PS).

As noted in Subsection O.1, for purposes of bounding impacts it is assumed that CH TRU waste would be shipped from all 10 generator facilities. It is likely, however, that only waste from the Rocky Flats Plant and the Idaho National Engineering Laboratory would be used.

Figure O.3.1 Location of bin-scale test, plan view

The estimated contaminated number of bins per waste type is shown in Table O.3.1. Other representative waste (i.e., high-activity, etc.) may be defined and tested during Phase 3.

TABLE O.3.1 Estimated number of bins

	PHASE 1	PHASE 2	DRUM VOLUMES
High-organic/newly generated (HONG)	24	24	280
Low-organic/newly generated (LONG)	12	6	96
Prepared sludges (PS)	12	14	144
High-organic/old waste (HOOW)	0	24	88
	48	68	608
Empty/gas reference bins	8		
Total	56	68	608

Most high-organic ("soft") and low-organic ("hard," primarily metal and glass) newly generated waste would be compacted at the Rocky Flats Plant. The advantage of using compacted waste in these tests is that the degradation behavior of compacted waste is expected to be very similar to regular (noncompacted) waste that has been crushed/compacted in situ by the long-term closure of the repository rooms. Thus, impacts on gas generation caused by compaction could be realistically evaluated during the course of these tests and factored into the performance assessment calculations.

Other bin-scale test parameters are as follows:

Moistness–
Dry (expected short-term)
Moistened with Salado Formation brine, about 1 percent by volume (expected case within several years)
Saturated with Salado Formation brine, about 10 percent by volume (probable in the long-term)
Saturated with Castile Formation brine (possible in the case of human intrusion).

Backfill (representative of the postoperational phase)—
None
Salt
Salt (70 percent) and bentonite (30 percent)
Salt, bentonite, and gas or radionuclide getter additives
Salt and others (e.g., grouts or others to be defined later).

The atmosphere inside selected test bins would be initially controlled and is expected to be representative of TRU waste in both the short-term post-emplacement period and later periods. HONG waste is expected to create its own anoxic (hydrogen and carbon dioxide) atmosphere primarily by radiolysis and would not require gas flushing. Similarly, no initial gas flushing for the inorganic PS waste would be conducted. The radiolytic depletion or production of oxygen from the PS waste would be quantified along with other evolved gases. The HOOW and LONG bins would be flushed with argon gas until an anoxic (no oxygen present) atmosphere is established. The study of potential anoxic corrosion of metals within the waste, as impacted by other ongoing degradation mechanisms, is one of the significant objectives of this test. All of the waste bins would be injected with (nonradioactive) tracer gases to help facilitate analysis and interpretation of the results.

Gas sample collection would begin as soon as each bin is emplaced, prepared, and sealed. The samples would be analyzed with an on-site gas chromatography-mass spectrometer to determine major and minor gas concentrations and changes in gas compositions as a function of time. The major gases to be analyzed, based on earlier laboratory testing (Molecke, 1979), include hydrogen, carbon dioxide, carbon monoxide, methane, oxygen, water vapor, nitrogen, and injected tracer gases. The minor gases to be potentially measured include: volatile organic compounds (VOC), radon, ammonia, hydrogen sulfide, nitrogen oxides, hydrogen chloride, and any other detectable gases.

Gas quantities and generation rates are significantly impacted by, and would be measured as a function of,

- several representative classifications and types of CH TRU waste;
- time (periodically, over several years);
- impacts of several types and quantities of intruding brines;
- impacts of waste interactions with salt, container metals, and backfill materials;
- aerobic and anaerobic environment conditions representative of the operational-phase and longer-term, postoperational phase of the repository, respectively; and,
- impacts of potential gas getter materials and engineered alternatives, particularly on gas consumption/production.

Waste gas production also includes the synergistic effects of radiolysis, microbial degradation, and corrosion. Different test conditions are tailored so that the effects of individual environmental variables on gas production can be separated from the effects of other variables.

The major gases are primarily generated or consumed by various waste degradation mechanisms occurring within the test bin or those remaining from the initial air atmosphere. The minor gases may arise in two ways: they may be sorbed on or in the waste before it is emplaced in the repository and eventually be volatilized in the repository, or they may be generated in the repository by waste degradation mechanisms. Determining whether VOCs and other hazardous gases are released from TRU waste is an important objective of the test program in order to adequately address compliance with RCRA regulations. Data and analyses would be incorporated into the performance assessment calculations, available on a near-continuous basis.

O.3.1.3 **Bin-Scale Test Phases and Schedule**

This bin-scale test program is planned to take place in several phases. Phase 1 would incorporate test bins where all components can be presently defined. Approximately 48 waste-filled bins of different waste compositions and backfills, including replicates, would be included in Phase 1. There would also be 8 other empty Phase I test bins used for gas baseline-reference purposes. Phase 2 tests would incorporate another 68 waste-containing bins, with more moisture conditions, with gas getter materials, and with the supercompacted high-organic and low-organic waste. Initiation of much of Phase 2 would be dependent on supporting laboratory data (Brush, 1989), particularly as to the composition of gas getters or other backfill material components and on the availability of supercompacted waste. Phase 2 tests would not be anticipated to start sooner than about early FY91. Phase 3 of the test program, including all contingencies and additions, is under evaluation. Future needs for additional test bins and drum-volumes of actual CH TRU waste would be based on upcoming developments, preliminary test results, perceived data needs, and/or possible WIPP project decisions. Details of Phase 3 tests would be incorporated into a future, separate Test Plan addendum (Molecke, 1989a).

Bin-scale testing would continue for a minimum of about 5 years, or until the data acquired are sufficient to provide confidence in the reliability of the information being obtained. At specific periods within the testing program, data would be analyzed and evaluated for input to ongoing performance assessment studies. At appropriate test intervals, data would be fully evaluated and documented in topical reports.

O.3.1.4 **Bin Preparation and Transportation**

Safe transportation of the waste-filled test bins from the generator facility to the WIPP is a critical step in the testing program. The conceptual program design includes the following assumptions with regard to waste packaging and transportation.

Two additions must be made to the preinstrumented bin before the waste is placed in the test bin. First, about a half-drum volume of backfill material would be placed in the bottom of the test bin. Second, about 6 drum-equivalents of bare, unpainted steel (mild steel wire mesh) would be placed along the bottom and side walls of the bin.

The bins would then be remotely filled with waste which would be characterized. Newly generated waste (HONG and LONG) could be loaded directly into the WIPP test bins at the generator facility. Previously packaged (drummed or boxed) waste (HOOW) could be emptied into the bins without the original waste packaging material. Sludges (PS) could be placed directly into the bins.

After the waste is placed in the bins, another half-drum volume of backfill material would be sprinkled on top of the waste materials. The mated bin-lid/liner-lid combination would then be attached to the bin and sealed. The filled bin would be checked for surface contamination and, if necessary, decontaminated following standard procedures of the generator facility.

The waste-filled test bins would be inserted into SWBs at the generator/preparer facility for transportation to the WIPP. The upper gas valves on the test bins (with HEPA filters) would be left in the open, gas-release position during transportation. Therefore, any gases vented would also be filtered through the redundant HEPA filter of the SWB. The SWBs would be loaded into the TRUPACT-II transportation containers and trucked to the WIPP. Removal of the waste bins from the SWBs would occur in the WIPP underground, just prior to emplacement.

O.3.2 **ALCOVE**

The alcove tests are designed to provide data on production, depletion, and composition of gases resulting from the in situ degradation of CH TRU waste. These types of data are needed to support performance assessment of long-term repository behavior and to evaluate long-term generation and release of hazardous constituents. Data on TRU waste degradation rates are needed from testing that includes not only waste that is representative of anticipated waste to be disposed of at WIPP, but also representative of the time from emplacement to the long-term postoperational phase. These tests would enable acquisition of this data in a controlled research mode and allow multiple degradation mechanisms and impacts to be assessed.

O.3.2.1 **Alcove Test Objectives**

The objectives of the alcove tests are to

1) Determine baseline gas quantities, composition, generation, and depletion rates for as-received, representative mixtures of TRU waste in a typical, operational phase repository room environment

2) Determine net gas quantities, composition, generation and depletion rates for a representative range of specially prepared mixtures of actual TRU waste (with and without compaction), backfill materials, gas getters, and intruding brine under representative, postoperational phase repository room conditions

3) Determine the amount of volatile organic compounds/hazardous gases released from the TRU waste under actual repository conditions

4) Provide an in situ test of gas getter effectiveness and demonstration of waste room backfilling procedures

5) Correlate large, alcove results of gas generation and interpretations with those of the laboratory and bin-scale tests of TRU waste degradation and gas production

6) Establish an acceptable level of confidence in the performance assessment calculations that include gas generation and depletion with actual in situ gas measurements and support validation of modeling assumptions.

O.3.2.2 **Alcove Test Summary**

The primary purposes of this WIPP in situ alcove CH TRU waste test program are to provide relevant data and technical support to the WIPP performance assessment program for predictive modeling studies, and to provide in situ data for the assessment of hazardous component release and consequent impacts on the WIPP, in relation to EPA concerns and regulations. Specific data to be obtained include the quantities, compositions, and kinetic rate data on gas production and consumption resulting from various CH TRU waste degradation mechanisms. Similar data on potentially hazardous volatile organic compounds released by the waste would also be provided.

This alcove test program involves, basically, the sampling and analysis of gases released from mixtures of CH TRU waste which have been emplaced within isolated, atmosphere-controlled test alcoves in the underground at the WIPP.

The alcove tests would be conducted in six sealed atmosphere-controlled test alcoves. Four alcoves would be in Panel 1 and the remaining two alcoves would be in Panel 2 (Figure O.3.2). Five of the test alcoves would be filled with waste. The sixth alcove would not have waste in order to collect "background" gases and establish baseline conditions. A test alcove would be about one-quarter the volume and one-third the length of a standard-size WIPP waste room. The test alcoves are smaller than standard rooms to increase the alcove stability with regard to short-term rock deformation and potential fracturing. (The behavior of the disturbed rock zone around full-sized rooms would continue to be examined in other experiments and by modeling during the Test Phase.)

The waste used in the alcove tests would be "as received" (no special processing), compacted, and specially prepared CH TRU waste. All CH TRU test waste would be prepared and packaged at DOE waste generator facilities. "Specially prepared" waste is a waste container that has been filled with waste, backfill and metal corrodants in specified amounts. Waste types, representative of the majority of waste to be isolated at WIPP, include:

High-organic/newly generated (HONG)
Low-organic/newly generated (LONG)
Inorganic processed sludges (PS)
High-organic/old waste (HOOW).

Figure O.3.2 Location of test alcoves, plan view

The approximate quantity of drums per waste type to be used in the alcove tests is based on a preliminary analysis (Batchelder, 1989) of waste currently stored at DOE waste generator facilities and extrapolated to exist through the year 2013. The required in situ alcove CH TRU waste gas data would be acquired in two phases. The alcoves in the Test Phase and the test parameters of each alcove are as follows:

PHASE 1

Test Alcove 1	Test Alcove 2
No waste	As-received, mixed CH TRU waste
Oxic atmosphere	Oxic atmosphere
Dry	Dry
No backfill	No backfill

PHASE 2

Test Alcove 3	Test Alcove 4
Specially prepared and noncompacted waste	Specially prepared, compacted waste
Anoxic atmosphere	Anoxic atmosphere
Moist, 1% brine	Moist, 1% brine
No backfill	No backfill

Test Alcove 5	Test Alcove 6
Specially prepared and noncompacted waste	Specially prepared, compacted waste
Anoxic atmosphere	Anoxic atmosphere
Moist, 1% brine	Moist, 1% brine
Backfill: salt, bentonite, gas getter material	Backfill: salt, bentonite, gas getter material

The alcove tests would be conducted in two phases. Phase 1 includes test alcoves 1 and 2. Test alcove 2 (TA2) would represent expected conditions in the short-term, operational phase of the repository. Test alcove 1 is the gas baseline room. It would provide gas composition data (i.e., trapped atmosphere and gases released from the host rock) necessary for comparison with waste-filled rooms.

Test alcove 2 would contain a representative mixture of about 1,050 drum or drum-volume equivalents of "as-received" CH TRU waste. This waste would be packaged at waste generator facilities into either standard 55-gallon drums or SWBs. Both types of containers would be vented and particulate-filtered. Alcove TA2 would be used to provide data on CH TRU waste gas generation under actual, in situ repository conditions (initial air atmosphere, dry/as-received, with no salt, backfill, or getter material in direct contact with the waste), and is specifically representative of the short-term, operational-phase of the repository. TA2 also provides the initial data for repository time t = 0, necessary for the Phase 2 tests.

Phase 2 of this alcove test program would include four alcoves, and is specifically tailored to be representative of the long-term, postoperational phase of the WIPP repository. Phase 2 test "tailoring" consists of three basic operations: alcove gas atmosphere control, waste special preparation, and brine injection of all waste. It is assumed in WIPP performance assessment that the repository will be anaerobic in the long-term, i.e., anoxic, less than 10 ppm O_2. Therefore, the atmosphere in each alcove would be initially prepared and kept anaerobic. This involves nitrogen gas flushing of each alcove and the continuous use of an oxygen-gettering reactant system. The TRU waste in each Phase 2 test container would be "specially prepared" and/or packaged, as follows. There will be a specific type of TRU waste, either noncompacted or supercompacted, within each test drum or SWB. Any plastic bags encapsulating this waste would be "prebreached," e.g., sliced, slashed, or similarly prepared at the waste generator/storage facility. This operation is beneficial for both testing and transportation (within TRUPACT-II) purposes. The waste would be sandwiched between added layers of backfill materials, 70 wt% WIPP crushed salt/ 30 wt% bentonite clay, and metal corrodant materials (mild steel wire mesh). One or two unbreached plastic bags would enclose all the prebreached waste and other components within one total environment. These all-encompassing plastic bags, at the periphery of the waste container, are used for contamination control during waste packaging operations at the generator facilities.

After emplacement in the WIPP, all Phase 2 TRU waste containers would be specifically moistened with about 1% by volume of Salado brine; this is to be representative of probable long-term brine intrusion. The brine is a mixture of 90% by volume of artificially prepared, and 10% of WIPP-collected Salado brine. Small amounts of brine, 2 liters/drum or 14 liters/SWB, would be injected through brine-injection septa on the top of each container into or onto the waste inside, breaching the all-encompassing plastic bags.

Phase 2 test alcoves TA3 and TA5 would include "specially prepared," noncompacted waste, and TA4 and TA6 would include "specially prepared," supercompacted waste. Alcoves TA5 and TA6 would also include both backfill and gas getters, e.g., reactant, sorptive materials that encapsulate the waste. Backfill and getter materials would be emplaced over and around the waste container stacks in these two test alcoves in a fully retrievable mode. All test waste would be emplaced in such a manner to ensure that post-test retrieval is possible. Waste backfilling would be conducted for gas mitigation test purposes, as well as for operational demonstrations. If other engineering modifications to minimize TRU waste gas generation are available in the appropriate time frame, they could also be added to alcoves TA5 and TA6 for testing of their in situ efficacy.

Four of the six test alcoves would be located along the northern edge of Panel 1; the remaining two alcoves would be located within Panel 2 (Figure O.3.2). Two of the conventionally-mined alcoves (1 and 2) would be 13 ft high by 25 ft wide by 100 ft long. Four of the test alcoves (3, 4, 5, and 6) would be 0.8 ft higher, for a total of 13.8 ft to accommodate compacted backfill on the floor. The available volume to store the TRU waste in each test alcove is about 32,500 ft^3. The alcove would be rock bolted and wire meshed to facilitate waste retrievability and to increase operational safety.

The access drifts would have a slightly smaller cross-sectional area, approximately 13 ft (4 m) wide by 14 ft wide, to facilitate sealing. The access drift would be 170 ft long. The height and width of the access drift are the minimum size possible to accommodate a mining machine and still allow sealing with an appropriately shaped closure seal.

The closure seal would be inflatable and adequate to control pressure of up to 1.5 pounds per square inch (psi) differential pressure without being pushed out. An internal differential pressure of 0.5 psi must be maintained within the test alcove. The test alcove seal would be constructed of materials that have a five-year durability when in contact with salt, gases and liquids expected within the test alcove and that are impermeable to air/oxygen (without generating volatile gases). The seals would contain instrumentation and access ports for the gas sampling system. Dual redundant closure seals would be placed in each access drift, in case one seal leaks while in place.

The test alcoves would contain either 150 seven-packs of drums or standard waste boxes, stacked four across and three high. Test alcoves 3 and 5 would contain a mixture of specially prepared and packaged waste that has not been compacted. Test alcoves with standard, noncompacted waste would contain about 1,050 drums or drum-volume equivalents (210 liter or 55-gallon). Test alcoves 4 and 6 would contain similar waste that has been compacted. Test alcoves with compacted waste would contain about 350 drums of waste. Waste quantities were selected based on statistical evaluations and practical matters.

Each test alcove would be equipped with remote reading thermocouples, pressure gages, and HEPA-filtered gas relief and gas volume monitoring gages. All instruments would be connected to a computerized data acquisition system. No appreciably elevated gas pressures would be present in the test alcoves. A gas recirculation system would be installed to mix gases for sampling; it would include inlet and outlet ducts that penetrate through the inflatable seal with gas sampling ports or septa. All instrumentation and hardware access would be through a sealed access port in the test alcove seal. After the waste, backfill, instruments, hardware and seals are installed, there would be no access to the test alcoves during the tests.

Tracer gases would be added to the test alcoves. Tracer gases would help monitor outflow from the test alcoves to the repository environment, and evaluation of the changes in concentration over time of these tracers would allow compensating corrections to be applied to all other gases being quantified. Separate tracers would be used in each test alcove to monitor any potential leakage from one alcove to another through fractures in the rock.

Gas quantities, compositions, and generation rates can be significantly affected by, and would be measured as a function of, several factors:

- representative classifications and types of CH TRU waste, and mixtures thereof
- time (periodically over several years)
- impacts of intruding, moistening brine
- impacts of waste interactions with salt, container metals, and backfill materials
- aerobic and anaerobic environment conditions, as representative of the operational-phase and longer-term, postoperational-phase of the repository, respectively
- impacts on gas consumption of potential gas getter materials that surround or encapsulate the waste containers.

The waste gas production results also include synergisms between the various waste materials and degradation modes.

Gases periodically collected from each test alcove would be analyzed using a gas chromatograph/mass spectrometer to determine major and minor gas concentrations, and changes in those concentrations as a function of time. This allows rates of generation and/or depletion to be determined. Evaluation of the changes in gas composition would help to determine the relative importance and kinetics of individual degradation mechanisms over time and of the subsequent impacts of degradation by-products on further gas production. The major and minor gases to be analyzed in the alcove tests are the same as those to be analyzed in bin-scale tests (see Subsection 0.3.1.2).

Gas data collection would begin as soon as each test alcove is filled with TRU waste, sealed, and the initial alcove gas atmosphere appropriately prepared. These tests are expected to start providing significant data within months after test emplacement. However, due to the expected slow rate of gas generation and the lack of sensitivity due to the large, masking amount of gas atmosphere initially in the alcoves, it is expected that almost one year will be required before there is an adequate quantity and quality of data for interpretations. WIPP alcove testing would continue for roughly 5 years, or until the data acquired are sufficient to provide confidence in the reliability of the information being obtained. Data would be analyzed and evaluated for input to ongoing performance assessment studies on a near-continuous basis. Data would be fully evaluated and documented in periodic, topical reports.

O.3.2.3 **Alcove Test Phases and Schedule**

Initiation of Phase 2 testing in alcoves TA5 and TA6 depends on supporting laboratory data, (Brush 1989) particularly as to the composition and quantities of gas getters, other backfill material components, or proposed engineered alternatives. These Phase 2 tests would not be expected to start sooner than FY91.

The first four test alcoves, TA1 - TA4, must be mined, equipped, and instrumented prior to the first receipt of waste at the WIPP, expected in FY90. This would be followed by sequential waste loading and filling for each alcove, alcove sealing, appropriate atmosphere preparation, and subsequent gas testing. In order to adequately meet WIPP performance assessment schedule needs, the first four alcoves must be on-line and generating data for about one year prior to the end of FY92 (DOE, 1989a). The remaining two needed test alcoves, TA5 and TA6, would be available for testing at a somewhat later date.

Detailed test planning for these in situ alcove CH TRU waste tests continued through 1989. Procurement activities for necessary test equipment, instruments, associated supplies, and the actual CH TRU waste will proceed through and beyond 1990. Site preparation, including any necessary mining and test installation, also began during FY90 and will continue for one year or more. Initial data acquisition from these tests, e.g., baseline-alcove gas analyses and interpretations, is anticipated to start during FY91. Further descriptions and technical details of these WIPP in situ alcove CH TRU waste tests will be found in the Test Plan (Molecke, 1989b).

O.3.2.4 **Waste Preparation and Transportation**

Safe transportation of the waste-filled test drums and/or standard waste boxes (SWB) from the generator/preparer facility to the WIPP is a critical step in the testing program. The conceptual program design includes the following assumptions with regard to waste packaging and transportation.

The specially prepared waste is placed in a polyethylene-lined drum or SWB. About 0.5 cubic ft of backfill material would be placed in the bottom of the container. A special metal corrodant (a mild steel wire screen or mesh) would be inserted in the container on top of the backfill layer. The container would then be nearly filled with CH TRU waste in prebreached plastic bags. An additional 0.5 cubic ft of backfill would be placed over the waste.

The waste-filled containers would be inserted into the TRUPACT-II at the generator/preparer facility for transportation to the WIPP. Gases released from the drums during transportation would be contained in the TRUPACT-II containers.

O.4 UNDERGROUND TEST OPERATIONAL SAFETY

Concerns regarding operational test safety are addressed in three categories: emplacement, test monitoring, and mine safety. The major safety consideration in the first two categories, emplacement and test monitoring, is personnel exposure to radioactive and/or hazardous constituents. The safety practices during emplacement operations would be similar to those planned for normal operations. During the test monitoring and sampling activities, concerns are focused on personnel exposure during sampling and ventilation due to release of gases from the test bins or rooms. The third category, mine safety considerations, is focused on room stability and waste retrieval.

O.4.1 EMPLACEMENT SAFETY CONCERNS

The emplacement operations for testing are anticipated to be similar to planned WIPP waste handling operations. WIPP waste handling operations would encompass a broad range of activities. The operating functions at the WIPP involve the handling of waste for emplacement, operation of surface facilities, and mining operations. Waste handling consists of shipping container receipt and unloading, waste handling from the surface to the underground facility, emplacement in the underground test area, and maintenance of required records. In support of waste handling activities, the surface and underground facilities would be operated in a manner to ensure operator and public safety in accordance with the "WIPP Operational Safety Requirements Administration Plan" and the "WIPP Radiation Safety Manual" (WEC, 1988a and 1988b).

Unlike plans for normal operations, the emplacement operations, and subsequent sampling and retrieval, would require operators to be in the downstream ventilation air flow on a routine basis. This air flow would be monitored for personnel safety. Use of waste container handling equipment during the Test Phase would be limited to emplacement and retrieval activities. Thus, the potential for an equipment handling accident would be restricted.

The operational safety requirements are based on the as low as reasonably achievable (ALARA) principle. The ALARA techniques applied to the WIPP facilities are based on DOE Order 5480.11, as well as DOE's exposure guide (DOE, 1980a), as appropriate for this first-of-a-kind facility. Radiation exposure to plant personnel is kept ALARA by continued review of operations, training, and the functioning of the Radiation Safety and Emergency Programs Section. The WIPP ALARA program is described in Section 2.0 of the WIPP Radiation Safety Manual (WEC, 1988b). The expected radiation and chemical doses to plant personnel described in Subsections 5.2.3 and 5.2.4 of this SEIS, respectively, are based on testing with 10 percent of the total projected waste and are far below regulatory guidelines. On this basis, the dose estimates in this SEIS can be considered a conservative upper bound.

O.4.2 **TEST OPERATIONAL SAFETY CONCERNS**

Safety concerns during the testing are related to radiological safety, hazardous material safety, and ventilation. In accordance with DOE Order 5480.5 (DOE, 1986), Operational Safety Requirements would be developed as necessary to ensure control of appropriate safety parameters during the Test Phase. Operating procedures would be developed by Westinghouse Electric Corporation, the WIPP operating contractor, in coordination with Sandia National Laboratories, the in situ test coordinator, to guide the testing and monitoring activities. These procedures would be approved by the Westinghouse Radiation Safety and Emergency Programs Section.

Radiological and hazardous material safety operations associated with the in situ testing of actual CH TRU waste would be guided by procedures, which would include specific monitoring and testing requirements. The program would, at the minimum, include the following requirements:

- Gas or other samples taken in the testing program will be monitored for radiation and volatile organic compounds prior to being removed from the test area, a defined Radioactive Materials Area.

- Appropriate personal protective equipment will be worn during sampling and monitoring activities.

- Radiation Work Permits will be prepared for most of the test activities conducted with the actual waste.

- Site Health Physics and Industrial Hygiene personnel will monitor sampling and other test-related activities.

- Westinghouse Radiation Safety and Emergency Programs Section personnel will review sampling and monitoring procedures.

The ventilation system for the WIPP underground facilities is designed to provide a suitable environment for personnel and equipment. It is also designed to remove potential airborne radioactive or hazardous material from the underground area during routine operations or through HEPA filters in the event of an accident. The ventilation system is an exhausting system in which the underground area is maintained below atmospheric pressure. The design airflow quantities are based on standard local, State, and Federal industrial and mining laws and practices. Air quantities supplied to the underground area have been determined to meet or exceed the criteria specified in the Mine Safety and Health Administration code.

All gases released through the pressure relief valves on bins and alcove seals would already have been filtered through a non-gas-sorbing HEPA filter. Therefore, the potential for a radioactive release from within the bins or drums is very small. Released gases are expected to be predominantly nitrogen, with low concentrations of carbon dioxide, carbon monoxide, hydrogen, oxygen, tracers, and possibly methane and other volatile organics. These released gases would be vented to the person-access area or directly to a mine ventilation duct to be carried away by normal mine ventilation. Separate chromatograph/mass spectrometry analyses of gases from the test bins and alcoves would provide a measure of the possible hazard of such gas released in small quantities. If necessary, samples of mine air in the immediate vicinity of the test room person-access areas may also be analyzed for safety assurances.

O.4.3 **MINE SAFETY CONCERNS**

Guaranteeing the retrievability of CH TRU waste emplaced and related operational mine safety are major concerns in the design of the underground testing program. The test areas must remain stable and open during the Test Phase and for several more years to assure retrievability. Concerns about rock spalling, fracturing, and slabbing would be mitigated by rock bolts and wire mesh.

In order to minimize the rock instability uncertainties, the roofs of the test alcoves and rooms would be supported using patterned rock bolting, which has been successfully used for stability in other portions of the underground. The rock bolt system, which was designed and installed in Panel 1, consists of three-fourths inch diameter by ten-foot long mechanically anchored bolts. A similar rock bolting pattern would be implemented in the alcoves. Wire mesh would also be added. The support system has been designed to support the full weight of the immediate roof beam up to the first anhydrite layer in the roof. The pattern is staggered in order to increase bolt hole distance, and, therefore, reduce the potential for fracturing between holes. It is not expected that the bolting will prevent creep of the salt nor stop the fracturing and separating that have been observed in the underground. Rather, the bolting would prevent roof rock from falling, once it has fractured and has become detached. In order to maintain the gas and brine leak-tight integrity of the test room roofs, certain precautions must be taken with regard to rock bolt installation, testing and sealing procedures. Appropriate types of caulking sealant would be injected into the rock bolt holes; degassing and volatilization of the sealant material would be kept to a minimum to limit interference with subsequent gas sampling and analyses.

O.5 **POST-TEST OPERATIONAL SAFETY**

Post-test operational safety concerns focus on three main issues: retrieval of bins, retrieval of drums, and options for disposition of the waste used in the tests. Safety concerns associated with bin retrieval include handling and processing the waste and possible exposure to radiation and hazardous materials. Radiological exposures to the workers and to the public from retrieval operations are discussed in Subsection 5.2.3 of this document. While potential drum handling accident scenarios are not different than during emplacement, the probability of container failure during handling may be higher, particularly for drums from the test alcoves because of the potential for drum corrosion or damage during the test period. In addition, retrieval of waste from back-filled rooms may be more complex resulting in a higher probability of an accident during retrieval operations. However, as discussed in Subsection 5.2.3, special procedures and provisions would be employed to reduce worker exposures in the event that retrieval of the waste is required. Disposition of the waste after the tests is subject to regulatory requirements and available disposal or storage facilities. A Waste Retrieval Plan (DOE, 1989d) is currently being developed to describe the processes, administrative controls and procedures, and organizational responsibilities that would be implemented to ensure safe and effective removal of emplaced TRU waste.

O.5.1 **BIN RETRIEVAL**

At the end of the test period, the bins would still be filled with various combinations of CH TRU waste, backfill, and brine. Gases, potentially with radioactive or hazardous constituents, are also expected to be in the bins. The gases would be purged by flushing through the HEPA filters on the bins. The HEPA filters would remove any radioactive particulates. The gases would be vented through the facility ventilation system. Any free liquids would be removed from the bins. The waste in the bins could be further desiccated by flushing the bins with warm air or injecting sorptive materials. Disposition of the liquid and the waste is discussed in Subsection O.5.2.

Safety precautions during the post-test period would be similar to those taken during the test period (Subsection O.4.2). Gas and liquids removed from the test bins would be monitored for radiation and volatile organic compounds prior to being removed from the test area. During all post-test activities, appropriate personal protective equipment would be worn. Site health physicists and industrial hygienists would monitor post-test-related activities. Radiation work permits would be prepared for the post-test activities conducted with the actual waste. The Radiation Safety and Emergency Programs Section personnel would review sampling and monitoring procedures in use during post-test activities.

O.5.2 **ALCOVE RETRIEVAL**

At the conclusion of the alcove test measurements, five of the alcoves would contain various combinations of waste, backfill, drums, and gas. The injected brine is expected to be predominantly sorbed on the waste matrix materials; very little free liquid is anticipated. If a decision to retrieve waste is made at the end of the Test Phase, a contamination control area would be established in the waste retrieval chambers during waste retrieval operations. Air flow in the control area would be maintained such that workers remain in the upstream flow of the working face of the waste stack. Current plans are to continuously filter area exhausts through a single HEPA filter, reducing the concentration of particulates released to the underground exhaust shaft by a factor of 1,000 before release to the atmosphere.

The gas atmosphere in each alcove would be purged (flushed, or simply released) into the normal mine ventilation system. The plug seals would then be removed. In the test alcoves where backfill was installed, the backfill would be removed, possibly by vacuuming as waste retrieval proceeds.

Safety precautions during the post-test period would be similar to those taken during the test period (Subsection O.4.2). Gas removed from the test bins and alcoves would be monitored for radiation and volatile organic compounds prior to being removed from the test area. During all post-test activities, appropriate personal protective equipment would be worn. Site health physics personnel and industrial hygienists would monitor post-test-related activities. Radiation Work Permits would be prepared for the post-test activities conducted with the actual waste. The Radiation Safety and Emergency Programs Section personnel would review sampling and monitoring procedures in use during post-test activities.

REFERENCES FOR APPENDIX O

Batchelder, H. M., 1989. "Meetings Held at Rocky Flats and INEL," memorandum, June 12, 1989, Westinghouse Waste Isolation Division, Carlsbad, New Mexico.

Brush, L. H., 1989. Test Plan for Laboratory and Modeling Studies of Repository and Radionuclide Chemistry, draft, Sandia National Laboratories, Albuquerque, New Mexico.

Brush, L. H., and D.R. Anderson, 1988. "Potential Effects of Chemical Reactions on WIPP Gas and Water Budgets," Sandia National Laboratories Memorandum, Sandia National Laboratories, Albuquerque, New Mexico.

Caldwell et al. (D. E. Caldwell, M. A. Molecke, R. C. Hallett, E. Martinez, and B. J. Barnhart), 1987. "Rates of CO_2 Production from the Microbial Degradation of Transuranic Wastes Under Simulated Geologic Isolation Conditions," SAND87-7170, Sandia National Laboratories, Albuquerque, New Mexico.

Clements, T. L. Jr., and D. E. Kudera, 1985. TRU Waste Sampling Program: Volume I & II, EGG-WM-6503, Idaho National Engineering Laboratory, Idaho Falls, Idaho.

DOE (U.S. Department of Energy), 1989a. Draft Final Plan for the Waste Isolation Pilot Plant Test Phase: Performance Assessment, DOE/WIPP 89-011, Carlsbad, New Mexico.

DOE (U.S. Department of Energy), 1989b. Final Safety Analysis Report, Waste Isolation Pilot Plant, Carlsbad, New Mexico, draft, DOE/WIPP 88-xxx, Albuquerque, New Mexico.

DOE (U.S. Department of Energy), 1989c. Radionuclide Source Term for the WIPP, DOE/WIPP-88-005, Carlsbad, New Mexico.

DOE (U.S. Department of Energy), 1989d. Waste Retrieval Plan, draft, DOE/WIPP 89-022, Carlsbad, New Mexico.

DOE (U.S. Department of Energy), 1986. "Safety of Nuclear Facilities," DOE Order 5480.5, Washington, D.C.

DOE (U.S. Department of Energy), 1980a. A Guide to Reducing Radiation Exposure to As Low As Reasonably Achievable (ALARA), DOE/EV/1830-T5.

DOE (U.S. Department of Energy), 1980b. Final Environmental Impact Statement: Waste Isolation Pilot Plant, DOE/EIS-0026, Washington, D.C.

Kosiewicz, S. T., 1981. "Gas Generation from Organic Transuranic Wastes, I. Alpha Radiolysis at Atmospheric Pressures," Nuclear Technology, Vol. 54, pp. 92-99.

Kosiewicz, S. T., 1980. "Cellulose Thermally Decomposes at $70^{o}C$," Thermochimica Acta, Vol. 40, pp. 319-322.

Kosiewicz et al. (S. T. Kosiewicz, A. Zerwekh, and B. Barraclough), 1979. "Studies of Transuranic Waste Storage Under Conditions Expected in the Waste Isolation Pilot Plant (WIPP)," Interim Summary Report, October 1, 1977 - June 15, 1979, LA-7931-PR, Los Alamos National Laboratory, Los Alamos, New Mexico.

Lappin et al. (A. R. Lappin, R. L. Hunter, D. P. Garber, and P. B. Davies, eds.), 1989. Systems Analysis, Long-Term Radionuclide Transport, and Dose Assessments, Waste Isolation Pilot Plant (WIPP), Southeastern New Mexico: March 1989, SAND89-0462, Sandia National Laboratories, Albuquerque, New Mexico.

MSHA (Mine Safety and Health Administration), Federal Mining Code for Metal and Nonmetallic Underground Mines, Title 30, Code of Federal Regulations, Part 57.

Molecke, M. A., 1989a. "WIPP Bin-Scale CH TRU Waste Tests," review draft, August 1989, Sandia National Laboratories, Albuquerque, New Mexico.

Molecke, M. A., 1989b. "Test Plan: WIPP In Situ Alcove CH TRU Waste Tests," management review draft, December 1989, Sandia National Laboratories, Albuquerque, New Mexico.

Molecke, M. A., 1979. Gas Generation from Transuranic Waste Degradation: Data Summary and Interpretation, SAND79-1245, Sandia National Laboratories, Albuquerque, New Mexico.

Sandia National Laboratories, 1979. Summary of Research and Development Activities in Support of Waste Acceptance Criteria for WIPP, SAND79-1305, Sandia National Laboratories, Albuquerque, New Mexico.

Tyler, et al. (L. D. Tyler, R. V. Matalucci, M. A. Molecke, D. E. Munson, E. J. Nowack, and J. C. Stormont), 1988. Summary Report for the WIPP Technology Development Program for Isolation of Radioactive Waste, SAND88-0844, Sandia National Laboratories, Albuquerque, New Mexico.

WEC (Westinghouse Electric Corporation), 1988a. WIPP Operational Safety Requirements Administration Plan, WP-04-7, Rev. 4, Waste Isolation Pilot Plant, Carlsbad, New Mexico.

WEC (Westinghouse Electric Corporation), 1988b. WIPP Radiation Safety Manual, WP-12-5, Waste Isolation Pilot Plant, Carlsbad, New Mexico.

Zerwekh, A., 1979. "Gas Generation for Radiolytic Attack of TRU-Contaminated Hydrogenous Waste," LA-7674-MS, Los Alamos National Laboratory, Los Alamos, New Mexico.

APPENDIX P

TRU WASTE RETRIEVAL, HANDLING, AND PROCESSING

TABLE OF CONTENTS

LIST OF FIGURES

LIST OF TABLES

P.1 **INTRODUCTION**

This appendix has been prepared in response to comments requesting that this SEIS evaluate TRU waste retrieval, certification, handling, and processing activities that would be conducted at the various generator/storage facilities for the purpose of preparing the waste for transport to the WIPP. In the 1980 FEIS, Subsection 9.8, and in this SEIS, Subsection 5.2.1, waste retrieval and processing at the Idaho National Engineering Laboratory are discussed. These discussions include: 1) waste characteristics and management methods; 2) the consequences of current operations from routine handling and potential accidents; and 3) the methods used to retrieve, process, and ship waste.

This appendix provides information that describes the current and planned TRU waste retrieval and processing activities at representative DOE generator/storage facilities. Many of these activities would support TRU waste certification and preparation for transport to the WIPP. However, these retrieval and processing activities would be applicable even if the No Action Alternative were implemented. For example, waste containers currently in retrievable storage on asphalt pads and covered with plastic and soil will ultimately have to be retrieved and altered (treated or repackaged) to avoid a release of materials from package degradation. Once this becomes necessary, it would be appropriate to assay the packages to better characterize the contents. Other treatments could be applied at this time as appropriate. Therefore, the processes described herein are not unique to WIPP operations. Appropriate NEPA documentation has been or will be prepared for any proposed modifications to TRU waste management activities of the various DOE facilities. This appendix also provides a description of bin and waste preparation that would occur at the generator/storage facilities prior to the Test Phase.

This appendix draws upon the following documentation:

- Idaho National Engineering Laboratory. A draft Environmental Assessment for the Process Experimental Pilot Plant (PREPP) has been prepared and is undergoing internal review. Other NEPA documentation will be prepared for other retrieval and process facilities as proposed.[1]

- Hanford Reservation. A Final Environmental Impact Statement (DOE/EIS-0113), "Disposal of Hanford Defense High-Level, TRU and Tank Waste" (DOE, 1987a), was published in December 1987 and a Record of Decision was issued on April 4, 1988 (53 FR 12449).

- Los Alamos National Laboratory. A draft Environmental Assessment addressing waste retrieval, processing, and shipment to the WIPP has been prepared and is undergoing internal review.[2]

---

1 Copies of preliminary drafts of documents in internal review are not yet publicly available; descriptive information and environmental consequences are preliminary and subject to change.

2 Copies of preliminary drafts of documents in internal review are not provided.

- Oak Ridge National Laboratory. A draft Environmental Assessment addressing CH TRU waste has been prepared and is undergoing internal review.[2] A similar Environmental Assessment addressing RH waste will be prepared in 1992.

- Savannah River Site. DOE/EA-0315, "Environmental Assessment on Management Activities for Newly Generated TRU Waste, Savannah River Plant" (DOE, 1988a) and a finding of no significant impact covers retrieval, treatment, and packaging for shipment to the WIPP.

- Rocky Flats Plant. DOE/EIS-0064, "Final Environmental Impact Statement: Rocky Flats Plant Site" was published in April, 1980. Also, an Environmental Assessment to consider the potential environmental impacts that may occur from construction and operation of a Supercompactor and Repackaging Facility and a Transuranic Waste Shredder has been prepared and is undergoing internal review.[2]

- WIPP Site. WIPP 89-011, "Draft Plan for the Waste Isolation Pilot Plant Test Phase, Performance Assessment and Operations Demonstration" has been prepared (DOE, 1989).

The DOE believes that the waste retrieval and processing activities described herein are representative of those that likely would occur at other DOE facilities that may eventually transport post-1970 TRU waste to the WIPP. This belief is based on the following:

- The similarity in retrieval and processing approaches at the various facilities and the nature of retrievable storage among facilities.

- The volume of retrievably stored CH TRU waste at the six DOE facilities described constitutes 98 percent of the total retrievably stored inventory (see Table 3.1).

- The magnitude of the consequences presented for the Idaho National Engineering Laboratory, Hanford Reservation, and Savannah River Site.

As noted elsewhere in this SEIS, the DOE will issue another SEIS at the conclusion of the Test Phase; such a SEIS would update the information contained in this Appendix for all 10 DOE facilities and would analyze in detail the system-wide impacts (including those from retrieval, handling, processing, and transportation) of disposal of post-1970 TRU waste in the WIPP.

P.2 **SAVANNAH RIVER SITE**

P.2.1 **RETRIEVAL AND PROCESSING**

TRU waste at the Savannah River Site is in retrievable storage on concrete pads or buried in shallow trenches. It is contained in concrete and steel boxes, concrete culverts, and galvanized steel drums covered with 4 ft of soil or tornado netting (in use since 1985).

The 4-ft soil cover would be removed from the stored waste pads by earth-moving equipment to within 6 to 12 inches of the waste containers. The remaining soil would be removed with the remotely operated, HEPA-filtered soil vacuum. Drums would be removed using a shielded lifting canister. Large steel boxes and concrete culverts would be lifted from the pads and placed directly on a transport trailer for shipment to the TRU Waste Processing Facility building.

Retrieved TRU waste and the newly generated TRU waste requiring processing prior to certification would be processed at a new TRU Waste Processing Facility. A flow diagram for TRU waste processing at the Savannah River Site is depicted in Figure P.2.1. The TRU Waste Processing Facility is scheduled to begin operation in 1995.

Waste containers would be received at the TRU Waste Processing Facility through an airlock into a high bay storage and opening area. The TRU Waste Processing Facility would be used to vent, purge, x-ray, and assay the storage containers; size-reduce the large waste not suitable for shipment; solidify free liquids, resins, and sludge; and repackage the waste to meet WIPP Waste Acceptance Criteria (WAC). (The WAC are described in Appendix A.) Large steel boxes would be opened in this area, and plywood boxes within the large steel boxes would be removed to be processed individually. Culverts would be opened remotely, and drums would be removed and placed into a cell where they would be vented, purged with inert gas, and fitted with a filter vent before going to the verification area. Any gases vented from the drums would pass through the building exhaust system.

In the verification area, drums and boxes would be assayed to determine curie content for inventory control and record purposes. Each container would then be x-rayed to verify compliance with the WAC.

After being x-rayed, containers not conforming to the WAC would pass through an airlock into the remote waste-preparation cell. This cell would have lead-shielded viewing windows and a remote operator's console. All waste-preparation activities would be performed remotely with the aid of a telerobot. This robot would handle several tools, including a plasma arc torch, to size-reduce large objects. The telerobot would remove any objects identified in the x-ray process that do not meet the WAC. An electric worktable would be provided so that the telerobot can work on large, bulky objects.

Figure P.2.1: Savannah River Site TRU Waste Management Plan **((to be pasted-up here))**

Drums and other pieces of equipment may be placed in a shredder for size-reduction. Some smaller equipment would be placed directly in a drum overpack for removal using bagless transfer systems. These systems would significantly reduce the amount of waste generated during the bagout operation by eliminating the need for drum liners and plastic bags. Operations in this cell would be completely remote. A closed circuit television would provide localized viewing of individual equipment operations.

Waste forms segregated as requiring additional processing, such as HEPA filters and respirable fines, would be stabilized or solidified in the TRU Waste Processing Facility to meet the WAC. An in-cell vacuum cleaning system would remove dust and contamination. Drums of processed waste would be removed from the processing area using the bagless transfer system and transported to the shipping area, where they would be prepared for shipment to the Waste Certification Facility. In the Waste Certification Facility, drums would be classified as low-level waste or WIPP-certified TRU waste. Low-level waste would be disposed of onsite. Certified drums of TRU waste would be sent to retrievable storage in the burial ground for eventual shipment to the WIPP.

P.2.2 **CONSEQUENCES**

P.2.2.1 **Routine Operations**

During routine operations at the Savannah River Site, the impact of atmospheric releases from TRU waste activities is negligible. Any releases from the TRU Waste Processing Facility and other activities would be well below applicable State and Federal standards.

Plutonium 238 and 239 would be the major radionuclides released to the atmosphere during normal operations. The annual release to the atmosphere is estimated to be less than 6.7×10^{-5} Ci of Pu-239 and/or Pu-238. The radiological doses to the maximally exposed individual members of the public and the general population at the Savannah River Site boundary, 7 mi from the TRU waste facility, have been calculated using methods described in ICRP Publication 30 (ICRP, 1979) and others. Radiation doses due to normal atmospheric releases are expected to result in a maximum individual dose of 3.5×10^{-4} mrem per year effective dose equivalent. These releases are significantly below the EPA standard of 25 mrem/year to members of the general public from radioactive emissions in 40 CFR Part 191 and 40 CFR Part 61. The collective effective dose equivalent is estimated to be 1.2×10^{-2} person-rem/year. These values are small compared with background whole-body doses of 93 mrem per year to the maximally exposed individual and 5.1×10^{4} person-rem per year to the population within 50 miles of the facility.

Routine TRU waste retrieval and processing operations would result in insignificant amounts of radiation exposure to the operating personnel. Occupational dose estimates for normal operations were based on overall occupational doses experienced at the Savannah River Site. Because the work that would be done in the TRU Waste Processing Facility would involve less potential for radiation exposure than most other Savannah River Site activities, this approach is expected to overestimate occupational radiation exposures. The average occupational dose during TRU waste normal operations was estimated to be 0.22 rem per year, a dose well within the DOE occupational exposure limit of 5 rem per year as stated in DOE Order 5480.11 (DOE, 1988a).

P.2.2.2 **Facility Accidents—Retrieval**

The potential impacts of retrieval are assumed to be similar to those resulting from current operations for burial ground TRU waste management activities. For the purposes of this subsection, the consequences of potential accidents to the onsite population, offsite maximally exposed individual, and offsite population are discussed.

P.2.2.2.1 Natural Phenomena. High winds (including straight winds, hurricanes, and tornadoes) could adversely impact the retrieval operations in the burial ground. TRU waste to be retrieved is stored on concrete pads. A 4-ft layer of soil was mounded over the containers until mid-1985. Since then, waste containers placed on concrete pads are covered with tornado netting. The total number of drums on concrete pads is approximately 4,500, but the drums at greatest risk from high winds are those potentially exposed on the perimeters of the pads, up to 420 drums during retrieval operations. The threshold damage speed for straight winds is estimated to be 100 mph. Winds in excess of 100 mph could cause some drum damage and partial content release. Straight winds of 100-150 mph could result in 10 percent (42) of the exposed drums being ruptured. An estimated 10 percent of the contents of the 42 drums (0.5 Ci/drum) would become airborne since the drums contain a variety of alpha-contaminated solid waste, some of which is not likely to be dispersed. An estimated 1 percent of that released would be respirable. Therefore, this event would result in a release of 2.1×10^{-2} Ci (assumed to be Pu-238). In the extreme case of winds over 150 mph, 20 percent of the perimeter drums would be ruptured, and 4.2×10^{-2} Ci would be released.

Failure of concrete culverts is not assumed to occur in even a 150 mph wind. Hence, drums requiring storage in the culverts would retain their integrity.

The threshold damage speed for tornado winds is estimated to be 113 mph. During tornadoes with wind speeds in excess of 113 mph, drums may become airborne for short distances, causing some to rupture. A windspeed of 113-157 mph is conservatively assumed to rupture 12 percent of the drums on the face of a half-filled pad, approximately 50 drums. A tornado of 158-206 mph would rupture 25 percent of the drums on the perimeter, or 105 drums. Using the same assumptions as for straight winds, the consequences would be 2.5×10^{-2} Ci and 5.3×10^{-2} Ci, respectively. The probabilities of tornadoes occurring at the Savannah River Site with these wind speeds are 4.5×10^{-5}/year and 4.0×10^{-6}/year, respectively.

P.2.2.2.2 Process-Related Accidents. Process-related accidents are the direct result of burial ground operations (e.g., criticality, fires and drum ruptures).

No criticality incidents have ever occurred at the Savannah River Site; however, where fissile materials are present, potential criticality incidents cannot be precluded. A nuclear criticality event would be no worse than an explosion with respect to the dispersal of particulate matter; and in this respect, the offsite consequences would be less severe than for fires. The greatest hazard of a nuclear criticality event would be direct radiation to the operating personnel. However, the overall frequency for a nuclear criticality event is so small that the risk can be ignored when compared to the risks from other abnormal events.

To date, no fires have occurred in any of the Savannah River Site TRU waste storage drums or culverts during operations. However, fire is a serious hazard in the burial ground because of the types of waste. Fires in drums could arise from spontaneous combustion, drum rupture, lightning, vehicle crashes, or aircraft crashes.

The release due to fires would depend upon the quantity of material involved. The pad could hold up to 4,500 drums. The quantity of TRU radionuclides in a 55-gal drum placed on the pad is limited to no more than 0.5 Ci, so the maximum quantity of TRU radionuclides on the uncovered pad would be 2,250 Ci. Although large quantities of radionuclides might be on the pad, few containers would actually be involved in a TRU pad fire. It is assumed that one 55-gal drum would be involved in a TRU pad fire. Previous studies have shown that in the event of fire, only those combustion products less than 10 microns are likely to travel beyond the plant boundary. Waste-producing combustion products smaller than 10 microns represent approximately 1 percent of the total material at risk or 5.0×10^{-3} Ci (0.5 Ci/drum).

If a fire occurred in a culvert, it would have a consequence only while the culvert lid is off to load additional drums. However, this could occur only in the TRU Waste Processing Facility because culverts remain closed during retrieval and transport into the TRU Waste Processing Facility. A culvert fire is assumed to involve only one drum containing an average of 167 Ci of Pu-238; therefore, the release is 1.7 Ci (1 percent of the total material is at risk).

No ruptures have occurred in the history of TRU waste storage at the Savannah River Site. Potential for rupture from internal pressure build-up is present in TRU waste drums containing alpha activity in contact with cellulosic material. If drum rupture occurred from such overpressurization, some radioactive material could be dispersed. As in the case of an internal fire, the drum lid seal would fail, allowing the overpressure to be relieved. Released radioactive material that is airborne and respirable should not exceed 1 percent of the drum contents. Conservatively assuming drum contents to be 0.5 Ci Pu-238, a release to the atmosphere is estimated to be 0.005 Ci Pu-238.

Drum damage can result from corrosion during storage or from mishandling during transport. Mishandling can result in drums being dropped, crushed, punctured, or dented. The release from such accidents would be localized since insufficient energy is available to disperse the radioactive nuclides. However, the potential for operator exposure remains. It is estimated that 1 percent of the contents of the damaged container would be released and 1 percent of the release, or 5.0×10^{-5} Ci Pu-238, would become airborne.

The maximally exposed offsite individual would receive the highest exposure from an accident in the burial ground which results in a fire in a culvert. The effective dose equivalent for this accident was calculated to be 4.4 rem, which is well below the DOE guide of 25 rem for postulated accidental releases for nonreactor nuclear facilities.

The upper-bound latent cancer risk to the total onsite and offsite populations would be about two additional deaths among the total population within 50 mi. This population is expected to experience about 110,000 cancer deaths during the same time frame from unrelated causes. The maximum individual risk off the site would represent less than a 1 percent increase in normal cancer risk. Consequences of all other postulated accidents are so much smaller than this example that they do not require analysis.

Table P.2.1 summarizes the consequences from postulated accidents at the burial ground.

P.2.2.3 **Facility Accidents--TRU Waste Processing Facility**

The following discussion of potential accidents in the TRU Waste Processing Facility is based on the analysis of potential processing accidents at the burial ground.

The categories of abnormal events analyzed are natural phenomena and process-related accidents. An aircraft crash or a criticality accident are not considered credible accidents because of the extremely low frequency of occurrence. The threshold damage speed for straight winds and tornado winds is 100 mph.

The accident in the TRU Waste Processing Facility resulting in the highest exposure to an offsite individual was determined to be a tornado (> 200 mph). The effective dose equivalent was calculated to be 2.0 rem, which is well below the DOE guideline of 25 rem. The upper-bound latent cancer risk to the total onsite and offsite populations would be about two additional deaths among the total population within 50 mi. This population is expected to experience about 110,000 cancer deaths during the same time frame from unrelated ("natural") causes.

Table P.2.2 summarizes the consequences for postulated accidents in the TRU Waste Processing Facility.

TABLE P.2.1 Summary of consequences from postulated accidents in the burial ground[a]

Accident	Curies released	Effective dose equivalent: On-site population (person-rem)	Off-site population (person-rem)	Offsite maximally exposed individual (mrem)
Winds[b]				
100 mph	$2.1x10^{-2}$	$1.6x10^{-1}$	4.4	$6.3x10^{-2}$
> 150 mph	$4.2x10^{-2}$	$2.2x10^{-1}$	6.3	$7.3x10^{-2}$
Tornado				
113-157 mph	$2.5x10^{-2}$	9.3	$1.6x10^{1}$	$1.3x10^{-2}$
158-206 mph	$5.3x10^{-2}$	$2.1x10^{1}$	$3.5x10^{1}$	2.7
Fire				
Drum in culvert	1.7	$9.3x10^{3}$	$2.0x10^{4}$	$4.4x10^{3}$
Drum on pad	$5.0x10^{-3}$	$2.8x10^{1}$	$6.1x10^{1}$	$1.3x10^{1}$
Drum rupture				
Internal pressure	$5.0x10^{-3}$	$2.8x10^{1}$	$6.1x10^{1}$	$1.3x10^{1}$
External pressure	$5.0x10^{-5}$	$2.8x10^{-1}$	$6.1x10^{-1}$	$1.3x10^{-1}$

[a] Estimated from the analysis of potential burial ground accidents reported in DPSTSA-200-10, Supp. 8.

[b] Straight winds.

TABLE P.2.2 Summary of consequences from postulated accidents in the TRU Waste Processing Facility[a]

Accident	Curies released Pu-238	Curies released Pu-239	Effective dose equivalent: On-site population (person-rem)	Effective dose equivalent: Off-site population (person-rem)	Effective dose equivalent: Offsite maximum individual (mrem)
Winds[b]					
100-150 mph	4.3	4.7×10^{-2}	5.1×10^{1}	7.3×10^{2}	1.1×10^{1}
> 150 mph	8.8	9.5×10^{-2}	7.3×10^{1}	1.1×10^{3}	1.8×10^{1}
Tornado					
100-200 mph	5.2	5.7×10^{-2}	1.9×10^{3}	2.8×10^{3}	2.5×10^{2}
> 200 mph	4.4×10^{1}	4.7×10^{-1}	1.5×10^{4}	2.3×10^{4}	2.0×10^{3}
Earthquakes					
0.09-0.2 g	4.3×10^{-2}	5.0×10^{-4}	3.4×10^{2}	4.3×10^{2}	1.1×10^{2}
Vehicle crash	2.2×10^{-2}	2.4×10^{-4}	1.7×10^{2}	2.1×10^{2}	5.5×10^{1}
Fire	8.7×10^{-3}	9.5×10^{-5}	7.3×10^{1}	9.3×10^{1}	2.5×10^{1}
Drum rupture					
Internal pressure	4.3×10^{-3}	4.7×10^{-5}	3.4×10^{1}	4.2×10^{1}	1.1×10^{1}
External pressure	4.3×10^{-5}	4.7×10^{-5}	3.5×10^{-1}	4.3×10^{-1}	1.1×10^{-1}

[a] Estimated from the analysis of potential ETWAF/WCF accidents reported in DPSTSA-200-17, Rev. 1.

[b] Straight winds.

P.3 **HANFORD RESERVATION**

P.3.1 **WASTE CHARACTERISTICS AND CURRENT MANAGEMENT METHODS**

TRU waste generated at the Hanford Reservation since 1970 has been retrievably stored. Most of this waste is contact-handled (CH) waste and is in 55-gal drums, stored as shown in Figure P.3.1. The containers are covered with plywood, plastic-reinforced nylon sheeting, and a 4-ft layer of uncontaminated soil to reduce surface radiation exposure rates. Hot cell remote-handled (RH) waste is stored in caissons such as those illustrated in Figure P.3.2. TRU waste unsuitable for asphalt pad or caisson storage because of size, chemical composition, security requirements, or surface radiation has been packaged in wooden, concrete, or metal boxes, and stored in dry waste trenches since approximately 1973. Each trench is covered with plywood and vinyl plastic and backfilled with dirt (see Figure P.3.3). Newly generated TRU waste is stored in approved storage facilities. These aboveground buildings meet all Federal, State, and local regulations.

P.3.2 **RETRIEVAL**

CH TRU waste in retrievable storage trenches and aboveground buildings is stored free of external contamination and packaged to maintain integrity for a minimum of 20 years. It is packaged so that the waste can be retrieved in an open environment without releasing airborne radioactivity. The soil overburden would be removed using conventional equipment and/or hand digging as required. Once the overburden is removed, the packaged waste would be removed by a forklift or crane.

The current inventory of retrievably stored CH TRU waste would be removed and transferred for certification to a Waste Receiving and Processing Facility (Subsection P.3.3). Waste not directly certifiable would be processed within the Waste Receiving and Processing Facility to produce waste packages that would meet the WAC.

Until about 1994 when the Waste Receiving and Processing Facility is scheduled to begin operation, newly generated TRU waste would be retrievably stored on pads or in buildings. Newly generated TRU waste would be retrieved and, if required, processed in the same manner as the existing retrievable TRU solid waste. After 1994, all CH TRU waste would be processed and packaged to meet the WAC in the facility as it is generated.

Special equipment would be used to recover the RH TRU waste in caissons. In the current retrieval scenario this equipment would not require an entry pit to gain access to the caissons. A recovery building would be positioned over the first caisson row and would contain a remotely operated manipulator and associated equipment. Movement of the building would require roadways. A new entry cut would be made

Figure P.3.1 TRU Waste Asphalt Pad Storage **((TO BE PASTED-UP HERE))**

Figure P.3.2 Typical Caisson for TRU Waste Storage **((TO BE PASTED-UP HERE))**

Figure P.3.3 Typical TRU Waste Burial Trenches **((TO BE PASTED-UP HERE))** into the caisson. The retrieval operations would be controlled remotely from an auxiliary control room. A grappler housing equipped with a telescoping articulated boom would retrieve the caisson waste stored mainly in 1-gal and 5-gal containers. An airlock and conveyor system would be used to transfer the remotely handled cask containing the retrieved caisson waste. This cask would be remotely sealed and decontaminated before placement on a truck. The cask would then be transported to a waste processing facility for conversion to a form suitable for geologic disposal.

A small amount of retrievably stored and newly generated RH TRU waste would also require processing. This waste may be routed to a Special Handling and Packaging Facility designed to process RH TRU waste (not in the Waste Receiving and Processing Facility). This facility would be functionally similar to the Waste Receiving and Processing Facility, and its operations would include specific processes required to meet WAC requirements.

P.3.3 **WASTE RECEIVING AND PROCESSING FACILITY**

The major functions of the Waste Receiving and Processing Facility would include: 1) providing for examination, processing, packaging, and certification of retrievably stored CH TRU waste; and 2) providing for examination and certification of newly generated CH TRU waste for repository disposal.

The Waste Receiving and Processing Facility is conceptually designed to support examination and certification (to the WAC) of CH TRU waste for permanent disposal and is scheduled to be constructed during the 1990s. Processing and packaging capabilities for CH TRU waste in retrievable storage would be provided in the Waste Receiving and Processing Facility.

In estimating product costs, emissions, and volumes of waste, it is projected that 40 percent of all CH TRU waste would be reclassified as low-level waste after the TRU waste content of each pack is measured. The projected 40 percent of waste to be reclassified is based on engineering judgment and historical records.

Waste process systems being considered include waste package inspection, assaying, repackaging, size reduction, compaction, sorting, shredding, and waste immobilization in grout. A conceptual process flow diagram for the Waste Receiving and Processing Facility using a shredding process without incineration is shown in Figure P.3.4.

P.3.3.1. **Waste Process Description**

P.3.3.1.1 Receiving Dock. The first step in the waste package flow would be to offload the waste onto the receiving dock. The dock would be constructed to facilitate offloading of trucks by forklift and possibly by crane. Once offloaded, the waste packages would initially be inspected to determine whether incoming waste meets the WAC or whether further processing is required. For inspection, the receiving dock would be equipped with instruments that measure surface contamination, surface exposure rates, and physical dimensions. Waste packages with exposure rates greater

Figure P.3.4 Waste Receiving and Processing Facility Flow Diagram **((TO BE PASTED-UP HERE))** that 200 mR/hr would be treated or placed in a canister overpack to reduce exposure rates. If it is not cost-effective to place waste packages in a canister overpack, thereby reducing exposure levels below contact handling limits, the waste would be treated as RH TRU and transferred to RH TRU waste storage.

P.3.3.1.2 Size-Reduction Rooms. Waste packages that exceed the WAC physical size requirements would be diverted to the size-reduction room. Here the waste would be repackaged into drums or steel boxes. The size-reduction area in the Waste Receiving and Processing Facility would consist of the following: 1) a waste container opening chamber (box-opening room), 2) a waste-entry air lock, and 3) a size-reduction cell. The box-opening chamber would be equipped with commercially available equipment that would open boxes and sample for internal airborne contamination. The size-reduction cell would be a large stainless steel enclosure equipped with glove ports and viewing windows. Operations would be performed both remotely and manually. The room would be equipped with a positioning table that rotates horizontally and vertically, manipulators and cranes, lightweight dismantling tools, and metal sectioning equipment including nibblers, mechanical saws, abrasive saws, electric saws, and/or plasma torches.

P.3.3.1.3 Nondestructive Assay and Examination Room. Waste packages that meet size, contamination, and exposure criteria would then be routed to the nondestructive assay and examination (NDA/NDE) room to determine 1) TRU waste content, 2) weight, and 3) the presence of noncomplying items such as free liquids or cylinders of compressed gases. Equipment potentially required for the NDA/NDE room includes: scale systems (both in-floor, drive-on scales and smaller scales), neutron- and gamma-scan assayers, x-ray fluoroscopy equipment, ultrasonic and eddy current systems, and visual examination instruments. All certified waste would be routed to the shipping dock for transport to the WIPP. Waste that does not meet WAC would be diverted to the waste-processing room.

P.3.3.1.4 Waste-Processing Room. Noncertifiable drummed waste would be sent through the waste-processing room. The room would include an opening and sorting glovebox and a shredding and immobilizing processor. The opening and sorting glovebox provides for removal of drum lids and for lifting, tilting, and unloading of the drum to a sorting table. The sorting table would be used to separate drum waste into certifiable categories and would be equipped with manipulator arms, glove ports, and tools. This glovebox would also be able to crush empty drums and repackage waste.

The WAC require immobilization of particulates and removal of all but residual quantities of free liquids. (See Appendix A for a description of the WAC.) The shredder and immobilizer would process drum waste to meet these immobilization criteria. The shredding/immobilization process line includes a slow-speed shredder with double rotors to shred 55-gal and 83-gal drums and other similarly sized containers. To minimize contamination and the potential for fire or explosion, the shredding process would be designed to control dust and sparks.

Packages would be opened and sorted when direct shredding of unopened packages is not practical. Examples of nonshreddable waste include pressurized gas cylinders and drums with potentially flammable or explosive contents. Opened drums would be sorted to remove noncertifiable contents for further processing. Uncertifiable waste items would be processed via direct immobilization or other processes as required. Remote operation and maintenance would minimize any damage resulting from contact with unshreddable items.

Processed waste would be transferred to a rotating grout-mixing chamber to be immobilized in grout. Grout formula(s) most suited to immobilize the shredded waste would be determined by experimental testing. To meet functional requirements, the grout must immobilize particulates and free liquids generated as a result of the shredding process. The grouting process would also provide for direct immobilization of various liquid waste streams. Grouting would probably be required to eliminate pyrophoric and/or corrosive characteristics of the waste, but other techniques could be used. The grout/shredded waste mixture would be injected into drums and sent to the drum-curing room for solidification.

P.3.4 **CONSEQUENCES OF WASTE RECEIVING AND PROCESSING OPERATIONS**

P.3.4.1 **Radiological Emissions**

Beginning about 1996, retrievably stored TRU waste would be processed and repackaged during a 5-year period, and the newly generated TRU waste would be processed during a subsequent 8-year period. Due to uncertainties associated with the distribution of the radionuclide inventory, it is conservatively assumed that the entire radionuclide inventory is present in the fraction of waste drums and boxes that are shredded. Projected annual releases from the Waste Receiving and Processing Facility are well below the limits established by the DOE for release in uncontrolled areas.

P.3.4.2 **Radiological Impacts**

Dose commitments to the general population and to the maximally exposed individual are presented in Tables P.3.1 and P.3.2, respectively. The values presented include doses from the processing of retrievably stored and newly generated CH TRU waste. Values are given for exposure periods of 1 year and 70 years. The projected population doses shown in Table P.3.1 are insignificant when compared to the 2.5×10^4 person-rem the offsite population would receive over the same time period from natural background radiation sources.

TABLE P.3.1 Population total-body dose commitments (man-rem) from the processing of retrievably stored and newly generated CH TRU waste at the Waste Receiving and Processing Facility

Pathway	Exposure period: 1 year	Exposure period: 70 years
Air submersion	5.0×10^{-11}	9.0×10^{-10}
Inhalation	1.2×10^{-5}	2.4×10^{-4}
Terrestrial (air paths)	2.0×10^{-7}	4.0×10^{-5}
Total doses	1.2×10^{-5}	2.8×10^{-4}

TABLE P.3.2 Maximum individual total-body dose commitments (rem) from the processing of retrievably stored and newly generated CH TRU waste at the Waste Receiving and Processing Facility

Pathway	Exposure period: 1 year	Exposure period: 70 years
Air submersion	3.7×10^{-16}	7.3×10^{-15}
Inhalation	9.7×10^{-11}	2.1×10^{-9}
Terrestrial (air paths)	3.6×10^{-12}	7.4×10^{-10}
Total doses	1.0×10^{-10}	2.9×10^{-9}

P.4 **LOS ALAMOS NATIONAL LABORATORY**

P.4.1 **RETRIEVAL AND PROCESSING**

CH TRU waste is generated at Los Alamos National Laboratory as a result of plutonium processing and research and development activities and is currently being placed into retrievable storage. Subsequently, this waste would be retrieved and processed by means such as size reduction and incineration, so that it can be certified for shipment and disposal at the WIPP.

RH TRU waste (contaminated with beta- or gamma-emitting nuclides) would also be shipped from Los Alamos National Laboratory to the WIPP. The volume of RH TRU waste is a small percentage (about 0.4 percent) of all retrievably stored TRU waste. Output of RH TRU waste would cease after the existing inventory of experimental materials has been processed and the residues from decommissioning and decontamination have been removed.

Newly generated certified waste would be stored aboveground on an asphalt pad and protected from the elements by plywood and a plastic cover topped with at least 3 feet of soil, much in the same manner in which waste has been retrievably stored since 1971.

The TRU waste facilities at Los Alamos National Laboratory would consist of the following facilities:

- Existing storage facilities
- TRU Waste Size Reduction Facility
- TRU Contaminated Solid Waste Treatment and Development Facility
- TRU Waste Preparation Facility
- TRU Waste Nondestructive Analysis and Examination (NDA-NDE) Facility
- TRU Waste Transportation Facility
- TRU Waste Corrugated Metal Pipe Saw-Processing Facilities
- Other related facilities: liquid waste treatment plant.

The TRU waste facilities would be capable of handling not only newly generated TRU waste but also stored waste and would, either individually or in conjunction with one another, produce certified TRU waste. The Size Reduction Facility and the Treatment and Development Facility are existing online facilities that would be modified. Retrievable storage is located at the Los Alamos National Laboratory's Radioactive Waste Storage Site. Radioactive liquid waste would be treated at the existing liquid waste treatment plant, which would require no modification. The process path for newly generated and stored TRU waste is presented in Figure P.4.1.

Each facility is discussed below.

P.4.2 **WASTE STORAGE SITE**

Since 1971, TRU waste has been packaged and stored in either subsurface trenches or aboveground earth berms at the waste burial site. Two types of packaging have generally been used. Small items have been stored in 55-gal steel drums (sealed and coated with bituminous corrosion protection material), and larger items have been placed in plywood crates (sealed and coated with fiberglass-reinforced polyester). Plywood storage crate sizes vary considerably with a maximum length of approximately 30 ft.

Retrieval work would require heavy earth-moving equipment (e.g., bulldozer, scraper) and a crane capable of about a 60-ft reach to remove the overburden. A small rubber track front-end loader would also be required to assist in the final stages of this operation. As the backfill cover is removed, personnel would probe the remaining cover over the waste with metal rods, measuring the thickness of that cover to ensure that waste packages would not be damaged. This method has proved effective in all prior excavations of this type. Final excavation of the last 4 inches would require manual labor to ensure that no packages are breached. Waste would then be removed using the crane for larger crates and a forklift for smaller crates and drums.

P.4.3 **TRU WASTE SIZE REDUCTION FACILITY**

The Size Reduction Facility has been modified to process large items of TRU waste and to package the cut pieces into certified containers. The facility was designed and built in the late 1970s and was modified in 1984-1985.

The Size Reduction Facility is a production-oriented prototype designed to repackage and reduce the volume of various types of metallic waste (such as gloveboxes, process equipment, and ductwork primarily resulting from decommissioning the old Los Alamos National Laboratory plutonium facility) contaminated with TRU levels greater than 100 nCi/g of material. The Size Reduction Facility enclosure is divided into four modules according to function: airlock, disassembly, cutting, and packaging/bagout.

To process a waste item, the package would be placed in the Size Reduction Facility building and the building would be locked. External packaging would be removed and the item brought into the airlock. The item would pass from the airlock to the disassembly area where attached combustible items would be removed. The item would then be moved into the cutting area where a plasma torch would be used to cut it into smaller pieces for packaging. The pieces would be placed into Department of

Newly
Generated
Waste

Size
Reduction

Treatment
Option
(Incineration)

Preparation

Certified
Waste Holding

Processing

NDA & NDE

Transport

Retrieve from
Uncertified
Storage

TRU Burial

LLW
Burial

TO
DISPOSAL

Retrieve from
Uncertified
Storage

Temporary
Storage
CMP Saw

Figure P.4.1 Los Alamos National Laboratory TRU Waste Process Flow

Transportation (DOT) Type A-approved metal containers in the bagout area, and the containers would be sealed for temporary holding at the waste storage site.

P.4.4 **TRU CONTAMINATED SOLID WASTE TREATMENT AND DEVELOPMENT FACILITY (TDF)**

The TRU Contaminated Solid Waste Treatment and Development Facility is essentially a controlled-air incinerator. The facility was designed and constructed as an option to reduce volume, stabilize chemical composition, and eliminate combustibility of TRU waste. It was built in the mid-1970s and modified in 1984-1985. The Treatment and Development Facility can reduce the volume of combustible waste and/or destroy hazardous or toxic solid and liquid chemical waste. Residues (ash) from the Treatment and Development Facility require additional processing (immobilization) and packaging in other facilities to meet the WAC. Liquid waste from the exhaust gas cleaning system would be piped directly to the Liquid Radioactive Waste Treatment Plant.

The principal component of the incineration process is a dual-chamber, commercially-available unit modified for TRU waste. The Los Alamos National Laboratory modified design could accept a low-density, combustible TRU waste and reduce it by a factor of up to 40:1 by weight and up to 120:1 by volume to produce a chemically-stable, dry product (ash). System components include a feed preparation and introduction train, an off-gas cleanup system, a scrub-solution recycling system, and an ash-removal and packaging station.

The feed preparation and introduction train assays waste and removes any materials not suitable for combustion. Noncombustibles are repackaged and processed as appropriate. The off-gas cleanup system removes particulates and acid gases from effluents and conditions the gas stream for passage through high efficiency particulate air (HEPA) filters before discharge. The scrub-solution recycling system supplies liquids at required pressures to the off-gas system and processes these liquids for recycling or discharge to the Liquid Radioactive Waste Treatment Plant.

P.4.5 **TRU WASTE PREPARATION FACILITY**

The TRU Waste Preparation Facility is a tension-support, polyester-fabric-covered shelter. The initial phase of retrieval operations on stored waste began in 1985 with retrieval of the plywood crates of decommissioned equipment. Retrieved waste drums would also be processed at the Waste Preparation Facility.

In the process of retrieving and certifying TRU waste materials, the Waste Preparation Facility would provide dedicated space for three functionally related operations:

- Cleaning: After retrieval, TRU waste drums and storage boxes would be cleaned, with excess soil removed from plywood storage boxes and excess soil and bituminous corrosion protection coatings removed from steel storage drums.

- Inspection: Steel storage drums would be examined and evaluated for structural integrity and the presence of internal or hidden corrosion using ultrasonic equipment and visual inspection; unacceptable containers would be overpacked as required for onsite transport. Storage boxes would be examined for structural integrity and trapped moisture; they would be drained if necessary and repaired as required for onsite transport.

- Staging: Waste drums and boxes would be staged for transport to the next step in the certification and shipping process.

Experience to date has indicated that high-pressure steam and hot water are most effective for the types of cleaning required in the Waste Preparation Facility. A commercial-type portable steam generator unit would be used. Operations at the TRU Waste Preparation Facility may necessitate periodic decontamination (washdown) of a portion of the interior of the facility (i.e., where the cleaning operation would be performed) and collection and processing of internal drainage.

All internal drainage and effluents emanating from the facility would be considered potentially contaminated and held for further processing. Consequently, these liquids would be collected in a storage tank for sampling and analyzing before periodic transfer to the liquid waste treatment facility. In addition, residues from removal of the bituminous corrosion protection coating on steel storage drums would be removed from the drainage system, processed as a potentially contaminated low-level waste material, and buried at the storage site.

P.4.6 TRU WASTE NONDESTRUCTIVE EXAMINATION AND ANALYSIS (NDE-NDA) FACILITY

Retrieved packages (drums) would be examined in the TRU Waste NDA-NDE Facility and analyzed to validate the nature of the waste matrix and the identity and level of radioactive elements contained in the waste. Where additional processing of the waste is not required, this operation would provide the basis for directly certifying a large portion of stored waste as meeting the WAC.

Drums of TRU waste would be delivered to the NDA-NDE Facility by truck from staging in the Waste Preparation Facility. Following offloading onto individual carts, the drums would be subject to nondestructive analysis (NDA) and examination (NDE) using an active-passive drum assay system and a real time x-ray radiography system. Drums meeting the WAC would be certified and transferred to the adjacent transportation facility for transport to the WIPP. Drums intended for additional processing would be transferred to the appropriate facility. A small fraction of newly generated waste drums may be reviewed as a quality assurance check on the certification process. Only metal drums would be examined in this facility. Steel boxes containing TRU waste packed at the waste size-reduction facility and sectioned pipe from the Corrugated Metal Pipe Saw-Processing Facility would undergo examination and analysis by a mobile assay system.

P.4.7 **TRU WASTE TRANSPORTATION FACILITY**

The Transportation Facility is constructed as a single building with the NDA-NDE Facility. These facilities share a common wall with the Corrugated Metal Pipe Saw-Processing Facility.

The Transportation Facility is a standard design warehouse where certified waste packages would be loaded into TRUPACT-IIs. A semitrailer and tractor would be brought inside the Transportation Facility to load the waste containers. A gantry crane would assist in transferring the waste into the TRUPACT-II. The sealed TRUPACT-IIs would then be inspected and tested prior to shipment.

P.4.8 **TRU CORRUGATED METAL PIPE (CMP) SAW-PROCESSING FACILITIES**

It is proposed that the Corrugated Metal Pipe Saw-Processing Facility be constructed adjacent to the NDA-NDE Facility-Transportation Facility. Though sharing a common wall, it would be independent with separate support systems.

The facility would be initially constructed as the Corrugated Metal Pipe Saw-Processing Facility. The initial operation would be to cut 158 corrugated metal pipes into sections to be packaged in accordance with the WAC. To be certified, a cut corrugated metal pipe section must be 4 ft or less in length to fit into steel boxes that are within the WAC weight limit (6,000 lbs). After the corrugated metal pipes have been processed (approximately 1 year) the facility would be decommissioned, decontaminated, and refitted as the TRU Waste Processing Facility. Operations of the TRU Waste Processing Facility would begin in 1993. It would have the capability of handling retrieved drums of plutonium processing waste and placing them in a special glovebox line for certification through sorting, shredding, fixation and immobilization, or repackaging. Waste such as HEPA filters, soils, and others identified as needing immobilization would also be processed in this facility.

The corrugated metal pipes measure 2.5 ft in diameter by 20 ft in length and weigh 12,000 to 15,000 lbs. They contain a TRU solidified cement paste from the treatment of Pu- and Am-contaminated aqueous waste. Corrugated metal pipes are plugged with uncontaminated concrete. All of the pipes were stored vertically in a 22-ft deep pit that was backfilled with 2 to 3 ft of tuff. In 1984, the TRU corrugated metal pipes were retrieved, decontaminated, and transported to the waste storage site. They later would be transported to the Corrugated Metal Pipe Saw-Processing Facility for processing.

At the Corrugated Metal Pipe Saw-Processing Facility, the pipes would be offloaded, stacked on skids, and covered with plastic sheets or canvas tarps during retrievable storage. During processing, the pipes would be loaded onto a trolley car by gantry crane and taken from the retrievable storage holding area to a staging area inside the facility. Here, any protective plastic film would be removed and the pipes x-rayed by the mobile assay system to locate large metallic objects such as electric motors, which could impair the cutting operation. The mobile x-ray unit would be a high-intensity source and would be designed with proper shielding to prevent adverse radiation exposures to personnel and/or the environment. Following x-ray, the pipes would be moved into a cutting area (a large, semi-hardened, HEPA-ventilated glovebox) for sawing or sectioning. A wet-cutting operation would be used to contain radioactive contaminants released in the cutting process. The process area would have curbing and a liquid waste collection system. Solids from the cutting operation would be collected in a sump in the liquid drain system where they can be removed, packaged, and immobilized in a cementation process. TRU liquid waste would be immobilized in a cementing operation at the processing facility.

After cutting, the sectioned pipes would be moved to a packaging area. Two 4-ft sections of pipe would be placed in a steel box using remotely operated grappling hooks similar to log-handling equipment. The box lid would be sealed by welding. The sealed boxes would be held in the packaging area or transported back to the storage site until space is available in the transportation operation. When space becomes available, they would be moved to the Transportation Facility and loaded into TRUPACT-IIs for shipment to the WIPP.

Upon the completion of the corrugated metal pipe processing, the facility would be stripped out and set up for other processing operations. The drum processing operations at the converted facility are scheduled to begin in early 1993 and continue through 1997.

Processing would involve opening drums and inspecting, sorting, shredding, and cement-fixing the contained TRU waste. Drums of TRU waste (generally 55 gal) that are known or suspected of requiring immobilization treatment (e.g., liquid wastes) would be brought to the Processing Facility from the Waste Preparation and NDA-NDE Facilities. Drums would be opened in a special glovebox line, and the contents removed and sorted. Combustibles would be taken to the TDF for incineration. Some noncombustibles may be certifiable without processing and others would be shredded and subsequently immobilized in a cement mix inside 55-gal metal containers to meet the WAC. The containers would be held until space is available in the Transportation Facility to prepare them for transport to the WIPP.

P.5 OAK RIDGE NATIONAL LABORATORY

TRU waste is generated in the main Oak Ridge National Laboratories complex, primarily in the Isotopes Area and the Radiochemical Engineering Development Center. Newly generated CH TRU waste is packaged in stainless steel drums at the point of generation and is transported within the Oak Ridge National Laboratory site boundary to the TRU waste storage area.

Following inspection for structural integrity and radiation surveys, the stored CH TRU waste containers would be removed from this area, using normal material-handling methods (crane, forklift, other mechanical equipment). From the staging or interim storage area, retrievably stored waste, along with newly generated CH waste, would be moved to the Waste Examination Assay Facility. Fig. P.5.1 provides a diagram of Oak Ridge National Laboratory CH TRU waste management activities. Here the individual containers of waste are nondestructively examined and assayed to determine whether they meet the WIPP WAC.

It is estimated that about 50 percent of the stored CH TRU waste and about 10 percent of the newly generated CH TRU waste would not meet the WAC as is and, therefore, would be repackaged.

The material that causes a drum to fail certification (generally free liquids or compressed gases) would be removed and disposed of in an appropriate manner.

Fine particle materials, in quantities greater than the WAC allow, would be immobilized and repackaged for shipment to the WIPP. Then the drum would be repackaged, sealed, and returned to the assay facility for certification. Transportation of materials between the repackaging facility and the assay facility would be entirely within the Oak Ridge National Laboratory site boundaries. Retrievable storage would be required for waste awaiting either repackaging or shipment to the WIPP, following certification. This retrievable storage would be provided in the existing retrievable storage facilities.

Figure P.5.1 Simplified diagram of Oak Ridge National Laboratory's contact-handled transuranic waste management activities

P.6 **IDAHO NATIONAL ENGINEERING LABORATORY**

P.6.1 **WASTE RETRIEVAL AND PROCESSING**

About 61 percent of the pad-stored defense TRU waste in the United States is located at the Radioactive Waste Management Complex (RWMC) of the Idaho National Engineering Laboratory. Subsection 9.8 of the WIPP FEIS analyzed impacts associated with retrieving, processing, and handling TRU waste at the RWMC. The following subsection updates the FEIS discussion by analyzing the environmental impacts of current TRU operations in Idaho and conceptually describing options under consideration for future processing facilities that would remove TRU waste from retrievable storage and prepare it for shipment to the WIPP.

P.6.1.1 **Waste Characteristics and Current Management Methods**

Since 1970, CH TRU waste received at the Radioactive Waste Management Complex has been stored at the 56-acre Transuranic Storage Area (TSA), a controlled area surrounded by a security fence. The waste is stored on three asphalt pads known as TSA-1, TSA-2, and TSA-R and in two covered enclosures. Approximately 2.3 million cubic feet of TRU waste is currently stored at the TSA.[3]

Solid TRU waste has been received from the DOE facilities in government-owned ATMX railcars or on commercial truck trailers in Type B shipping containers. The ATMX shipments were made under the authority of a special permit issued by the Department of Transportation (DOT Exemption 5948). The waste is contained in 4 x 4 x 7 ft metal boxes with welded lids, 55-gal steel drums with polyethylene liners, and 4 x 5 x 6 ft steel bins. (Earlier, some of the waste placed on the TSA was stored in containers of nonstandard sizes.) The containers are intended to be retrievable and contamination free for at least 20 years.

In the past, the drums and boxes were stacked on the TSA pads with boxes around the perimeter and drums in the center. The drums were stacked vertically in layers, with a sheet of 1/2-inch plywood separating each layer. When the stack reached a height of approximately 16 feet, a cover consisting of 5/8-inch plywood, nylon-reinforced polyvinyl sheeting, and 3 feet of soil was emplaced.

Precertified waste (i.e., in compliance with the WIPP WAC) has been received from the generators and is stored in a covered enclosure.
Other current TRU waste operations at the RWMC include the retrieval of drummed waste that has been stored in a covered enclosure located on the TSA-2 pad, and certification of that waste for compliance with the WIPP WAC and appropriate transportation requirements.

---

[3] Prior to 1982, TRU waste was defined as having a concentration of alpha-emitting radionuclides greater than 10 nCi/g TRU. In 1982, the definition was changed to include only that waste with TRU concentrations greater than 100 nCi/g. As a result, about 1/2 of the 2.3 million ft^3 of waste stored at the RWMC is expected to be reclassified as low-level waste, and is not proposed to be shipped to the WIPP.

This certification takes place in the Stored Waste Examination Pilot Plant (SWEPP) that provides nondestructive examination and assay capabilities to examine TRU waste. The facility contains a real-time x-ray radiography (RTR) system to examine the contents of both boxes and drums, an assay system to determine fissile and transuranic content, and a container integrity system to assure the waste drums meet DOT metal thickness requirements for Type A containers. In addition, the facility provides capabilities to puncture a drum lid (using a sparkless tool) and install a carbon composite filter to vent any radiolytic-produced gas and provide for pressure equilibrium.

All drums retrieved are vented and examined at this facility. Retrieved waste boxes are also examined using the RTR and the box assay system. Those waste packages that meet the WIPP WAC and transportation requirements are so labeled and stored. Those waste packages that do not meet the WIPP WAC would be further processed and repackaged before being shipped to the WIPP.

More complete descriptions of the Idaho National Engineering Laboratory, the RWMC, the TRU waste storage and examination facility, and the TRU waste stored on the TSA pads can be found in the Safety Analysis for the Radioactive Waste Management Complex at the Idaho National Engineering Laboratory (DOE, 1986).

P.6.1.2 **Environmental Effects of Current Operations**

The radiological effects associated with retrieving, examining, venting, and storing TRU waste are presented below. These impacts are discussed for both workers and the general population as a result of normal operations and releases due to potential accidents and violent natural phenomena.

Routine Operations. Measurable exposure to the public or adverse effects on the surrounding environment would not be expected from the extremely small airborne releases experienced during routine operations involving TRU waste at the RWMC. No liquid effluents are expected during routine operations. Releases during normal operations are discussed in annual DOE environmental monitoring reports for the Idaho National Engineering Laboratory (DOE, 1987a). In keeping with the ALARA (as low as reasonably achievable) philosophy, the radiological exposures to workers during normal operations are limited by monitoring accumulated personnel dose equivalents and by job preplanning. The maximum radiation exposure on external waste container surfaces is restricted to less than 200 mR/hr. Annual dose equivalents to RWMC personnel including operators, health physics technicians, and supervisors for all RWMC activities, including TRU waste operations, vary from a maximum of 306 mrem to less than 20 mrem. This is well below the established DOE occupational exposure limit of 5 rem per year (DOE, 1988a).

Accident Conditions. Safety documentation prepared for the current operations of the RWMC complex, which includes all TRU operations, evaluates the dose commitments and risks associated with potential operational accidents (e.g., fires, explosions, dropped containers), as well as those associated with potential natural disasters (e.g., earthquakes, volcanoes, lightning) (DOE, 1986). The projected consequences and risks of the dominant accident scenarios for the general public and workers are summarized in Tables P.6.1 and P.6.2, respectively.

The maximum exposure to an individual member of the public is shown in Table P.6.1

to be 2 x 10^{-2} rem committed whole-body dose equivalent. This exposure is associated with the occurrence of a tornado with 280 mile per hour winds, which has an extremely low probability of occurrence at the Idaho National Engineering Laboratory. The highest population exposure is also associated with the tornado and results in a collective dose equivalent of 1 person-rem. The excess risk to the total exposed population would be 2.8 x 10^{-4} excess cancer fatalities based on a multiplier of 2.8 x 10^{-4} latent cancer fatalities/person-rem.

Table P.6.2 indicates that the highest exposure to the maximally exposed worker is 0.7 rem, resulting from a fire in the air support weather shield. The risks of excess cancer to both the workers and average members of the public are presented in Table P.6.3.

P.6.1.3 **Methods for Retrieving and Handling Waste**

Several operations would be involved in removing the waste and preparing it for shipment to the WIPP: retrieving waste from earthen-covered cells and potential processing and packaging of the waste to meet current WIPP WAC and transportation criteria. The FEIS evaluated several options for each operation.

Three methods of retrieving waste containers were considered: 1) manual handling by the operators; 2) handling by means of operator-controlled equipment; and 3) handling by means of remotely controlled equipment. A combination of the first two methods is currently being performed for retrieval of drummed waste located at the TSA-2 pad and would likely be used for the remaining post-1970 TRU waste.

Four confinement methods for waste retrieval were considered: 1) open-air retrieval (no confinement); 2) the use of an inflatable fabric shield to protect against the weather; 3) the use of a movable, solid-frame structure operating at ambient pressure; and 4) the use of a movable or nonmovable, solid-frame structure operating at subatmospheric pressure. The last method is the only one that provides positive control against the possible release of contamination.

TABLE P.6.3 Excess cancer risks due to accidents associated with RWMC/SWEPP operations with TRU stored waste

	Excess cancer risk[a,b,c]		
Event	Maximally exposed individual	Average member of population[d]	Maximally exposed worker[e]
Tornado	6×10^{-6}	2×10^{-9}	nc[f]
Earthquake	6×10^{-11}	7×10^{-13}	3×10^{-5}
Fire in ASWS/CS	3×10^{-10}	7×10^{-12}	2×10^{-4}
Breached container	6×10^{-12}	7×10^{-14}	3×10^{-6}
Explosion	6×10^{-7}	4×10^{-13}	6×10^{-7}

[a] Health risks are expressed as the probability of an individual contracting a fatal cancer during his/her lifetime as a result of RWMC/SWEPP related activities.

[b] Risk of contracting fatal cancer: 2.8×10^{-4} fatalities/person-rem (BEIR, 1980).

[c] Health effects risk estimates for genetic effects would be somewhat lower than the numbers presented in the table for cancer fatalities--by a factor of 0.918.

[d] Risk to an average member of the population is the product of the collective population exposure (Table P.6.1) by 2.8×10^{-4} fatalities/person-rem divided by an estimated population of 129,000.

[e] Risk based on exposure within the facility (Table P.6.2).

[f] Not calculated.

Four potential processing options were also considered in the FEIS: 1) shipping as is, 2) overpacking, 3) repackaging only, and 4) treatment and packaging. A slagging pyrolysis incineration (SPI) process was proposed for waste treatment and was analyzed in detail in the FEIS. Incineration was the selected processing technology because it was anticipated that free liquid and combustible limitations in the WIPP WAC would make some of the stored waste unacceptable. Waste feed to the SPI was to be blended with glassforming compounds (soil) so the noncombustible ash would be melted at the incineration temperature and form a glass-like slag with low leachability. The molten slag was to be packaged in steel drums. Since 1980, this process was evaluated on an experimental basis and was proven inadequate for development for reliable treatment of stored TRU waste (Tait, 1983). No further DOE development of the process has occurred.

The following subsections discuss conceptual operations of facilities that may be proposed for the retrieval and processing/packaging of TRU waste at the Idaho National Engineering Laboratory. At such time that specific facilities are proposed, the appropriate NEPA documentation will be prepared for these new facilities and operations.

P.6.1.4 **Retrieval Building and Operations**

The retrieval building currently under conceptual design would be either a mobile or large, fixed single-walled structure. Subatmospheric pressure would be maintained inside to prevent the escape of contaminants during retrieval operations. The ventilation system would include roughing filters and a bank of high-efficiency particulate air (HEPA) filters, for an estimated overall decontamination factor of 1,000.

Prior to erection of the building over the retrieval area, most of the soil cover may be removed. After the building is in place, the remainder of the soil, the polyvinyl sheeting, and the plywood cover would be removed to expose the waste containers and permit retrieval.

Waste containers would be inventoried and examined to confirm their integrity. Any breached containers would be placed in a waste transfer container and loaded into a transfer vehicle. Forklifts would remove the intact containers from the stacks and place them into the transfer vehicle. The waste would be transferred from the retrieval building to drum-venting and -examining facilities. Following venting and examining, the container would be placed in storage modules for eventual transfer to a processing facility or a transporter loading facility. All transfers would be made using the controlled roadways within the RWMC.

P.6.1.5 **Processing to Meet WIPP WAC**

Facilities are also being conceptually designed to provide for the storage, treatment, and repackaging of the retrieved waste to meet the WIPP WAC. Noncertifiable drums and boxes would be segregated, based on nondestructive examination, into waste packages containing large metallic components, packages containing liquids or respirable/dispersible particulates, and oversize packages that do not meet transportation requirements. Treatment processes under consideration include size reduction using mechanical and plasma arc cutting to size-reduce metallic components, immobilization to stabilize free liquids or respirable/dispersible particulates, and shredding/compaction to shred and repackage waste.

These facilities would be designed to ensure two levels of containment (in addition to the waste container) for all waste processing and repackaging areas. The ventilation system would be designed to maintain progressively lower pressures between the outside atmosphere and the waste processing areas. All air removed by the ventilation systems would pass through appropriate HEPA filtration systems for an estimated overall decontamination factor of 1,000.

Prior to construction of these facilities under conceptual design, NEPA documentation will be prepared to analyze the impacts of the proposed retrieval, treatment, and repackaging activities at the Idaho National Engineering Laboratory; alternatives would be considered.

P.6.2 PROCESS EXPERIMENTAL PILOT PLANT

The 1980 FEIS discussed in Subsection 9.8 the effects of removing the stored TRU waste from the Idaho National Engineering Laboratory. Three methods of processing were considered: slagging pyrolysis, repackaging only, and overpacking. Further investigation indicated that slagging pyrolysis would not meet performance objectives. As an alternative, shredding and incineration were considered and an experimental research and development process plant known as the Process Experimental Pilot Plant (PREPP) was constructed to demonstrate the efficacy of a process to certify a limited volume of TRU waste in retrievable storage.

The PREPP is designed to process waste to

- provide processing and repackaging to meet DOT 49 CFR 173 transport requirements
- comply with current EPA land disposal restrictions per 40 CFR Part 268
- reduce waste volume by incineration
- process materials into a form meeting the WIPP or other disposal facility waste acceptance criteria (see Appendix A)
- any combination of these requirements.

P.6.2.1 Existing Facilities and Process

The PREPP is located at the Test Area North (TAN) site on the Idaho National Engineering Laboratory. This area also includes the Water Reactor Research Test Facility, Special Manufacturing Capability Facility, Spent Fuel Technology Facilities, and the Technical Support Facility.

The PREPP occupies a portion of the TAN-607 building that was originally designated as the north machine bay. It is a two-story, double-walled, steel enclosure, with the interior separated into compartments by concrete floors, internal steel walls, and air locks.

Waste containers (drums or boxes) would be delivered to PREPP and unloaded in the shipping/receiving area or waste storage facility using mechanical methods. Containers would then be visually inspected for shipping damage, and the container information would be logged into a waste tracking system.

To initiate processing, the waste containers would be transported from the receiving area through airlocks to the opening and verification enclosure. Containers would then be transferred to the shredder enclosure or maintained in the opening and verification enclosure until unprocessable items are removed from the container. The waste containers would then be fed into an electric-powered shredder with counter-rotating intermeshing teeth. The shredded waste would then be transferred by a conveyor and auger feed system to the rotary kiln.

The refractory-lined kiln and secondary combustion chamber comprise the incineration system. In the kiln, the shredded waste would be exposed to a 1,500 to 1,800^o F (815 to 982^o C) oxidizing environment maintained at a slightly negative pressure. All combustibles would be burned or gasified. Combustion gases would then pass to the secondary combustion chamber, where they would be subjected to temperatures in the range of 1,800 to 2,300^o F (892 to 1,260^o C), ensuring complete combustion. The gases would then be directed to the offgas treatment system.

Following incineration, the solid waste residue would drop onto the discharge conveyor. After cooling, this ash would be separated into coarse and fine components by the trommel ash segregator. This unit consists of two rotating concentric drums with holes such that fine ash would drop into the hopper below while larger pieces would continue through the trommel to the drum fill enclosure. The fine ash would then be transferred from the trommel hopper to filtering hopper tanks by a pneumatic transport system.

The transport air would be separated from the fine ash by fabric bag filters and would continue through a high efficiency particulate air (HEPA) filter and eventually exhaust from the building via the filtered HVAC system. When the bag filter accumulates a cubic foot of fine ash, it would discharge to a blender tank below. After thorough blending, the fine ash could be sampled to ascertain chemical and physical properties. This information would then be used to properly mix the ash waste into the grout.

Coarse material arrives at the drum fill enclosure room, where operators using glove ports, enclosed rakes, grapples, and leaded acrylic viewing ports would transfer it to the fill drum.

The grout mixer is also located in the drum fill enclosure directly above the fill drum. The grout mixer is designed to produce one drum or less of grout to minimize grout set-up problems and cleaning requirements. Sand, cement, fines, sludge, solution from the offgas cleaning system, and, if necessary, potable water, would be added to the grout mixer. Material coming from the fines blender would be weighed in the fines weigh tank. Discharge from this tank to the grout mixer would be controlled by a metering valve. Sludge that has been accumulated in the sludge tank would then be added directly to the grout mixer. Water would be provided to the mixer from the potable water system. A plasticizer can be added directly to the grout mixer, reducing the amount of water required in the mixture and improving the flow characteristics of the grout around the shredded material in the drum.

After mixing, the wet grout would be discharged to the fill drum below. During the filling process, the operator can mix the grout and coarse material into the drum in layers, turning the drum vibrator on for short time periods to settle the contents and to fill any voids.

Once the drums have been filled, they would be surveyed for radiation, decontaminated if required, weighed, sampled, labelled, and temporarily sealed. After curing for approximately 3 days, each drum would undergo a final inspection, decontamination if required, and permanent installation of the lid. Containers meeting final inspection criteria would then be placed outside the containment area for shipment to SWEPP or an approved disposal site.

The offgas treatment system is designed to cool and neutralize the offgases and remove particulates. This system is composed of seven major treatment components: a wet quencher, a venturi scrubber, an entrainment eliminator, a mist eliminator, gas reheaters, four prefilters and four banks of dual-stage HEPA filters, and three induced draft fans.

The PREPP HVAC system consists of supply and exhaust fans, HEPA filters, ductwork and dampers, air conditioning units, instrumentation, and controls. The system would be automatically controlled to maintain three pressure control zones for contamination confinement. This type of pressure zone configuration will ensure that air flows from areas of least contamination potential (such as the control room) to areas of most contamination potential (such as the kiln room).

After monitoring for oxygen and carbon monoxide levels (to allow evaluation and control of the incineration process), the combustion gases will enter the quencher, where they would be cooled and neutralized by sodium carbonate solution spray. They would then pass to the venturi scrubber, where particulates are removed and the gas further neutralized. The entrainment eliminator and the mist eliminator would remove moisture. The gas would then be heated by reheaters and directed through the dual-stage HEPA filter bank.

Offgas air would be then directed to the stack. After entering the stack, a representative sample would be drawn off and routed to a continuous stack monitor. The stack monitor would be used to quantify and characterize any radioactive material in the stack exhaust. This information would be used to verify that stack radioactive releases are below regulatory requirements and to notify operating personnel if limits are being approached so that process adjustments could be initiated.

In addition to process monitoring equipment, PREPP would have instrumentation throughout the facility to warn personnel of direct radiation or airborne radiological contamination. Air samples would also be taken and analyzed to determine if organic hazardous chemicals are present, outside of process equipment. The HVAC system, which provides room ventilation, would also be equipped with a radiological monitoring system similar to the one identified for the offgas system.

P.6.2.2 **Waste Characteristics**

Waste materials that could be treated at the PREPP consist of construction and demolition materials, laboratory equipment and materials, process materials, process equipment, protective clothing, maintenance equipment, decontamination materials, and miscellaneous materials. Waste forms include sludges; combustibles, including rags, plastics, and wood; inorganics, including glass; and oxidized lead and other metals. It is anticipated that uncontained free liquids are present in some containers. Absorbed liquids would also be present in the feed, as absorbent material would be added to the drums by the waste generators before the containers are sealed for shipment. The waste currently identified is contained in either plywood boxes covered with fiberglass-reinforced polyester, 55-gal steel drums with 90 mil polyethylene liners, or steel bins.

PREPP operations would generate solid incinerator residue and offgas emissions. Scrubber solution and liquid effluent would be reused or mixed with grout to encapsulate incinerator residues in the final product drums. Airborne emissions would be minimized by using the best available control technology.

No radioactive or hazardous liquid waste would be released from PREPP. All of the liquid used in the process would either be recycled in the process or mixed with the final grout in the product waste drums. Approximately 65 ft^3 ($1.8m^3$) of solid waste would be generated each month due to processing operations. This solid waste includes filter media and decontamination/maintenance materials. Whenever possible, these materials would be processed through the incinerator.

Processed TRU waste would be returned to RWMC for certification at SWEPP if necessary and for storage and eventual loading for transport to the WIPP. The cemented wastes leaving PREPP are expected to meet the WIPP WAC. Containers would not be allowed to have an alpha contamination level on the outside of the container greater than 20 dpm/dm^2, or 200 dpm/dm^2 for beta-gamma isotopes. Also, the surface gamma dose rates shall be no greater than 200 mR/h; the average rate is expected to be less than 10 mR/h.

P.7 **ROCKY FLATS PLANT**

P.7.1 **PROCESSING**

The Rocky Flats Plant Supercompaction and Repackaging Facility and TRU Waste Shredder would process solid waste which is newly generated during routine production operations, maintenance activities, and laboratory support operations and may process waste in permitted storage. The Colorado Department of Health currently recognizes eight permitted storage areas at the Rocky Flats Plant for TRU mixed waste. The areas differ in size for a total permitted storage capacity of 1,601 yd^3. The storage units are within existing structures having concrete floors covered with epoxy paint and fenced areas within the buildings, which allow segregation of the storage facility from adjacent operations.

Two categories of waste would be processed: soft or combustible waste and hard or noncombustible waste. Combustible waste includes such items as paper and plastic. Noncombustible waste includes miscellaneous metals, piping, motors, glass, Raschig rings, process filters, and high-efficiency particulate air (HEPA) filters. The waste types are separated into designated drums at the point of generation, and separation is maintained throughout the waste management operations.

Hard waste packaged in 35-gal steel drums would be directly supercompacted (drum and all) into "pucks," and the pucks would be loaded into 55-gal steel drums for final disposal. Bags of soft waste, initially packaged in 55-gal drums, would be unpacked and precompacted into 35-gal drums, and then the 35-gal drums would be supercompacted as described above. Figure P.7.1 shows a process flow diagram.

The Rocky Flats Plant TRU Waste Shredder would be used to process discarded graphite molds and filters. Approximately 80 percent of the waste to be processed in the TRU Waste Shredder would be filters. The remaining 20 percent would be graphite molds.

The graphite molds would be crushed in the shredder. Approximately 10 to 20 55-gal drums of classified graphite molds would be processed in 1 month. Each drum would contain approximately 100 to 150 pounds of molds. Weighing approximately 20 pounds each, the molds would be individually wrapped in heavy vinyl bags inside the drums. They would be removed from the drums prior to shredding. Once processed, they would be considered TRU waste.

The filter waste that would be shredded includes HEPA filters and process filters. Approximately 30 to 70 55-gal drums of combined filter types would be processed in 1 month. The HEPA filters with their frames would be individually wrapped in heavy vinyl and contained in cardboard boxes. The process filters would be contained in 55-gal drums. The filters would be shredded for volume reduction and packaged in

Figure P.7.1 Supercompaction and repackaging facility process flow diagram

35-gal steel drums for supercompaction in the Supercompaction and Repackaging Facility as hard waste.

P.7.2 SUPERCOMPACTION AND REPACKAGING FACILITY EQUIPMENT DESCRIPTION

Most of the Supercompaction and Repackaging Facility equipment would be contained in a 1,105 cubic foot single-walled, unshielded glovebox, which would be subdivided into nine sections:

- the hard-waste airlock entry chamber and associated interlocks
- the soft-waste airlock entry chamber and associated interlocks
- the 30-ton precompactor area
- the drum piercing station
- the press loader/unloader
- the 2,200-ton supercompactor area, which includes a small liquid waste collection system
- the puck conveyer
- the monorail/hoist
- the load-out area.

The glovebox enclosure would be equipped with two airlock chambers for the introduction of waste into the system, and two drum ports for the removal of compacted waste from the system. One of the airlock chambers would receive soft waste contained in polyethylene bags (i.e., soft-waste airlock). The second chamber would receive empty 35-gal steel drums and 35-gal steel drums containing hard waste.

Safety interlocks would be installed in each of the two airlock chambers. The airlock chambers would each be equipped with an inner and an outer door. The interlock system would control operation of the door by allowing only one of the four doors to be opened at any given time. In addition, a minimum airflow of 150 feet per minute directed into the glovebox would automatically be maintained across the opening of each door.

The remainder of the equipment would be located outside of the glovebox enclosure and would include a downdraft table with a stainless steel hood and sliding glass doors for unloading soft waste; hydraulic systems to operate the compactors and the press loader/unloader; a control station; and peripheral equipment, which includes instrumentation, associated piping, ductwork, and electrical utilities.

Drums would be scanned for the presence of free liquids by the real time radiography unit prior to being transported to the Supercompaction and Repackaging Facility. If liquids were detected, the drums would be repackaged.

Drums which are to be compacted in the Supercompaction and Repackaging Facility unit would first be sent to one of several drum counters to determine the plutonium content of each drum. Administrative controls would be used to ensure that drums entering the Supercompaction and Repackaging Facility do not exceed the established 50-gram plutonium limit. If a drum were found to exceed the limit, it would not be supercompacted but would be repackaged in the Advanced Size Reduction Facility. Drums and their associated plutonium content would be logged prior to processing in the Supercompaction and Repackaging Facility. Drums would be arranged for processing according to the type of material contained, compatibility, the plutonium content of each of the drums and the final overpacked drum (maximum of 100 grams of plutonium), and the maximum combined weight (not to exceed 800 pounds). Additionally, selection of drums for processing in the Supercompaction and Repackaging Facility would be based on the compatibility of the material contained (i.e., the expected height following compaction to provide the most efficient packaging of the final drums and, therefore, maximize volume reduction).

P.7.2.1 **Hard-Waste Entry into the Supercompaction and Repackaging Facility**

Drums of hard waste would be transported from the staging area by a forklift. A 35-gal steel drum containing double-bagged hard waste surrounded by a polyethylene liner would be placed by forklift onto the roller table adjacent to the hard-waste entry airlock. The outer airlock door would be opened from the airlock control station and the drum would be pushed manually into the airlock. The outer door would be closed and the interlock systems would then allow the inner door to be opened. The drum would be automatically conveyed into the glovebox by operators working at the control panel and the inner airlock door would be closed.

P.7.2.2 **Soft-Waste Entry and Precompaction**

A downdraft table would be located outside of the glovebox at the soft-waste airlock. It would be equipped with a negative pressure (HEPA) filtration system to minimize the unlikely spread of radioactive and hazardous contaminants within the room while waste is being introduced to the glovebox. A stainless steel hood with sliding glass doors would be placed over the table to increase the effectiveness of the ventilation exhaust system. The enclosure would be operated at negative pressure with the air flow directed into the HEPA filtration system.

Prior to admittance of soft waste into the Supercompaction and Repackaging Facility, an empty 35-gal steel drum would be entered into the glovebox. The drum would be transferred to the precompactor area, where it would be clamped to the precompactor.

Polyethylene lined 55-gal drums containing soft TRU waste would be transported to the staging area to the downdraft table. The lid of the drum would be removed and contamination surveyed by radiation monitoring personnel. The drum liner containing double-bagged contents would be removed from the 55-gal drum as a unit. The soft-waste airlock chamber outer door would be opened from the airlock control station, and the liner and waste would be manually entered as a unit into the chamber. Waste would be manually moved into the glovebox by personnel working outside the glovebox through gloveports.

Personnel working from outside the glovebox through gloveports would cut open the drum liner and remove the inner plastic bags containing soft waste. The inner bags and the liner would then be placed into each empty 35-gal drum located on the precompactor. The precompactor is a 30-ton force hydraulic compactor. The waste would be precompacted, and more bags of soft waste (maximum of three additional 55-gal drums) would be introduced into the glovebox by the method described above. The bags would be added to the 35-gal drum and precompacted until the drum reaches capacity.

Following precompaction, a lid would then be placed on each 35-gal drum and secured by an operator working from outside the glovebox. The drum would be unclamped from the precompactor and the conveyor would then be activated by the operator to move the drum to the drum piercing station and then to the hydraulic loader/unloader, where it would be loaded into the supercompactor.

Photoelectric cells located at the centerline of the gloveports on either side of the precompactor would be connected to safety shutoff devices that disable the precompactor ram if personnel have their hands in the gloves during actual precompaction.

P.7.2.3 **Supercompaction**

Precompacted soft-waste drums and hard-waste drums ready for processing would be conveyed by motorized conveyer to the drum piercing station. Each drum would be pierced with four holes prior to supercompaction. The procedure would allow any entrapped air to escape from the drum and would thereby ensure a greater volume reduction. The piercing procedure would also reduce the possibility of the drum springing back following compaction.

A hydraulic loader/unloader would automatically load the drums onto the supercompactor for compaction. A mold would be hydraulically lowered around the drum to contain lateral expansion during supercompaction. Once the mold is in position, the supercompactor power unit will pressurize the hydraulic ram cylinder. Then the ram will descend and compact the drum and its contents into a puck, measuring 2 to 18 inches in height. The loader/unloader would again be used to move the puck from the supercompactor onto an automated conveyer in the load-out section of the glovebox.

P.7.3 **TRU WASTE SHREDDER DESCRIPTION**

P.7.3.1 **TRU Waste Shredder Equipment Description**

All of the TRU Waste Shredder equipment except the downdraft table would be contained in a single-walled, lead-shielded glovebox. Unlike the Supercompaction and Repackaging Facility glovebox, the TRU Waste Shredder would be composed of the following equipment:

- a downdraft table at the glovebox airlock
- an airlock chamber with safety interlock system
- a shredder (hopper, cutting chamber, and material compressors)
- drum ports in the load-out area
- a dry-chemical fire-suppression system
- a scale.

P.7.3.2 **TRU Waste Shredder Process Description**

The TRU Waste Shredder would be used to size-reduce graphite molds, HEPA filters, and process filters by shredding and compressing the material. The graphite molds and process filters would be contained in 55-gal drums. The incoming whole HEPA filters would be wrapped in heavy vinyl and contained in lined cardboard boxes.

All drums destined for processing in the TRU Waste Shredder unit would first be sent to one of several drum counters. The plutonium content of each drum and box would be determined. Drums and filter boxes entering the TRU Waste Shredder unit would not exceed established fissile material limits.

Waste Entry. The downdraft table, airlock chamber, and safety interlock system would be similar to those found in the Supercompaction and Repackaging Facility. Boxes containing filters and drums containing filter media and graphite waste to be processed in the TRU Waste Shredder system would be staged in the drum storage area. One box or drum at a time would be transported on a dolly to the TRU Waste Shredder drum hoist. The box or drum would be raised to the TRU Waste Shredder platform level in front of the downdraft table. The downdraft table hood door would be opened and the contents of the box or drum would then be opened and the contents manually transferred from the downdraft table into the airlock chamber. When the chamber has been loaded, the outer airlock door would be closed and the inner door opened. The molds or filters would then be manually moved from the chamber into the glovebox by operators working outside the glovebox through gloveports.

Shredding and Compacting. The graphite molds, HEPA filters, process filters, and filter media would be batched separately for shredding. The waste would be loaded onto a conveyer and manually transported to the shredder feed hopper. The shredder would be gravity fed through the hopper, located above the shredder chamber. The shredder would consist of two counter-rotating shafts with knives able to shred molds into declassified scraps measuring 1 inch by 2 inches by 2 inches or smaller, and HEPA filters into similar-sized small pieces. The shredder would be equipped with an automatic kick-out device which would reject unshreddable materials from the shredding chamber. The unshreddable materials would be removed through kickout doors and would be disposed along with shredded material.

Shredder material would be extruded through the bottom of the shredder into the material compressor. Waste material would be compressed and extruded through a discharge into a tray located on the floor or the glovebox. The material compressor would be used for further volume reduction of the shredder material. The hopper loading-shredding-material compression operation would be repeated until all molds or filters have been processed.

P.8 **BIN-SCALE TESTS**

During the Test Phase, the DOE proposes to operate the WIPP with limited amounts of waste. For this SEIS, it is assumed that the maximum amount of TRU waste that would be used during the Test Phase is 10 percent of the TRU waste (by volume) that could ultimately be emplaced at the WIPP. It is also assumed that waste would be shipped from all 10 facilities, although it is now likely that only waste from the Rocky Flats Plant and Idaho National Engineering Laboratory would be used during the initial phase of the proposed Test Phase. The actual amount and source of waste proposed for the Test Phase will be determined by the Secretary of Energy.

The bin-scale tests involve testing in multiple large, instrumented metal "bins" with specially prepared TRU waste and appropriate material additives. The "prepared" waste includes up to 6 drum-volume-equivalents of a specific type of actual CH TRU waste with added backfill materials (including salt), metal corrodants (mild steel wire mesh), and brine (to be injected at WIPP). Within each individual test bin there will be a specific type of TRU waste, either noncompacted or compacted. Any plastic bags encapsulating this waste will be "prebreached"; that is, the bags will be sliced or slashed, or the waste itself will be shredded. This "prebreaching" permits contact between, and interactions of, the waste with other added components within the bin, and within a time frame shorter than expected in the repository. Additional details regarding the bin-scale tests are presented in Appendix O.

Special preparation of the waste and bin preparation would occur at the generator facilities. The program design includes the following assumptions with regard to waste packaging and transportation.

Two additions must be made to the preinstrumented bin before the waste would be placed in the test bin. First, about a half-drum volume of backfill material would be placed in the bottom of the test bin. Second, about 6 drum-equivalents of bare, unpainted steel (mild steel wire mesh) would be placed along the bottom and side walls of the bin. The bins would then be remotely filled with waste.

Prior to bin loading, a waste characterization effort would be undertaken. Although this characterization effort is evolving, it is currently anticipated that the volatile organic compounds in the headspace of each drum would be sampled and its constituents' concentrations determined. After gas sampling, each drum would be opened and its contents qualitatively assessed by visual inspection (i.e., relative evaluation of waste type and form). In addition, for processed sludges only, a sample would be collected from each drum and would be subjected to a complete chemical and radiological analysis by recognized protocols. After sampling, waste would be placed in the bins.

After the waste is placed in the bins, another half-drum volume of backfill material would be sprinkled on top of the waste materials. The mated bin-lid and liner-lid combination would then be attached to the bin and sealed. The filled bin would be checked for surface contamination and, if necessary, decontaminated following standard procedures of the generator facility.

The waste-filled test bins would be inserted into standard waste box (SWB) facilities for transportation to the WIPP. The upper gas valves on the test bins (with HEPA filters) would be left in the open, gas-release position during transportation. Therefore, any gases vented would also be filtered through the redundant HEPA filter of the SWB. The SWBs would be loaded into the TRUPACT-II transportation containers and trucked to the WIPP. Waste bins would be removed from the SWBs in the WIPP underground and brine would be injected just prior to emplacement. The procedures for loading and assembling TRUPACT-IIs are presented in Appendix L.

REFERENCES FOR APPENDIX P

BEIR, 1980. "The Effects on Populations of Exposure to Low Levels of Ionizing Radiation: 1980," National Academy of Sciences, Committee on the Biological Effects of Ionizing Radiation.

DOE (U.S. Department of Energy), 1989. Draft Plan for the Waste Isolation Pilot Plant Test Phase: Performance Assessment and Operations Demonstration, DOE/WIPP 89-011, Carlsbad, New Mexico.

DOE (U.S. Department of Energy), 1988a. "Environmental Assessment on Management Activities for Newly Generated TRU Waste, Savannah River Plant," DOE/EA-0315.

DOE (U.S. Department of Energy), 1988b. "Radiation Protection for Occupational Workers," DOE Order 5480.11, Washington, D.C.

DOE (U.S. Department of Energy), 1988c. "Radiation Protection of the Public and the Environment," DOE Order 5400.3, draft, Washington, D.C.

DOE (U.S. Department of Energy), 1987a. "Disposal of Hanford Defense High-Level, TRU and Tank Waste," DOE/EIS-0113.

DOE (U.S. Department of Energy), 1987b. "Safety Analysis Report for the Stored Waste Examination Pilot Plant (SWEPP)," Rev. 1, WM-PD-87-023.

DOE (U.S. Department of Energy), 1986. Safety Analysis for the Radioactive Waste Management Complex at the Idaho National Engineering Laboratory, WM-PD-86-011.

DOE (U.S. Department of Energy), 1986. "Safety Analysis Report for the Radioactive Waste Management Complex at the Idaho National Engineering Laboratory," WM-PD-86-011, Addenum A, "Safety Assessment for Pad A Penetrator at RWMC", DRR-32-89, July 27, 1989, and Addendum B, "Safety Assessment for Vapor Vacuum Extraction Demonstration at RWMC," DRR-35-45, October 30, 1989.

ICRP (International Commission on Radiological Protection), 1979. Annals of the ICRP, ICRP Publication 30, Limits for Intake of Radionuclides by Workers, Pergamon Press, Washington, D.C.

Tait, T. D., 1983. Demonstration Test Assessment of the Slagging Pyrolysis Incinerator for Processing INEL Transuranic Waste, Abstract, EGG-TF-6192, EG&G Idaho, Inc., Idaho Falls, Idaho.