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1.1 Facility Background and Mission

The United States Department of Energy (DOE) was authorized by Public Law 96-164[1] to provide a research and development facility for demonstrating the safe permanent disposal of transuranic (TRU) wastes from national defense activities and programs of the United States exempted from regulations by the U.S. Nuclear Regulatory Commission (NRC). The Waste Isolation Pilot Plant (WIPP), located in southeastern New Mexico near Carlsbad, was constructed to determine the efficacy of an underground repository for disposal of TRU wastes.

In accordance with the 1981 and 1990 Records of Decision (ROD),[2,3] the development of the WIPP was to proceed with a phased approach. Development of the WIPP began with a siting phase, during which several sites were evaluated and the present site selected based on extensive geotechnical research, supplemented by testing.

The site and preliminary design validation phase (SPDV) followed the siting phase, during which two shafts were constructed, an underground testing area was excavated, and various geologic, hydrologic, and other geotechnical features were investigated. The construction phase followed the SPDV phase during which surface structures for receiving waste were built and underground excavations were completed for waste emplacement.

At the conclusion of the construction phase, the DOE proposed a test phase, to be followed by the disposal phase for waste emplacement operations. The test phase was to involve the use of limited quantities of contact-handled (CH) TRU waste to conduct tests in the WIPP underground to provide data for reducing the uncertainties in the performance assessment required for compliance with the long-term waste isolation regulations of the U.S. Environmental Protection Agency (EPA), Subpart B of 40 CFR Part 191.[4] To enable the receipt of CH-TRU waste at the WIPP site for the tests the Congress enacted the WIPP Land Withdrawal Act[5] of 1992 (Public Law 102-579). The law also provided for authorizations of detailed regulatory requirements for the WIPP.

As a result of major programmatic redirection in October 1993, the WIPP test phase was modified by substituting the previously planned WIPP underground radioactive tests with laboratory tests. In conjunction, WIPP operations would proceed directly with the disposal phase CH TRU waste emplacement operations starting in mid-1998, assuming successful demonstration of compliance with applicable federal and state laws and regulations, and successful completion of the WIPP CH Operational Readiness Review (ORR). The CH ORR closely examined the safety bases of the facility and the status of attendant conformance to ensure that the facility was operationally ready and that CH waste emplacement operations would be conducted safely.

Disposal operations began in March 1999. The disposal phase currently scheduled to last 35 years,[6, 7] will consist of receiving, handling, and emplacing TRU waste in the repository for disposal, and will end when the design capacity of the repository has been reached.

The decommissioning phase, during which the repository will be prepared for permanent closure, will follow the disposal phase. Surface facilities will be decontaminated and decommissioned, underground excavations will be prepared for closure, and shaft seals will be emplaced. This phase is currently projected to last for 10 years. The post-decommissioning phase will consist of active and passive institutional controls. Active institutional controls will include activities such as control of access to the site, implemented consistent with applicable regulations and permit conditions and will continue for at least 100 years[8].

These controls will be designed to ensure that the potential for future, inadvertent human intrusion is reduced to a level that renders such intrusion unlikely.

This Safety Analysis Report (SAR) documents the safety analyses that develop and evaluate the adequacy of the WIPP CH TRU safety bases necessary to ensure the safety of workers, the public, and the environment from the hazards posed by WIPP waste handling and emplacement operations during the disposal phase and hazards associated with the decommissioning and decontamination phase.

The analyses of the hazards associated with the long-term (10,000 year) disposal of TRU and TRU mixed waste, and demonstration of compliance with the requirements of 40 CFR 191, Subpart B[4] have been addressed in detail in the WIPP Compliance Certification Application (CCA).[8] The Environmental Protection Agency (EPA) reviewed the CCA and subsequently certified that the WIPP was in compliance with the requirements in 40 CFR 191, Subpart B and C on May 13, 1998.[9] SAR Section 5.5, Long-Term Waste Isolation Assessment summarizes the assessment.

References for Section 1.1

1. Public Law 96-164, Department of Energy National Security and Military Applications of Nuclear Energy Authorization Act of 1980, December 29, 1979.

2. U.S. Department of Energy, 46 FR 9162, Record of Decision, Waste Isolation Pilot Plant, January 28, 1981.

3. U.S. Department of Energy, 55 FR 256892, Record of Decision, Waste Isolation Pilot Plant, June 22, 1990.

4. U.S. Environmental Protection Agency, 40 CFR 191, Environmental Radiation Protection for Management and Disposal of Spent Nuclear Fuel, High Level and Transuranic Wastes, Subpart B, Environmental Standards for Disposal, December 1993.

5. Public Law 102-579, Waste Isolation Pilot Plant Land Withdrawal Act, US Congress, October, 1992 [as amended by Public Law 104-201].

6. DOE/NTP-96-1204, Revision 0, The National Transuranic Waste Management Plan, U. S. Department of Energy, Carlsbad Area Office, September 30, 1996, Section 2.1.

7. DOE/EIS-0026-S-2, WIPP Disposal Phase Final Supplemental Environmental Impact Statement, U. S. Department of Energy, Carlsbad Area Office, September, 1997.

8. Title 40 CFR 191 Compliance Certification Application for the Waste Isolation Pilot Plant, DOE/CAO-1996-2184, October 1996, Section 7.1.4.

9. EPA (U.S. Environmental Protection Agency), 1998. Criteria for the Certification and Recertification of the Waste Isolation Pilot Plant's Compliance with the Disposal Regulations: Certification Decision: Final Rule, Federal Register, Volume 63, pages 27354 through 27406, May 18, 1998, Radiation Protection Division, Washington, D.C.

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1.2 Facility Overview

1.2.1 Facility Design

The WIPP is located in Eddy County in southeastern New Mexico, 26 miles (41.6 km) east of Carlsbad as shown in Figure 1.2-1. The amount of land that has been set aside for the WIPP includes an area of 10,240 acres (41 km^2). The WIPP is located in an area of low population density with less than 30 permanent residents living within a ten-mile radius. The area surrounding the facility is used primarily for grazing, and development of potash, oil, salt, and gas resources. Development of these resources results in a transient population (non-permanent) consisting principally of workers at three potash mines that are located within ten miles of the WIPP. The largest population center nearest the WIPP is the city of Carlsbad, 26 miles (41.6 km) to the west, with approximately 25,000 inhabitants. Two smaller communities, Loving (population approximately 1300) and Malaga (population approximately 200), are located about 20 miles (32 km) southwest of the facility. As the result of the WIPP Land Withdrawal Act of 1992, no mineral resource development is allowed within the WIPP Site Boundary (with the exception of existing leases).

The WIPP is designed to receive and handle a maximum of 500,000 ft^3/yr (14,160 m^3/yr) CH TRU waste and 10,000 ft^3/yr (283 m^3/yr) remote handled (RH) TRU waste. The CH TRU waste will be contained in 55-gallon (208 L) drums, standard waste boxes (SWBs), ten drum overpacks, 85-gallon (322 L) drum overpacks, 55-gallon (208 L) drums overpacked in SWBs, and pipe containers in 55-gallon (208 L) drums. The WIPP facility is designed to have a disposal capacity for TRU waste of 6.2×10^6 ft^3 (1.76×10^5 m^3). Current design is that RH waste will be packaged in steel canisters and transported to the WIPP facility in shielded road casks. The WIPP facility has sufficient capacity to handle the 250,000 ft^3 (7,080 m^3) of RH TRU that was established in the ROD[1] as a total volume. In addition, the WIPP Land Withdrawal Act of 1992[2] limits the total RH TRU activity to 5.1×10^6 curies.

CH TRU wastes will be disposed of in the 100-acre (0.4 km^2) disposal area on a horizon located 2,150 feet (655 meters) beneath the surface in a deep, bedded salt formation. Waste will be transferred from the surface to the disposal horizon through a waste shaft using a hoisting arrangement. The disposal phase is currently scheduled to last for 35 years.[3, 4]

The Department of Energy - Carlsbad Area Office (DOE-CAO) has determined that waste emplacement will only follow a decision, by DOE and by appropriate regulatory agencies, that permanent disposal in the WIPP facility protects human health and the environment. When initiated, the placement of waste in the WIPP will be for the purpose of permanent disposal with no intent to retrieve. However, if in the future it is determined that recovery of disposed waste is required, prior to commencement of recovery operations: (1) principal design and safety criteria for structures, systems, and components (SSCs) that protect the public, workers, and the environment from hazards posed by recovery shall be developed, and (2) those hazards associated with the recovery design and process will be analyzed to address recovery.

The WIPP is divided into three basic groups: surface structures, shafts, and subsurface structures as shown in Figure 1.2-2. The WIPP surface structures (see Figure 1.2-3) accommodate the personnel, equipment, and support services required for the receipt, preparation, and transfer of waste from the surface to the underground. The surface structures are located in an area within a perimeter security fence. The primary surface operations at the WIPP are conducted in the Waste Handling Building (WHB), which is divided into the CH TRU waste handling area, the RH TRU waste handling area, and support areas. The CH TRU waste handling area includes the entrance air locks, CH Bay, a shielded holding area, and CH TRU support facilities.

The current design of the RH TRU waste handling area includes an RH Bay, cask receiving and preparation areas, hot cell complex, and a shielded cell for shielded road cask unloading, waste canister inspection, overpacking canisters, as required, and facility cask loading prior to transfer underground.

The vertical shafts extending from the surface to the underground horizon (see Figure 1.2-2) are the waste shaft, the salt handling shaft, the exhaust shaft, and the air intake shaft. These shafts are lined from the shaft collar to the top of the salt formation (about 850 ft [259 meters] below the surface), and are unlined through the salt formation. The shaft lining is designed to withstand the full piezometric water pressure associated with any water-bearing formation encountered. The waste shaft is located between the CH TRU and RH TRU areas in the WHB. It is nominally 19 feet (5.8 meters) in diameter and is serviced by a hoist utilizing a hoist cage that is primarily used for transportation of CH TRU and RH TRU wastes from the surface to underground disposal areas.

The underground areas (see Figure 1.2-4) consist of the waste disposal area, and the support area. The disposal area has four main entries (two entries for fresh air and two entries for return air) and a number of disposal rooms. The layout of the shafts and entries allows mining and disposal operations to proceed simultaneously. The first disposal panel is used to dispose waste while the next panel is being mined. Successive stages follow in a similar manner.

A typical disposal panel consists of seven disposal rooms. Each room is 33 feet (10 meters) wide, 13 feet (4 meters) high, and 300 feet (91.5 meters) long. The disposal rooms are separated by pillars of salt 100 feet (30.5 meters) wide and 300 feet (91.5 meters) long. Panel entries at the end of each of these disposal rooms are also 33 feet (10 meters) wide and 13 feet (4 meters) high and will be used for waste disposal, except for the first 200 feet (61 meters) from the main entries which are 22 feet (6.7 meters) wide by 14 feet (4.3 meters) high. This first 200 feet (61 meters) will be used for installation of panel closure systems.

1.2.2 Facility Operations

The principal operations of the WIPP involve the receipt of TRU and TRU mixed waste and emplacement in the underground salt repository for disposal (see Figure 1.2-5). Transporters carrying TRU waste arrive at the WIPP and are unloaded outside the WHB. The shipments are surveyed for external contamination prior to their movement into the WHB for unloading.

CH TRU waste will be shipped to the WIPP in Nuclear Regulatory Commission (NRC)-certified shipping packages. After the CH TRU waste shipping container is inspected for contamination, the loaded shipping container is moved into the WHB and placed on a handling dock. The container is opened, surveyed for radiation and contamination levels, and the waste containers are removed and placed on a facility pallet. This pallet is then transferred to the conveyance loading car, which is moved into the hoist cage in the waste shaft for transfer to the disposal horizon.

At the disposal horizon, the pallet is removed from the hoist cage, placed on the underground transporter, and moved to the CH TRU waste disposal room. In the disposal room, the containers are removed from the pallet and placed in the waste stack. The empty pallet is returned to the surface for reuse.

The waste received for placement in the WIPP facility must conform with the WIPP Waste Acceptance Criteria (WAC).[5] The operational philosophy at the WIPP facility is to start radiologically clean and stay radiologically clean. Consequently, any containers of waste that are found to be externally contaminated or damaged will be decontaminated or placed in a larger container (overpacked at the location contamination is found or damage occurs), or returned to the generating/shipping facility. Also, any local area of contamination will be isolated and/or decontaminated prior to continuation of the waste handling process.

Analyses in this SAR address CH TRU waste emplacement operations only. **Existing RH TRU design and operations information were retained for design configuration management purposes only (Changes to RH SSCs are evaluated through the configuration management process, for their impact on CH design and operations as evaluated in this SAR).** RH TRU waste handling and emplacement operations will be updated in future revisions of this SAR.

References for Section 1.2

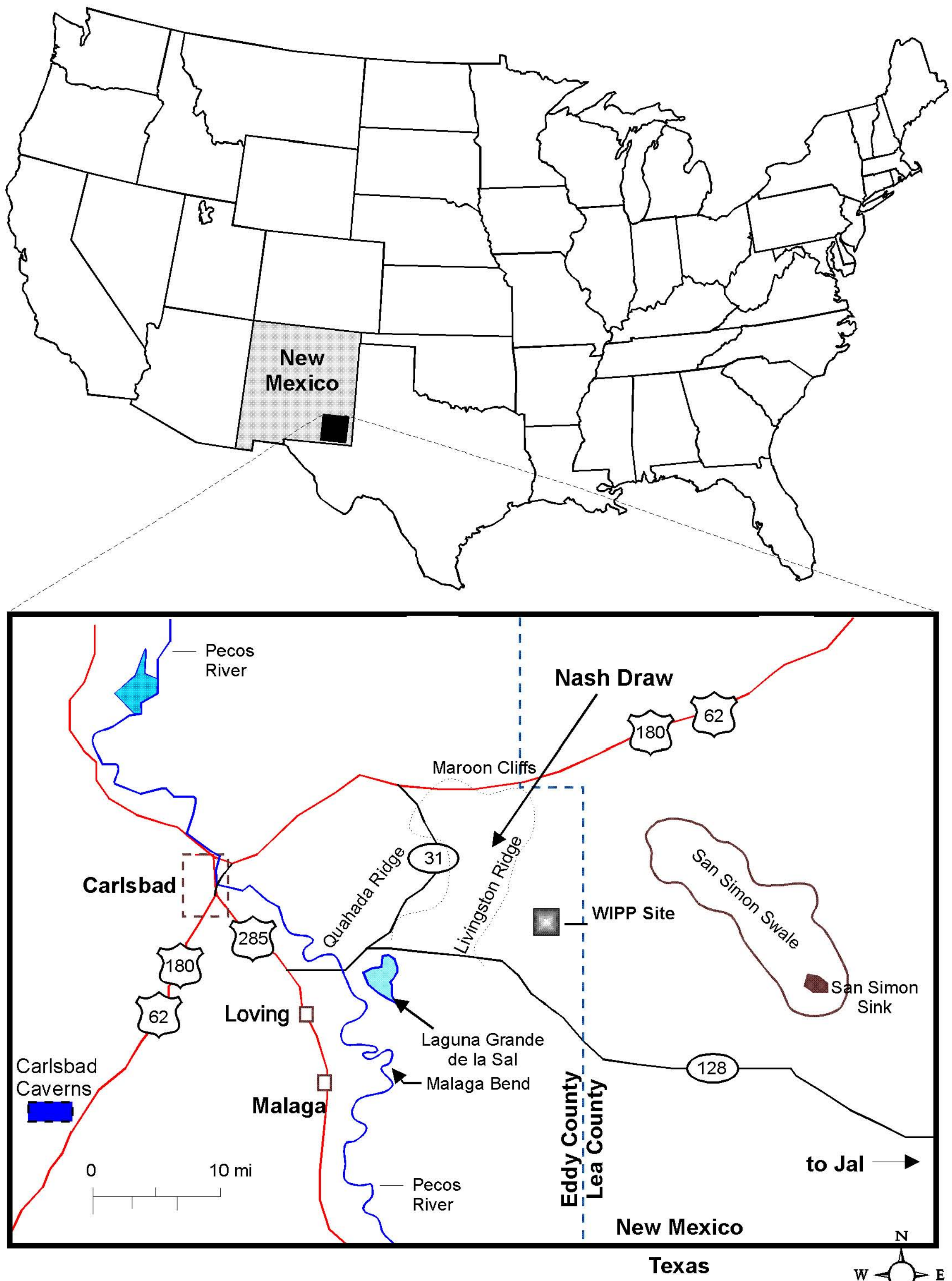
1. U.S. Department of Energy, 46 FR 9162, Record of Decision, Waste Isolation Pilot Plant, January 28, 1981

2. Public Law 102-579, Waste Isolation Pilot Plant Land Withdrawal Act, October, 1992 [as amended by Public Law 104-201].

3. DOE/NTP-96-1204, Revision 0, The National Transuranic Waste Management Plan, U. S. Department of Energy, Carlsbad Area Office, September 30, 1996, Section 2.1.

4. DOE/EIS-0026-S-2, WIPP Disposal Phase Final Supplemental Environmental Impact Statement, U. S. Department of Energy, Carlsbad Area Office, September, 1997.

5. WIPP-DOE-069, TRU Waste Acceptance Criteria for the Waste Isolation Pilot Plant, Revision 5, April 1996.



Source: Modified from Sandia, 1992, 3, 1-48. 2300.1

Figure 1.2-1, WIPP Location in Southeastern New Mexico

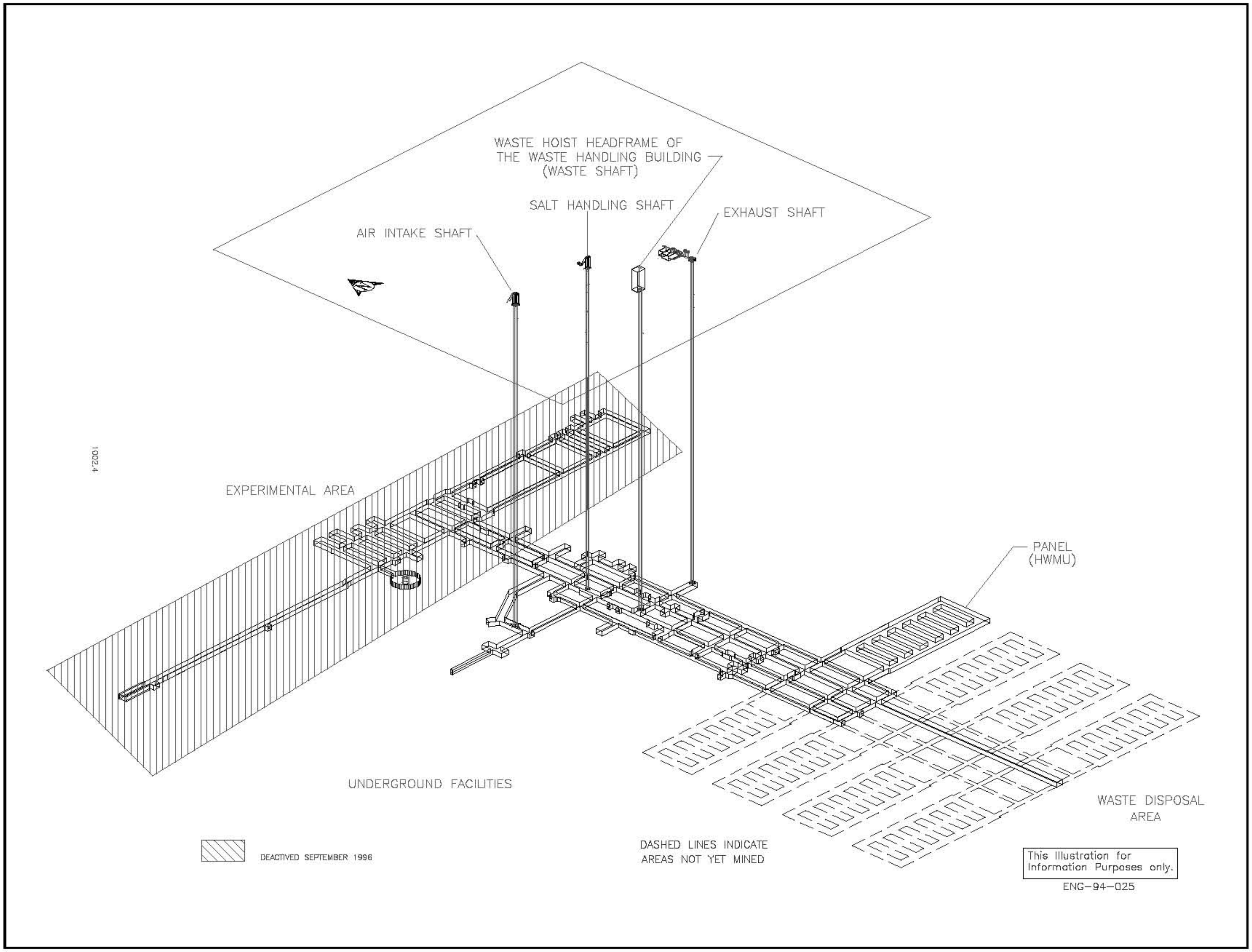


Figure 1.2-2, Spatial View of the WIPP Facility

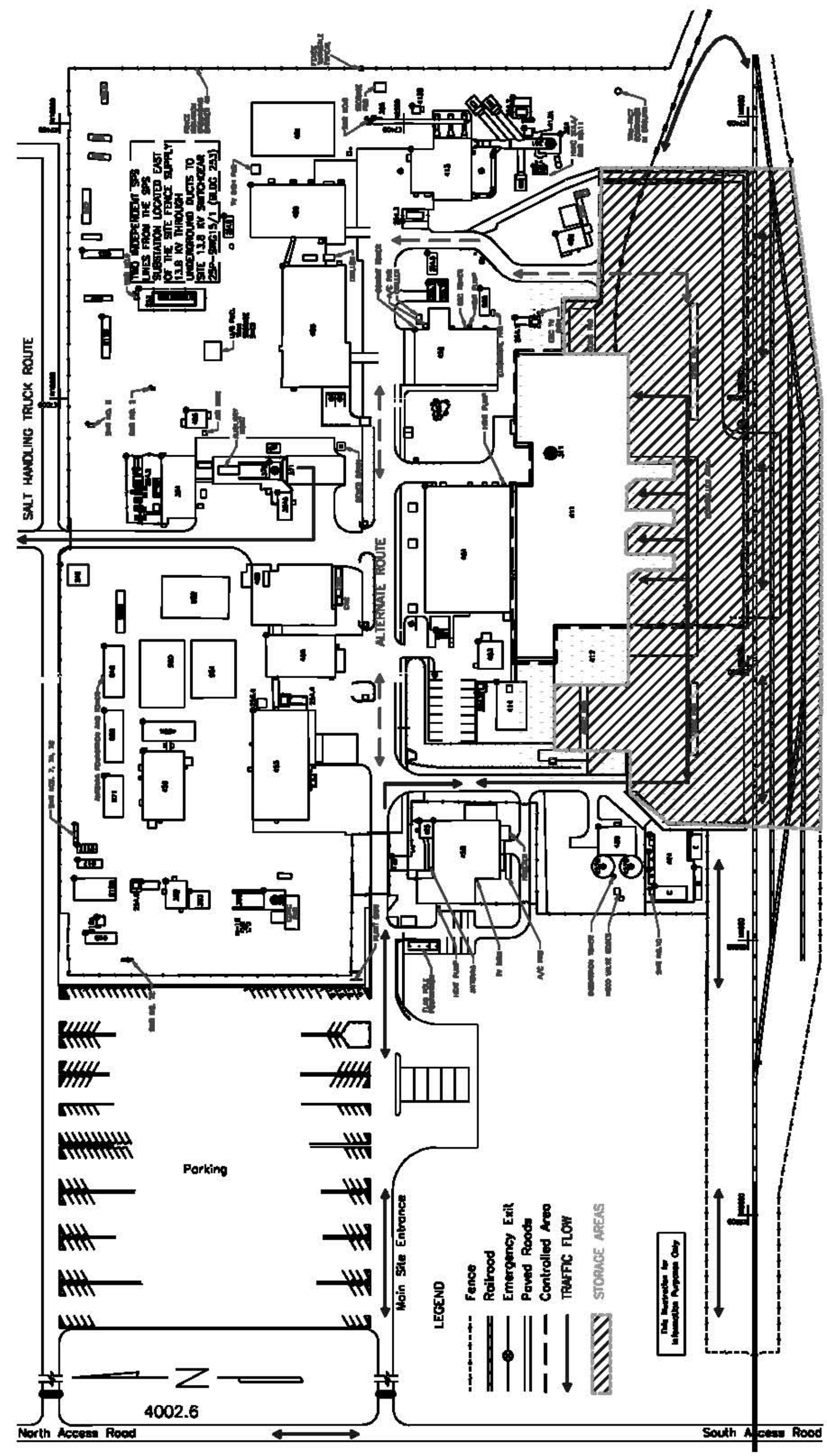


Figure 1.2-3a, WIPP Surface Structures

BLDG./ FAC. #	DESCRIPTION
242	NORTH GATEHOUSE
253	13.8 KV SWITCHGEAR 25P-SWG15/1
254.1	AREA SUBSTATION NO.1 25P-SW15.1
254.2	AREA SUBSTATION NO.2 25P-SW15.2
254.3	AREA SUBSTATION NO.3 25P-SW15.3
254.4	AREA SUBSTATION NO.4 25P-SW15.4
254.5	AREA SUBSTATION NO.5 25P-SW15.5
254.6	AREA SUBSTATION NO.6 25P-SW15.6
254.7	AREA SUBSTATION NO.7 25P-SW15.7
254.8	AREA SUBSTATION NO.8 25P-SW15.8
254.9	AREA SUBSTATION NO.9 25P-SW15.9
255.1	BACKUP GENERATOR #1 25-PE 503
255.2	BACKUP GENERATOR #2 25-PE 504
311	WASTE SHAFT
351	EXHAUST SHAFT
361	AIR INTAKE SHAFT
362	AIR INTAKE SHAFT/HOIST HOUSE
363	AIR INTAKE SHAFT/WINCH HOUSE
364	EFFLUENT MONITORING INSTRUMENT SHED A
365	EFFLUENT MONITORING INSTRUMENT SHED B
366	AIR INTAKE SHAFT HEADFRAME
371	SALT HANDLING SHAFT
372	SALT HANDLING SHAFT HEADFRAME
384	SALT HANDLING SHAFT HOISTHOUSE
384A	SALT HOIST OPERATIONS
411	WASTE HANDLING BUILDING
412	TRUPACT MAINTENANCE FACILITY
413	EXHAUST FILTER BUILDING
413A	EFFLUENT MONITORING ROOM A
413B	EFFLUENT MONITORING ROOM B
414	WATER CHILLER FACILITY & BLDG
451	SUPPORT BUILDING
452	SAFETY & EMERGENCY SERVICES FACILITY
453	WAREHOUSE/SHOPS BUILDING
455	AUXILLIARY WAREHOUSE BUILDING
456	WATER PUMPHOUSE

BLDG./ FAC. #	DESCRIPTION
457N	WATER TANK 25-D-001A
457S	WATER TANK 25-D-001B
458	GUARD AND SECURITY BUILDING
459	CORE STORAGE BUILDING
459A	SANDIA ANNEX
463	COMPRESSOR BUILDING
465	AUXILIARY AIR INTAKE
468	TELEPHONE HUT
473	ARMORY BUILDING
474	HAZARDOUS WASTE STORAGE FACILITY
474A	HAZARDOUS WASTE STORAGE BUILDING
474B	HAZARDOUS WASTE STORAGE BUILDING
474C	OIL & GREASE STORAGE BUILDING
474D	GAS BOTTLE STORAGE BUILDING
474E	HAZARD MATERIAL STORAGE BUILDING
474F	WASTE OIL RETAINER
475	GATEHOUSE
480	VEHICLE FUEL STATION
481	AUXILIARY WAREHOUSE
482	EXHAUST SHAFT HOIST EQUIP. WAREHOUSE
485	COMPRESSOR BUILDING
486	ENGINEERING BUILDING
489	TRAINING BUILDING
H-16	SANDIA TEST WELL (NOT IDENTIFIED)
908B	HBS TRAILER
910	ENVIRONMENTAL MONITORING TRAILER
911G	SANDIA OFFICES TRAILER

BLDG./ FAC. #	DESCRIPTION
917	AIS MONITORING
918	VOC TRAILER
918A	VOC AIR MONITORING STATION
918B	VOC LAB TRAILER
950	WORK CONTROL TRAILER
951	PROCUREMENT / PURCHASING
952	TRAILER (7-PLEX)
965	SAMPLE PREPARATION LAB
971	HUMAN RESOURCES TRAILER
982	TRAILER
986	PUBLICATIONS & PROCEDURES TRAILER
992	SANDIA CALIBRATION LAB TRAILER
993	SANDIA OFFICES TRAILER
SWR NO.1	SWITCHRACK NO. 1
SWR NO.2	SWITCHRACK NO. 2
SWR NO.3	SWITCHRACK NO. 3
SWR NO.6	SWITCHRACK NO. 6
SWR NO.7,7A,7B	SWITCHRACK NO. 7, 7A, 7B
SWR NO.7C	SWITCHRACK NO. 7C
SWR NO.8	SWITCHRACK NO. 8
SWR NO.9	SWITCHRACK NO. 9
SWR NO.10	SWITCHRACK NO. 10
SWR NO.11	SWITCHRACK NO. 11

4003.8

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Figure 1.2-3b, Legend for Figure 1.2-3

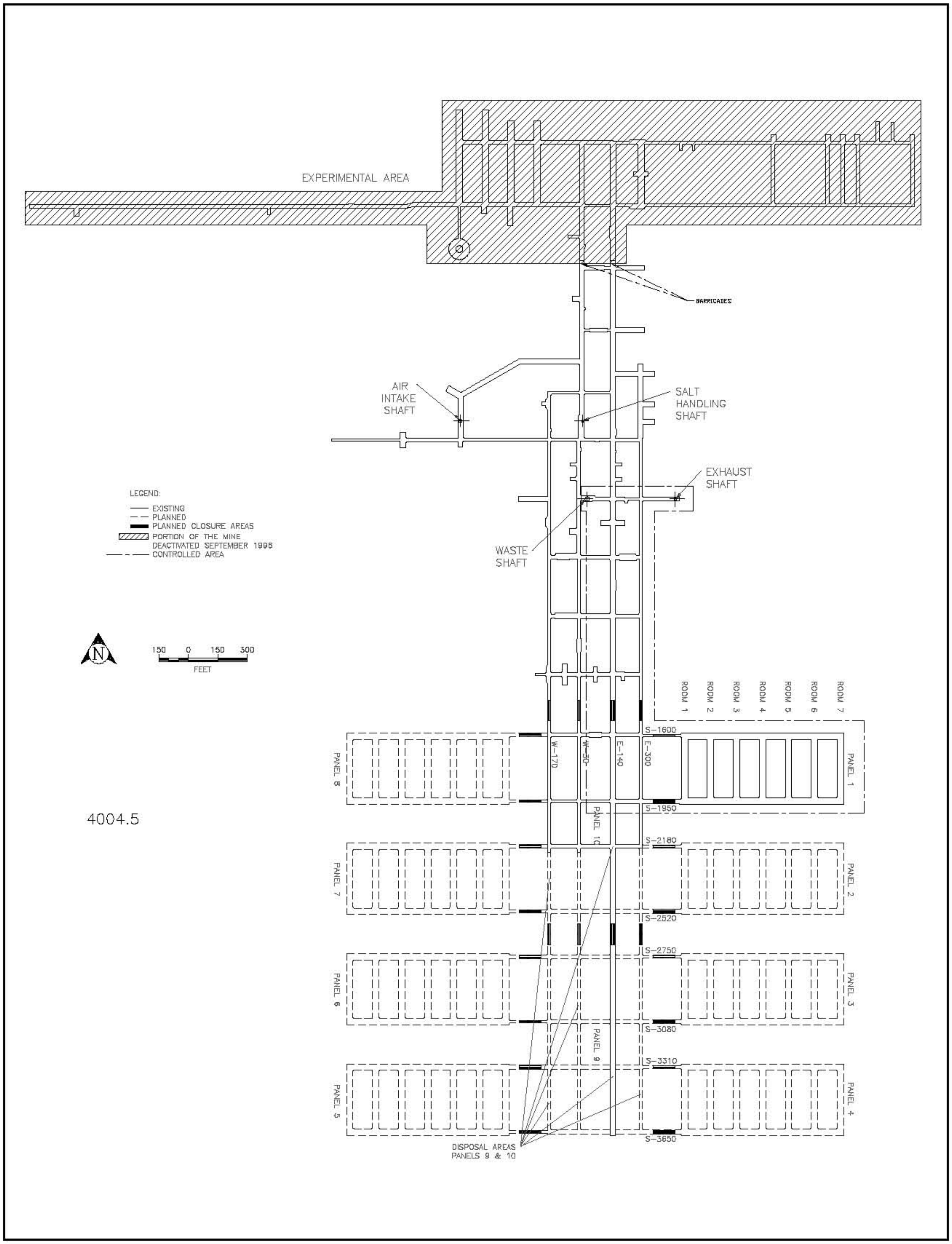

Figure 1.2-4, Underground Subsurface Areas

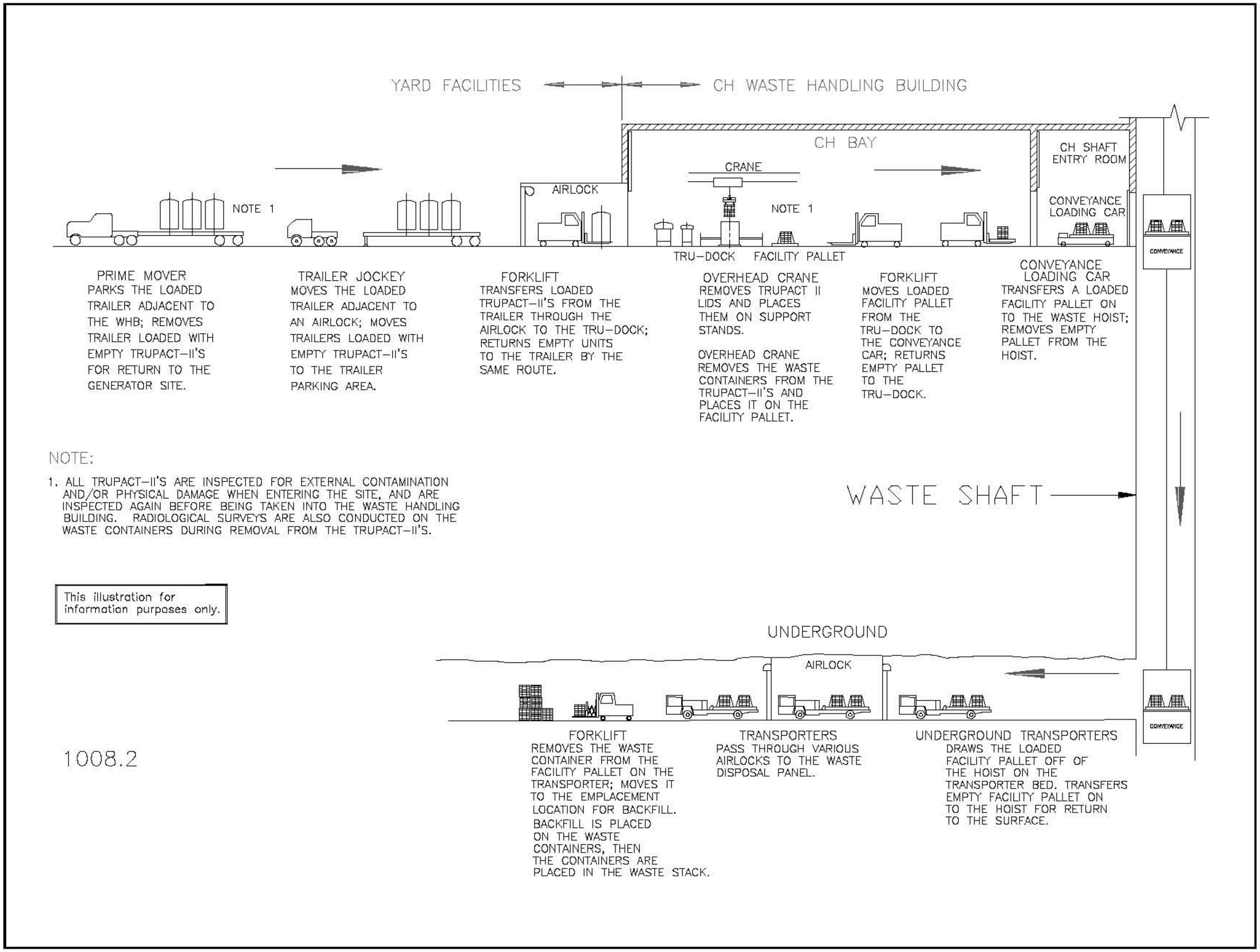

Figure 1.2-5, CH TRU Waste Emplacement Process

1.3 Safety Analysis Overview and Conclusions

1.3.1 Safety Analysis Report Strategy and Approach

The WIPP SAR, originally issued in May 1990 following approval by the Department of Energy, Office of Environmental Restoration and Waste Management (DOE-EM), was prepared to satisfy: (1) the commitments in the Working Agreement for Consultation and Cooperation[1] (WACC) (Article III, Section C and Article IV, Section K, known as the Working Agreement) between the State of New Mexico and the U.S. Department of Energy; and (2) the requirements of DOE Order 5481.1B, Safety Analysis and Review System[2] and DOE Albuquerque Operations Office AL Order DOE-AL 5481.1B.[3]

Since the original approval by DOE-EM, the WIPP SAR has been reviewed and updated: (1) annually in the Fiscal Year (FY)-92 through FY-97 updates; and (2) to ensure compliance with the requirements of DOE Orders 5480.21, Unreviewed Safety Questions,[4] 5480.22, Technical Safety Requirements,[5] 5480.23, Nuclear Safety Analysis Reports,[6] and 5480.24, Nuclear Criticality Safety.[7] Due to the cancellation of DOE Order 5481.1B, the SAR is being maintained per the requirements of DOE Order 5480.23. This SAR represents a statement and commitment by the DOE that the WIPP can be operated safely and at acceptable risk. It also represents the "Final" SAR indicating that the WIPP facility is ready to begin operating versus "Preliminary," which generally refers to a facility in the design or construction stage.

In accordance with the requirements of DOE Order 5480.23,[6] the SAR documents the safety analyses that develop and evaluate the adequacy of the safety bases. The safety bases are defined by DOE Order 5480.23[6] as:

"the combination of information relating to the control of hazards at a nuclear facility (including design, engineering analyses, and administrative controls) upon which DOE depends for its conclusion that activities at the facility can be conducted safely."

This SAR establishes and evaluates the adequacy of the WIPP CH TRU safety bases in response to plant normal and abnormal operations, and postulated accident conditions. The WIPP safety bases analyzed include; (1) the adequacy of the design basis of WIPP CH SSCs, and the application of appropriate engineering codes, standards, and quality assurance requirements, (2) the selection of principal design and safety criteria, (3) the assignment of Technical Safety Requirements (TSRs), and (4) the management, conduct of operations, and institutional dimensions of safety assurance.

Analyses in this SAR address CH TRU waste emplacement operations only. **Existing RH TRU design and operations information were retained for design configuration management purposes only (Changes to RH SSCs are evaluated through the configuration management process, for their impact on CH design and operations as evaluated in this SAR).** RH TRU hazards and accident analyses will be included in a RH TRU Preliminary Safety Analysis Report (currently scheduled for FY-99).

The following provides a summary of the specific issues as they relate to the CH TRU safety bases:

(1) Safety Analysis Report Organization

The WIPP SAR was originally structured to satisfy the specific commitments made in the WACC Agreement.[1] The WACC format is different from the 20 chapter SAR concept of DOE Order 5480.23,[6] and DOE-STD-3009-94.[8] By applying the graded approach concepts as discussed in DOE-STD-3009-94, 10 of the 20 DOE Order 5480.23 chapters were consolidated into other identified chapters. This resulted in a 10 chapter WIPP SAR format that is similar to the WACC Agreement format. This graded approach consolidation and reformatting is consistent with the discussion in DOE Order 5480.23 Attachment 1, Sections 4.f.(1)(c), and 4.f.(3)(d). SAR chapter titles are retitled to follow selected DOE-STD-3009-94 or DOE Order 5480.23 titles and to be consistent with their individual contents. The WIPP SAR format is as follows:

Chapter 1 - Executive Summary
Chapter 2 - Site Characteristics
Chapter 3 - Principal Design and Safety Criteria
Chapter 4 - Facility Design and Operation
Chapter 5 - Hazards and Accident Analysis
Chapter 6 - Derivation of Technical Safety Requirements
Chapter 7 - Radiological and Hazardous Material Protection
Chapter 8 - Institutional Programs
Chapter 9 - Quality Assurance
Chapter 10 - Decontamination and Decommissioning

Table 1.3-1 provides a correlation between the WACC Agreement SAR Format and Content requirements and the WIPP SAR format, and Table 1.3-2 provides a correlation between the SAR topics required by DOE Order 5480.23.

(2) Facility Hazard Categorization

The hazard classification categorization was determined in accordance with DOE-STD-1027-92, *Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports*.[9] A deterministic approach was taken without considering facility segmentation, form location or dispersibility of the material at risk. The material at risk for the determination of the categorization was defined as the maximum radiological contents of a single CH waste container as derived in Chapter 5. The WIPP Facility is classified as a Hazard Category 2 facility based on this single waste container inventory in comparison to the threshold quantities provided in Table A-1 of DOE-STD-1027-92.[9]

(3) Design and Operation

The System Design Descriptions[10] (SDDs) for the WIPP provide the design information for Chapter 3, Principal Design and Safety Criteria, and Chapter 4, Facility Design and Operation. The SDDs provide the most currently available final engineering design information on waste emplacement operations throughout the disposal phase up to the point of permanent closure. Design and operations information were also obtained from the Title 40 CFR 191 Compliance Certification Application for the Waste Isolation Pilot Plant, DOE/CAO-1996-2184, October 1996.[12] Also, the criteria which define the TRU waste to be acceptable for disposal at the WIPP facility are summarized in Chapter 3 based on the *Waste Acceptance Criteria (WAC) for the Waste Isolation Pilot Plant*.[11]

WACC Agreement SAR requirements for Long Term Waste Isolation Assessment, are summarized in Chapter 5. The Long Term Waste Isolation Assessment is covered in the WIPP Compliance Certification Application (CCA).

The systematic evaluation of the human factors associated with the design and operation of the WIPP to meet the requirements of DOE Order 5480.23[6] is incorporated in Chapter 4. The evaluation determined that well established policies and procedures are in place ensuring normal and emergency procedures are implemented, adequate directions have been provided to shift personnel concerning actions to be taken in a potential accident environment, and adequate procedures are available for follow-up response. A detailed summary of the human factors evaluation is provided in Section 1.3.2.2.6.

The WIPP site description in terms of geology, hydrology, meteorology, geography, demography, nearby facilities, and cultural and natural resources are based on information provided in the Title 40 CFR 191 Compliance Certification Application for the Waste Isolation Pilot Plant, DOE/CAO-1996-2184, October 1996.[12]

(4) Hazard Analysis

The WIPP CH TRU handling process was qualitatively evaluated using a Hazard and Operability Study (HAZOP)[13] (Summarized in Appendix C). This systematic approach to hazard analysis was conducted by a leader knowledgeable in the HAZOP methodology and consisted of personnel from various disciplines familiar with the design and operation of the WIPP (HAZOP Team). The HAZOP Team identified deviations from the intended design and operation of the waste handling system that could: (1) result in process slowdown or shutdown, (2) result in worker injury or fatality, and (3) result in the release of waste container radiological and nonradiological materials.

The HAZOP Team assigned a qualitative consequence and frequency ranking for each deviation. A hazard evaluation ranking mechanism utilized the frequency and the most significant consequences to separate the low risk hazards from high risk hazards that may warrant additional quantitative analysis of consequences to the maximally exposed individual (MEI), noninvolved worker, and immediate worker. Based on this ranking approach HAZOP deviations whose combined hazard rank were identified to be of moderate or high risk (see Table 1.3-3) were selected for quantitative analysis in Section 5.2 to: (1) verify and document the basis for the qualitative frequency and consequence assignments in the HAZOP, and (2) identify the need for safety (safety-class or safety-significant) SSCs and Technical Safety Requirements (TSRs).

The HAZOP replaces previous hazards analyses in existing documentation including the Final Environmental Impact Statement (FEIS),[14] Final Supplement Environmental Impact Statement (SEIS),[15] WIPP Fire Hazards and Risk Analysis,[16] and Failure Modes and Effects Analyses (FMEAs), for the purposes of identifying initiating events for quantitative accident analysis in Section 5.2. However, these documents were reviewed to ensure that all hazards associated with CH TRU waste handling were identified in the HAZOP. A detailed summary of the hazards analysis results is provided in Section 1.3.2.2.1.

Since the performance of the HAZOP, an update of the WIPP Fire Hazards Analysis[17] has been performed to meet the requirements of DOE O 420.1.[18] The updated Fire Hazards Analysis confirms the previous evaluation that the frequency of room or structural fire, as an accident in the Waste

Handling Building (WHB) resulting in a direct release of radioactive material from the waste containers engulfed in the fire, is beyond extremely unlikely (<1E-06/yr).

(5) Defense in Depth

A defense in depth section identifies layers of defense against the abnormal and accidental release of radiological and nonradiological hazardous materials. The WIPP approach provides three layers of defense which include conservative design of the facility's SSCs, protection against anticipated operational occurrences and unlikely events, and passive features that may be on line continuously or automatically/manually activated.

The ultimate safety objective of the first, or primary layer of WIPP defense in depth is **accident prevention**. The reduction of risk (as the product of frequency and consequence) to both workers and the public from WIPP CH TRU waste handling and emplacement operations is primarily achieved by reducing the frequency of occurrence of postulated abnormal events or accidents. The conservative design of the facility's SSCs, with operations conducted by trained/qualified personnel to the standards set forth in approved procedures, provides the first layer. Specific preventative measures are identified in Appendix C for each postulated deviation as identified in the HAZOP, and in Table 1.3-3 for each deviation considered for quantitative accident analysis.

The second layer of defense in depth provides protection against anticipated and unlikely operational events that might occur in spite of the protection afforded by the first layer of defense. The second defense layer is characterized by detection and protection systems, and controls that: (1) indicate component, system, or process performance degradation created by compromises of the first layer, and (2) provide adequate mitigation and accommodation of the consequences of those operational accidents which may occur.

The third layer of defense in depth supplements the first two layers by providing protection against extremely unlikely operational, natural phenomenon, and external events. These events represent extreme cases of failures and are analyzed in Section 5.2.3 using conservative assumptions and calculations to assess the radiological and nonradiological effects of such accidents on the MEI, noninvolved worker, and immediate worker to verify that a conservative design bases has been established. A detailed summary of the WIPP defense-in-depth strategy is provided in Section 1.3.2.2.7.

(6) Accident Analysis

The accident analyses utilize currently available DOE Orders, standards and guidance as documented in DOE-STD-3009-94[8] and DOE-STD-1027-92[9], for determination of safety of the public, worker, and the environment. This SAR provides an analysis of the potential hazards that may exist at the WIPP at the level of analytical effort based on the magnitude of the hazards and the complexity of the CH TRU waste operations conducted at the WIPP. The accidents selected for quantitative analysis are considered "Derivative Design Basis Accidents," (DBAs) as defined in DOE Standard 3009-94. These derivative DBAs are used to estimate the response of WIPP SSCs to "the range of accident scenarios that bound the envelope of accident conditions to which the facility could be subjected" in order to evaluate accident consequences.

The principal purpose of the accident analysis is to evaluate the derivative DBAs for the purposes of identifying safety (safety-class or safety-significant) SSCs and TSRs necessary to maintain accident consequences resulting from these derivative DBAs to within the accident risk evaluation guidelines.

For the purposes of establishing safety SSCs, the consequences of these accidents are analyzed to a noninvolved worker conservatively assumed to be 328 ft (100 meters) from each release point, and to the MEI located at the WIPP Exclusive Use Area. An evaluation of operational accidents "beyond" the derivative design basis is conducted by evaluating the accident scenarios in response to the bounding conditions as derived from the WIPP Waste Acceptance Criteria (WAC).[11] For simplicity, the term "derivative" is dropped for the remainder of this chapter; DBA refers to derivative DBAs.

DOE Standard 3009-94 states that use of a lower binning threshold such as 1E-06/yr is generally appropriate, but should not be used as an absolute cutoff for dismissing physically credible low frequency operational accidents without an evaluation of preventative or mitigative features. As such, DBAs identified in this section whose frequency are less than 1E-06/yr (beyond extremely unlikely) are also analyzed quantitatively for the sole purpose of providing perspective on the risk associated with the operation of the facility. The results of these analyses are found in the respective accident evaluation in Section 5.2.3.

An assessment of immediate worker accident consequences is also conducted for the operational <u>waste handling</u> scenarios whose frequency is greater than 1E-06/yr (waste container breaches due to drop or impact), that may be initiated by waste handling equipment failure or directly through human error by a worker performing a waste handling operation. Again, accidents whose frequency are less than 1E-06/yr (beyond extremely unlikely) are also analyzed quantitatively in the respective accident evaluation in Section 5.2.3 for the sole purpose to provide perspective of the risk to the immediate worker associated with the operation of the facility. The immediate worker is that individual directly involved with the waste handling operation for which the accident is postulated. As discussed in Sections 5.1.2.1.2 and 5.1.7, the assessment of immediate worker consequences will ensure that the maximum allowable radionuclide inventory, in conjunction with the other layers of defense in depth, will preclude worker risk from being unacceptable.

The models and assumptions used in the analysis for determining the amount of radioactivity released to the environment and the extent of exposure to the MEI, noninvolved worker, and immediate worker are provided in Section 5.2. Activity releases to the environment are given for each postulated accident. Committed Effective Dose Equivalents (50 yr CEDE) were calculated for what are considered to be hypothetical individuals located: (1) at the WIPP Exclusive Use Area boundary (MEI) and the site boundary (16 Section Boundary), (2) at 328 feet (100 m) from each release point (noninvolved worker), and (3) within the immediate area of the accident (immediate workers). The meteorological conditions under which these doses are evaluated are discussed in Section 5.2.1.

In evaluating hypothetical accidents, a level of conservatism is used in the safety analysis assumptions to provide consequences which result in postulated releases that are overestimated rather than underestimated. The level of conservatism in each of the safety analysis variables is consistent with DOE-STD-3009-94 and its draft appendix. Although draft documents are not necessarily appropriate for reference in this SAR, the draft appendix provides reasonable guidance for consideration and use. The level of conservatism chosen, bounding the full range of possible scenarios (although several of those scenarios are considered to be beyond extremely unlikely), provides reasonable assurance that when considering the variability in waste form, TRU activity content, and radionuclide distributions that: (1) the safety envelope of the facility is defined, (2) the design of the facility is adequate in response to the accident scenarios analyzed, and (3) the Technical Safety Requirements (TSRs) assigned will provide for the protection of the public, the worker, and the environment. A detailed summary of the accident analysis frequency and consequence results is provided in Sections 1.3.2.2.2 and 1.3.2.2.3.

Analyses in this SAR address CH TRU waste emplacement operations only. Existing RH TRU design and operations information were retained for design configuration management purposes only (Changes to RH SSCs are evaluated through the configuration management process, for their impact on CH design and operations as evaluated in this SAR). RH TRU hazards and accident analyses will be included in a RH TRU Preliminary Safety Analysis Report, currently scheduled for FY-1999.

(7) Verification of Design

The hazard and accident analysis results are used to indicate whether safety (safety-class or safety-significant) SSCs are required for the WIPP to prevent or mitigate accidental radiological or nonradiological consequences to the MEI and noninvolved worker to within the risk evaluation guidelines.

Section 5.2.4.1, Evaluation of the Design Basis, discusses in detail: (1) the identification of defense-in-depth SSCs, (2) the evaluation of safety-class and safety significant SSCs, and (3) the applicability of functional and performance requirements and controls. A detailed summary is provided in Section 1.3.2.2.5.

(8) Technical Safety Requirements

Technical Safety Requirements (TSRs) are developed based on the requirements provided in DOE 5480.22,[5] Technical Safety Requirements (TSRs). Based on the requirements and the results of the hazard and accident analysis, no Safety Limits, Operational Limits, or Surveillance Requirements are defined for the WIPP. Supporting the first layer of defense in depth (the prevention of accidents), WIPP TSR Administrative Controls (ACs) are established as follows:

- To maintain the design, quality, testability, inspectability, maintainability, and accessibility of the facility, TSR ACs are required relating to: (1) configuration and document control, (2) maintenance, (3) quality assurance, and (4) geotechnical monitoring. These ACs are important to ensure the frequency of events and the availability of the operating and design conditions remain as analyzed in Section 5.2.3.

- To ensure that the facility operations are conducted by trained and certified/qualified personnel in a controlled and planned manner, TSR ACs are required relating to: (1) facility operations chain of command and responsibilities, (2) facility staffing requirements, (3) procedures, (4) staff qualifications, (5) conduct of operations, and (6) training. These ACs are important to ensuring the low frequency of the accidents analyzed in Section 5.2.3, in particular to those waste handling accidents where human error is the major contributor to the likelihood of the accident initiating event (CH3, CH4, and CH9).

- To ensure that hazards are limited within the bounds assumed in Section 5.2, or that the occurrence of a deviation from the assumed hazard bounds are at an acceptably low frequency, TSR ACs are required relating to: (1) waste characteristics (Waste Acceptance Criteria), (2) waste container integrity, and (3) criticality safety. The TSR AC for waste characteristics limits the radionuclide content of each waste container, restricts the fissile content of the containers, and restricts the presence of waste characteristics unacceptable for management at the WIPP facility. Container integrity ensures the robustness reflected in the waste release analyses, while criticality safety is a designed in-storage and handling configuration that ensures (in conjunction with waste characteristics) that active criticality control is not required.

Supporting the second and third layers of defense in depth, WIPP TSR ACs are identified which establish programs for radiation protection (including radiation monitoring equipment and airborne radioactivity monitoring), and emergency management. Basic elements and requirements defined for TSR AC programs are enforced by the associated implementing WIPP procedures.

(9) Protection of Immediate Workers From Accidents

The HAZOP[13] for the CH TRU Waste Handling System identified a number of waste handling process hazards that could potentially lead to events resulting in immediate worker injury or fatality, or exposure to radiological and nonradiological hazardous materials.

The HAZOP Team identified a significant number of existing preventative safeguards that lower the likelihood of occurrence of each deviation, substantially reducing the risk of injury or fatality to workers. The HAZOP Team concluded, consistent with the first layer of defense in depth, substantial safeguards currently exist at the WIPP to prevent or reduce the likelihood of such deviations from occurring. Identified preventative safeguards generally include the following:

- Facility and equipment design, application of appropriate design classification and applicable design codes and standards,
- Programs relating to configuration and document control, quality assurance, and preventative maintenance and inspection,
- Administrative controls including the WIPP WAC, waste handling procedures and training, and the WIPP Emergency Plan and associated procedures.

Consistent with: (1) Paragraph 6 of Attachment 1 of DOE Order 5480.22, Technical Safety Requirements, (2) the defense-in-depth philosophy discussed in Section 5.1.6, and (3) the philosophy of Process Safety Management (PSM), as published in 29 CFR 1910.119, Process Safety Management of Highly Hazardous Chemicals,[21] reduction of the risk to workers from accidents is accomplished at the WIPP primarily by identifying controls to **prevent the event from happening**. (note: Compliance with 29 CFR 1910.119 is not required by WIPP. However, the WIPP philosophy of reduction of accident risk discussed in this section, is consistent with this standard.) As stated in paragraph 6 of Attachment 1 of DOE Order 5480.22, "The TSRs are not based upon maintaining worker exposures below some acceptable level following an uncontrolled release of hazardous material or inadvertent criticality; rather the risk to workers is reduced through the reduction of the frequency and potential impact of such events."

Consistent with this statement, in conjunction with the defense-in-depth philosophy described in the previous section, total risk is evaluated in this SAR by: (1) performing engineering analyses in the form of event tree/fault tree analysis to identify systems, structures, components, processes, or controls that contribute most to the accident phenomena frequency for the purposes of verifying their adequacy or identifying improvements to reduce the accident frequency and therefore risk, and (2) evaluating human error as an initiating event.

Section 5.2.3 evaluates the accident dose consequences to immediate workers from operational waste container handling accidents whose frequency is greater than 1E-06/yr and may be initiated by waste handling equipment failure or directly through human error by a worker performing a waste handling operation. These accidents include crane failure, and waste container drops or puncture in the Waste Handling Building and the underground. The immediate worker is that individual directly involved with the waste handling operation for which the accident is postulated. This evaluation will ensure that the maximum allowable radionuclide inventory, in conjunction with the other layers of defense in depth, will preclude worker risk from being unacceptable. A detailed summary of the evaluation of the WAC maximum allowable radionuclide inventory is provided in Section 1.3.2.4. Releases from such accidents are conservatively assumed to be instantaneous, and, although procedures dictate that workers exit the area immediately, such accidents present an immediate risk due to the inhalation of airborne radionuclides to the worker performing the waste handling operation.

To evaluate the risk to immediate workers from extremely unlikely operational accidents such as roof fall in the underground and waste hoist failure, the direction of resources in this SAR is focused on the evaluation of system/facility reliability (accident prevention) than on an in-depth evaluation of radiological consequences to an immediate worker and post accident mitigative systems and controls. This evaluation is conducted in the event tree/fault tree analysis in Appendix D, and the accident scenario and evaluation of design adequacy descriptions for each applicable accident in Section 5.2.3.

The risk to workers from extremely unlikely process inherent events such as spontaneous ignition, is a result of the failure of the WIPP WAC to restrict waste elements (such as the presence of pyrophorics) that may cause the initiating event. Again, the direction of resources is focused more on the evaluation of the adequacy of the WAC certification process to prevent this type of accident, rather than on the evaluation of a survivable, specified radiological consequence for which mitigative SSCs or administrative controls may be derived. This evaluation is conducted in the event tree/fault tree analysis in Appendix D, and discussed in Section 5.1.2, and the accident scenario descriptions for CH1 and CH7 in Section 5.2.3. In addition to these fault tree analyses, human error as an initiating event has been evaluated in the WIPP Human Factors Evaluation.

As derived from the WIPP HAZOP, the risk to immediate workers from severe natural phenomenon (design basis earthquake and/or tornado), is dominated by worker fatality due to the energetic phenomenon during the event, as opposed to a specified radiological dose for which additional mitigative SSCs or administrative controls may be derived. This SAR is focused more on the evaluation of the existing facility design when subjected to the severe natural phenomenon (to reduce the likelihood of worker fatality, as well as breach of waste containers), rather than on the evaluation of radiological consequences to an immediate worker. This evaluation is conducted in the accident scenario and evaluation of design adequacy descriptions for each applicable accident in Section 5.2.3.

Due to the importance of these preventative features in the WIPP defense-in-depth safety approach, and for providing worker protection from accidents, TSR ACs are assigned in Chapter 6 and required in the WIPP TSR Document (Attachment 1 to the SAR).

(10) Waste Acceptance Criteria

The WIPP WAC[11] provides the initial set of criteria in Section 3.1 for use in the hazards and accident analyses. The waste accepted for placement in the WIPP facility must conform with the WIPP WAC unless an exception to the WAC has been approved as a result of examination in relation to the SAR. Based on the hazards and accident analyses presented in Chapter 5, specific waste characteristics used in the development of the safety analysis, are required in Chapter 6 to be incorporated as WAC Operations and Safety Requirements. A TSR AC for Waste Characteristics require that the safety analysis criteria be incorporated into the WAC.

Estimates of the radiological waste container inventory for safety analysis calculations were established from a June, 1996 query of the *WIPP Transuranic Waste Baseline Inventory Report* (BIR)[19] database, examining the radionuclide inventory by final waste form, stored waste volume, and waste site. The data reported by the generator sites for 569 individual waste streams was organized by the waste stream, final waste form, and radionuclide concentration (expressed in terms of PE-Ci/equivalent 55 gallon (208 L) drums).

Past WIPP safety analyses established a waste container radionuclide inventory (CI) for use in accident analysis calculations based: (1) strictly on the weapons grade mix (Pu-52 distribution), or (2) based on an average or representative waste container content. Additionally, an arbitrarily chosen radionuclide inventory of 1000 PE-Ci was previously used for bounding accident analysis consequence calculations, and established as the WIPP WAC Pu-239 Equivalent Activity Operations and Safety limit.

Past safety analysis consequence calculations were performed predicated on the WIPP WAC Operations and Safety requirement that waste materials be immobilized if >1% by weight is particulate material <10 microns in diameter, or if > 15 % by weight is particulate material < 200 microns in diameter. However, deletion of this constraint is desirable due to the risk and cost associated with characterizing the size distribution of deposited radionuclide surface contamination on combustible and noncombustible solids. This SAR has evaluated a reasonable range of CIs for "untreated" (not solidified, vitrified, or overpacked) CH TRU waste. Based on a maximum reasonable CI, used in conservative safety analysis with updated airborne release and respirable fractions and the radionuclide limitations for untreated waste derived below, the potential dose consequences due to inhalation by immediate workers, the noninvolved worker, and the MEI from operational accidents with frequencies greater than 1E-06/yr are within the risk evaluation guidelines in Section 5.2.2. As a result, immobilization is no longer required as a WAC criterion.

In conjunction with this goal, the establishment of the radionuclide CI for use in accident analysis calculations must also involve: (1) an evaluation of existing safety analysis orders and guidance documents to establish the appropriate level of conservatism for the CI for safety analysis calculations, (2) consideration of the projected waste inventory in Appendix A and the desire to encompass as much of the Pu-239 and Pu-238 operations waste as possible with the least design or operational impacts to both the waste generator and the WIPP, and (3) evaluation of the existing WAC transportation constraints (nuclear criticality (Pu-239 FGE) and Thermal Power (<40 watts per TRUPACT-II) criteria).

The adequacy of the WIPP facility design, and operational administrative controls (the maximum CI derived below, and elimination of the immobilization requirement as a WAC criterion) is evaluated, based on the accident results in Section 5.2, in detail in Section 5.2.4, and summarized in Section 1.3.2.4.

The source term equation radiological CI used in the accident analyses, is based on the analyses in Section 5.1.2. DOE-STD-3009-94 and its draft appendix state that the source term material at risk (MAR = CI * containers damaged, CD) should "represent a reasonable maximum for a given process or activity, as opposed to artificial maximums unrepresentative of actual conditions." Additionally, Section A.3.1 of the draft appendix to DOE-STD-3009-94, states that documentation may be used to "back off" of bounding estimates of the MAR. Consistent with this statement, based on the data found in Appendix A (as discussed in Section 5.1.2.1), since CH TRU waste operations accidents may result in more than one container damaged in a postulated accident (CD > 1), for safety analysis calculation purposes it is <u>conservatively</u> assumed that one waste container contains the maximum radionuclide inventory and the remaining waste containers each contain an average radionuclide inventory.

As described in Section 5.1.2.1, the maximum drum radionuclide inventory is 80.0 PE-Ci and the maximum SWB radionuclide inventory is 130 PE-Ci. For accident scenarios which involve single waste containers (CD = 1), it is conservatively assumed that the waste container contains the maximum radionuclide inventory. The value CD is determined in each specific accident scenario.

As described in Section 5.1.2.1, the maximum drum radionuclide inventory used to formulate the MAR that is not solidified, vitrified, or overpacked is 80.0 PE-Ci, and the maximum SWB radionuclide inventory that is not solidified, vitrified, or overpacked is 130 PE-Ci. As a defense-in-depth approach to prevent potential unacceptable dose consequences to the MEI, noninvolved worker, and immediate worker (the primary receptor of concern for evaluation of the adequacy of the immobilization criterion) from high PE-Ci untreated waste, the WAC requires that waste containers exceeding the 80 PE-Ci (drums) or 130 PE-Ci (SWBs) values must be overpacked (drum within a SWB or TDOP), or solidified or vitrified (thus immobilized) prior to acceptance at WIPP. Solidification and vitrification both greatly inhibit the release of the waste form should a container be breached during an accident. Overpacking provides an additional barrier that will greatly reduce the frequency of breach during accidents. These two factors, combined with the low percentage of high activity TRU waste volume that currently exists in the inventory, are judged to make the risks associated with high PE-Ci waste forms small compared to those estimated for the "reasonable maximum" MAR.

As discussed above, the WIPP WAC Thermal Power TRUPACT-II requirement limits the maximum total PE-Ci for a TRUPACT-II shipment of Pu-238 waste to approximately 1,117 PE-Ci. Therefore, the WAC Pu-239 Equivalent Activity Operations and Safety maximum allowable waste container radionuclide inventory of 1,100 PE-Ci for overpacked waste and 1,800 PE-Ci for solidified/vitrified is established.

The adequacy of these assumptions and the WIPP CH TRU facility design basis are evaluated in detail based on the accident results in Section 5.2.4, and summarized in Section 1.3.2.4. Receipt of waste for disposal at WIPP that does not meet the applicable Operations and Safety Requirements of the WIPP WAC will first require the performance of an Unreviewed Safety Question Determination (USQD) in accordance with the requirements of DOE Order 5480.21, Unreviewed Safety Questions.[4]

(12) Programs and Procedures

It is the firm commitment of the WIPP management that occupational radiological exposures are kept As Low As Reasonably Achievable (ALARA). This policy, as reflected in administrative programs and procedures established in accordance with 10 CFR 835[22] and the WIPP Radiation Safety Manual,[23] ensures that the safety basis of the WIPP facility will maintain individual occupational radiation exposures to ALARA. As part of normal operations activities at the WIPP, the waste containers (having met the WIPP WAC) are closely inspected and surveyed for radiation, contamination, and damage before transfer to the underground repository. Most significantly, the cleanliness of containers is required to not be in excess of the DOE's free release limits (20 disintegrations per minute (dpm) alpha per 100 cm^2, or 200 dpm beta/gamma per 100 cm^2) prior to shipment from the generator sites. (See Chapter 7 for the basis for radiological and hazardous material protection limits.) WIPP normal operations do not entail any planned or expected releases of airborne radioactive materials which may present an internal occupational radiological hazard to workers, or present a hazard from the airborne pathway to the off-site public.

The institutional programs provide an inclusive strategy to support the safe operation of the facility through implementation of programs and procedures. These programs and procedures fulfill the objectives of radiological protection, project management system, safety management policies and programs, procedures and training, initial testing, in service equipment monitoring, maintenance, operational safety, quality assurance, emergency preparedness, and decontamination/decommissioning.

1.3.2 Safety Analysis Conclusions and Assessment of the CH Design Basis

1.3.2.1 Safety Analysis Overview

Safety analysis was performed for the WIPP to ensure that: 1) potential hazards are systematically identified, 2) unique and representative hazards that may develop into accidents are evaluated, 3) applicable reasonable measures to eliminate, control, or mitigate the accidents are taken, and 4) safety (safety-class or safety-significant) SSCs and accident specific TSRs, based on comparison of accident consequences to the MEI and noninvolved worker to the off-site and on-site risk evaluation guidelines respectively, are identified.

The predicted waste (radioactive/chemical content) to be received in 55-gallon (208 L) drums and SWBs at the WIPP was conservatively estimated based on data[19] from the generating sites, process knowledge, and limiting criteria provided in the WAC.[11] These estimates provided bounding container inventories used in the determination of potential consequences from postulated accidents.

Hazards associated with the facility processes were evaluated through a systematic hazard analysis process. The analysis encompassed the waste receipt, handling and disposal of CH TRU waste in the WIPP. The hazards analysis involved a multi-step process which included: 1) identification of the potential hazards associated with the CH TRU waste handling process, 2) characterization of the waste expected at the WIPP, and 3) a hazard evaluation in the form of a HAZOP[13] for the CH TRU waste handling process. This multi-step process provided a comprehensive examination of the potential hazards which may require quantitative evaluation in the accident analysis.

The major hazard associated with the CH TRU waste handling process is associated with the radiological and nonradiological hazardous materials within the waste containers. Hazards associated with mining operations are considered standard industrial hazards governed by Occupational Safety and Health Administration (OSHA) and Mine Safety and Health Administration (MSHA) regulations and are considered only when they may be an initiating event leading to the accidental release of radiological or nonradiological hazardous materials. Waste handling operations at the WIPP do not involve high temperature and pressure systems, electromagnetic fields or the use of toxic material in large quantities outside of the waste containers. Therefore, for the purposes of establishing an inventory of radiological and nonradiological material, only that material contained in the waste containers was considered, with the dispersive forces being mechanical damage to the containers, or chemical reaction within the containers.

The hazard analysis process identified potential accident scenarios in the categories of: 1) operational accidents (caused by initiators internal to the facility), 2) natural phenomena events (e.g., earthquakes, tornadoes), and 3) external events (caused by man made initiators external to the facility). These potential accident scenarios were then qualitatively ranked in terms of consequence to the public and relative probability to determine unique and representative accidents for further quantitative analysis see Table 1.3-3.

Review of the WIPP Land Management Plan[24] indicates that public access to the WIPP 16-section area up to the exclusive use area shown in Figure 5.2-1 is allowed for grazing purposes, and up to the DOE off limits area" for recreational purposes. In accordance with DOE Order 6430.1A,[20] Section 1300-3.2, the location of the MEI is located at the "closest point of public access," or the DOE “exclusive use area." The location of the MEI is also consistent with guidance for the implementation of 40 CFR 191,[26] Subpart A. Calculations are also performed in Appendix E for a member of the public at the site boundary for reference purposes.

Although prevailing winds are towards the northwest at the WIPP Site, the closest distance to the exclusive use area (without regard to direction) from the exhaust shaft vent and the WHB vent was used in the dose assessment calculations. The closest distance to the exclusive use area boundary from the exhaust shaft vent lies south at approximately 935 ft (285 meters) and the closest distance to the exclusive use area boundary from the WHB lies southeast at approximately 1150 ft (350 meters) (Figure 5.2-2).

The noninvolved worker is assumed to be a worker not directly involved with the waste handling operation for which the accident is postulated. The maximally exposed noninvolved worker is assumed to be located at a distance of 328 ft (100 meters) from each release point due to the restrictions on dispersion modeling used in this safety analysis, at close-in distances.

A summary of the noninvolved worker and MEI radiological and toxicological consequences of analyzed accidents and comparison to risk evaluation guidelines is presented in Tables 1.3-4, 5, 6, and 1.3-7. Off-site risk evaluation guidelines based on ANSI/ANS-51.1[25] are adopted by the WIPP to compare accidental releases from postulated events to the MEI based on estimated frequency of occurrence. Noninvolved worker dose consequences are compared to on-site risk evaluation guidelines developed from available supporting DOE and ANSI guidance. DOE-CAO adopts the same conceptual approach used for the on-site risk evaluation guidelines as for the off-site (public) dose.

However, on-site risk evaluation guidelines are greater than those for the public as DOE-CAO accepts the basic premise that entry onto the site implies acceptance of a higher degree of risk than that associated with the off-site public. This assumption is not considered remiss with regards to safety assurance because the on-site risk evaluation guidelines do not result in any acute health effects noticeable to exposed individuals at frequencies greater than 1.0E-4 event per year and would not result in any acute life-threatening effects.

The methodology for verifying the annual occurrence frequencies, qualitatively estimated in the HAZOP, of operational initiating events is based on the evaluation of process inherent events (spontaneous ignition), equipment failures, and human error. Appendix D contains the detailed assessment of occurrence frequencies of the accidents evaluated in this section. The occurrence frequencies for process events are estimated based on existing references and engineering judgement. The occurrence frequencies for equipment failures and human errors are based on information from other DOE sites with similar operations, and from generic industry data bases when available, applicable, and appropriate.

Equipment failure rates and human error probabilities were combined with WIPP specific operational data to obtain WIPP specific initiating event occurrence frequencies. A detailed event tree/fault tree analysis for each postulated accident is included in Appendix D. The event trees include the analysis of failure of associated preventative and mitigative systems and develops the annual occurrence frequency for both mitigated and unmitigated accident sequences. The annual occurrence frequencies derived from the event tree/fault tree analysis are not intended to represent detailed probabilistic calculations requiring sensitivity or uncertainty analysis. The annual occurrence frequencies derived from the event tree/fault tree analysis are used to provide reasonable assurance that an accident frequency is in a specific qualitative frequency range (i.e. extremely unlikely) or "bin" for the purposes of selecting an appropriate risk evaluation consequence guideline.

For the purposes of establishing safety (safety-class or safety-significant) preventative and mitigative SSCs, an iterative process is performed. The safety (safety-class or safety-significant) iterative process (see Section 3.1.3) initially involves comparing the "unmitigated" accident consequences to the MEI and noninvolved worker (with associated "unmitigated" accident frequency from the event tree analyses in Appendix D) to the off-site and on-site risk evaluation guidelines respectively. The process is continued taking credit for additional preventative/mitigative SSCs until the risk evaluation guidelines are met. Systems required to keep estimated consequences below the risk evaluation guidelines are designated as safety (safety-class or safety-significant) SSCs.

The assessment of the immediate worker accident consequences is based on the evaluation of operational waste handling scenarios (waste container breaches), whose frequency is greater than 1E-06/yr, that may be initiated by waste handling equipment failure or directly through human error by a worker performing a waste handling operation. The immediate worker is that individual directly involved with the waste handling operation for which the accident is postulated. Although procedures dictate that workers exit the area immediately, such accidents present an immediate risk due to the inhalation of airborne radionuclides to the worker performing the waste handling operation. As discussed in Sections 5.1.2.1.2 and 5.1.7, the assessment of immediate worker consequences provides quantitative information in evaluating the adequacy of the WIPP defense-in-depth features (identified in the qualitative HAZOP) in keeping worker dose from accidents as low as reasonably achievable. No current risk evaluation guidelines exist for the assessment of accident consequences to immediate workers. Therefore, in the absence of guidelines, and for conservatism, the on-site radiological guidelines were used as a reference point for the assessment of consequences to immediate workers and the evaluation of the adequacy of the WIPP defense-in-depth features.

1.3.2.2 Safety Analysis Conclusions

1.3.2.2.1 Hazards Analysis Results

The HAZOP Team concluded that:

- Safeguards currently exist at the WIPP to prevent or reduce the frequency of such deviations from occurring. Identified safeguards include facility and equipment design, procedures, training, preventative maintenance and inspection, and administrative controls including the WIPP WAC (see Table 1.3-3, and Appendix C).

- Mitigation exists to reduce the consequences of any postulated deviation to acceptable levels. Identified mitigation includes confinement/ventilation systems and associated HEPA filtration systems (see Table 1.3-3, and Appendix C).

As qualitatively concluded from this HAZOP, the design of the WIPP CH TRU Waste Handling System is sufficient to ensure the safety of the public, workers and the environment. The HAZOP Team identified no substantial recommendations for the WIPP management to consider to reduce the severity or frequency of any of the postulated deviations.

Based on the results of the HAZOP (Table 1.3-3), operational events are binned into two major accident categories (fire and breach of waste container). Since breach of waste containers may occur due to drop or vehicle impact, accidents involving both of these breach mechanisms are evaluated. Accidents involving waste container drops are further evaluated based on the energy involved due to drop height. Due to the differences in release and dispersion mechanisms possible, accidents of each category are evaluated in the above ground and underground areas of the facility. Operational, Natural and External initiating events that require further evaluation as determined by the hazard analysis are listed below:

1. Operational Events

   Fires

   - CH1 Spontaneous Ignition (Drum) in the WHB
   - CH7 Spontaneous Ignition (Drum) in the Underground

   Waste Container Breaches

   - CH2 Crane Failure in the WHB
   - CH3 Puncture of Waste Containers by Forklift in the WHB
   - CH4 Drop of Waste Containers by Forklift in the WHB
   - CH5 Waste Hoist Failure
   - CH9 Drop of Waste Containers by Forklift in the Underground
   - CH11 Underground Roof Fall

2. Natural Events

   - CH6 Seismic Event
   - CH10 Tornado Event

3. External Events

   - CH8 Aircraft Crash

The WIPP is classified as a Hazard Category 2 facility based on bounding estimates of a single waste container inventory of radiological material. The safety analysis utilized this category as a preliminary indication of the level of detail that should be contained in the SAR. In addition to the category, the level of detail was also determined by the level of complexity and potential hazards which may exist during operation of the facility.

1.3.2.2.2 Accident Analysis Frequency Results

As shown in Section 5.2.3, the quantitative frequency analysis for each accident produced the following grouping of accidents:

Unlikely Range (10^{-2}/year > frequency > 10^{-4}/year)

CH2, Crane Failure in the Waste Handling Building (WHB)

CH3, Puncture of Waste Containers in the Waste Handling Building

CH4, Drum Drop in WHB

CH9, Drum Drop in the Underground

Extremely Unlikely Range (10^{-4}/year > frequency > 10^{-6}/year)

CH7, Spontaneous Ignition in the Underground (For the population of drums < 8 PE-Ci)

Beyond Extremely Unlikely Range (10^{-6}/year > frequency)

CH1, Spontaneous Ignition in The Waste Handling Building

CH5, Waste Hoist Failure

CH7, Spontaneous Ignition in the Underground (For population of drums > 8 PE-Ci/drum)

CH11, Roof Fall

For all accidents, the quantitative frequency analysis has verified that the qualitative frequency ranges assigned for these scenarios in the Hazard and Operability Study (HAZOP) were either correctly or conservatively assigned. The unmitigated release frequency is as derived from the event tree (Appendix D) for the associated scenario, and includes: (a) the likelihood of the initiating event, and (b) the conditional likelihood of waste container damage/failure as derived from test data.

Additional quantitative frequency analyses in the form of event/fault tree analyses were performed to identify SSCs, or processes that contribute most to the accident phenomena frequency for the purposes of verifying their adequacy or identifying improvements to reduce the accident frequency and therefore risk to immediate workers (as well as to the MEI and noninvolved worker). Specific accidents evaluated in this manner were: (1) CH1 and CH7, Spontaneous Ignition in the WHB and Underground, (2) CH2, Crane Failure in the WHB, (3) CH5, Waste Hoist Failure, and (4) CH11, Roof Fall in the Underground. With the exception of the Waste Handling Building 6-ton bridge crane (CH2) and spontaneous ignition in drums containing <8 PE-Ci/ drum in the underground (CH7), the event tree/fault tree analyses indicate that the unmitigated frequency of the identified accidents occurring are beyond extremely unlikely (frequency<1E-06/yr).

1.3.2.2.3 Accident Analysis Consequence Results

Based on the CH accident source term and release mechanism analyses presented in Section 5.2.3, for scenarios with a frequency greater than 1E-06/yr (CH2, CH3, CH4, and CH9), the calculated unmitigated accident consequences to the noninvolved worker, and MEI, were found to be well below the selected accident risk evaluation guidelines for the extremely unlikely range (See Tables 1.3-4, 1.3-5, 1.3-6, and 1.3-7). The worst-case consequences are obtained from CH3, with an estimated 3.8 rem (38 mSv) to the noninvolved worker (100 m [328 ft]) (4% of 100 rem [1 Sv] on-site guideline), and 440 mrem (4.4 mSv) to the MEI at the exclusive use area (2% of 25 rem [250 mSv] off-site guideline). It should be noted that: (1) **the MEI unmitigated consequences for scenarios with a frequency greater than 1E-06/yr (CH2, CH3, CH4, and CH9), are also well within the value of 500 mrem (5 mSv) temporary annual dose limit for normal operations derived from DOE Order 5400.5**, and (2) the noninvolved worker unmitigated consequences are within the 5 rem (50 mSv) annual dose limit for workers for normal operations. The unmitigated release frequency for the worst-case consequences is as derived from the event tree (Appendix D) for the associated scenario, and includes: (a) the likelihood of the initiating event, (b) the conditional likelihood of waste container damage/failure as derived from test data, and (c) the conditional likelihood of the worst-case CI from Table A-5 of Appendix A.

Additionally, the accident analysis evaluation of the unmitigated consequences at 100 m confirms the WIPP facility hazard categorization classification as a Hazard Category 2 facility. The calculated 100 m (noninvolved worker) consequences for CH2, CH3, and CH9 exceed the 1 rem criteria established in DOE-STD-1027-92 as the basis for the Category 2 threshold values.

The worst-case consequences to the immediate worker from CH3 are estimated to be 32 rem (320 mSv). No current risk evaluation guidelines exist for the assessment of accident consequences to immediate workers. Therefore, in the absence of guidelines, and for conservatism, the on-site radiological guidelines were used as a reference point for the assessment of consequences to immediate workers and the evaluation of the adequacy of the WIPP defense-in-depth features. The consequences to the immediate worker from CH3 are also well within the on-site risk evaluation guidelines. Therefore, no specific additional worker protection engineering or administrative controls (such as respiratory protection, more stringent maximum waste container inventory, or additional WAC controls such as immobilization), beyond those already qualitatively identified as providing defense-in-depth for the immediate worker, are needed based on the quantitative consequence assessment results.

For scenarios with a frequency less than 1E-06/yr (CH1, CH5, CH7, and CH11), the calculated unmitigated accident consequences to the noninvolved worker, and MEI were also found to be below the selected accident risk evaluation guidelines. The worst-case noninvolved worker and MEI consequences are obtained from CH5, with an estimated 60 rem (600 mSv) to the noninvolved worker (100 m [328 ft]) (60% of 100 rem [1 Sv] on-site guideline) and 9 rem (90 mSv) to the MEI at the exclusive use area (36% of 25 rem [250 mSv] off-site guideline). Risk evaluation guidelines are not identified for events with frequency < 1E-06/yr, however, the 25 rem (250 mSv) risk evaluation guideline for the extremely unlikely range (25 rem siting criteria in DOE Order 6430.1A) is used for evaluating the risk associated with these scenarios. It should be noted that the MEI (exclusive use area) unmitigated consequences for all accidents analyzed, regardless of frequency, were found to be well below 25 rem (250 mSv) risk evaluation guideline.

The worst-case calculated dose to an immediate worker is from CH5 with an estimated 500 rem (5 Sv). Although the immediate worker dose for CH5 exceeds the on-site risk evaluation guidelines for the extremely unlikely range, no specific additional worker protection engineering or administrative controls are identified. The risk associated with this potential exposure is deemed acceptable for the following reasons:

- The conservatism in the risk evaluation guidelines as discussed in Section 5.2.2, as well as the application of the on-site guidelines to the immediate worker,

- Consistent with Section 1.3.1 (9), Protection of Immediate Workers From Accidents, the very low frequency of this scenario is primarily due to the design changes and identification of administrative controls which significantly enhance the system safety and reliability. As identified in EEG-59,[28] the performance of preoperational tests are of paramount importance to system reliability (for the waste hoist, as well as other WIPP SSCs), and as such, is a primary element of the first layer of WIPP defense in depth. Section 8.3.4 discusses the elements of preoperational checks as required by the conduct of operations program, and a TSR AC is derived in Chapter 6 for inclusion in the WIPP Technical Safety Requirements,

- The conservatism inherent in all of the accident analysis source term variables used to estimate the above consequences,

- The existing elements for protection of the worker discussed in detail in Section 5.1.7.

1.3.2.2.4 Comparison to Standards of 40 CFR 61 and 40 CFR 191

As required by Working Agreement for Consultation and Cooperation,[1] signed by the U.S. DOE and the State of New Mexico, July 1981, this SAR will document DOE's ability to comply with the provisions of 40 CFR 191, Subpart A.[26] Paragraph 191.03(b) of 40 CFR 191 Subpart A specifies that the combined annual dose equivalent to any member of the public in the general environment resulting from the discharge of radioactive material and direct radiation from the management and storage of TRU waste shall not exceed 25 millirems (0.25 mSv) to the whole body and 75 millirems (0.75 mSv) to any critical organ. In addition, paragraph 61.92 of 40 CFR 61 Subpart H[27] specifies that emissions of radionuclides to the ambient air from DOE facilities shall not exceed those amounts that would cause any member of the public to receive in any year an effective dose equivalent of 10 mrem/yr (0.10 mSv/yr).

WIPP normal operations do not involve or entail any planned or expected releases of airborne radioactive materials to the workplace or the environment. Waste containers accepted for disposal at the WIPP are required to meet the 10 CFR 835 external contamination limits. To insure compliance, the containers are surveyed both prior to release from the generator sites and as the TRUPACT-II containers are opened at the WIPP. Since radioactive material remains in the waste containers unless an accident occurs, there will be no emissions of radionuclides to the ambient air during normal WIPP waste handling, and the public will not be subjected to direct radiation. Therefore, the public is expected to receive a negligible dose during normal operations. As a result of the above arguments, it may be concluded that the WIPP will be operated in compliance with the release standards of 40 CFR 191 Subpart A[55] and 40 CFR 61 Subpart H.[56] Effluent sampling will be conducted to demonstrate compliance with the annual release limits in those standards.

As shown in this SAR for WIPP, only accidents have the capability of producing a dose to the public. For accidents, 40 CFR 191, Subpart A does not require demonstration of compliance with the release standards. However, the following discussion provides a comparison of the calculated dose consequences to the release standards. As the provisions of 40 CFR 191 Part A guidance impose no restrictions on systems that may be considered in the evaluation of dose to the public, comparison of the WIPP accident analysis results to the standards in paragraph 191.03(b) include the availability and effectiveness mitigation systems that are expected to be in operation should an accident occur. As shown in the accident analysis, these systems are not required in order to meet the safety criteria established by DOE Orders. However, the plant design and operating procedures do provide them for defense in depth and additional assurance that releases that might result from accidents will be as low as reasonably achievable. As shown in Appendix E, based on a decontamination factor of 1E-06 provided by the waste handling building and underground HEPA filtration systems, **the worst-case mitigated accident doses to the maximally exposed individual for all accidents analyzed, regardless of occurrence frequency, will be much less than the annual release limits imposed by 40 CFR 191 Subpart A and 40 CFR 61, Subpart H.**

1.3.2.2.5 Evaluation of the Design Basis

The accident analyses indicate that safety (safety-class or safety-significant) SSCs are not required for the WIPP to mitigate any MEI or noninvolved worker accident radiological and nonradiological consequence to below risk evaluation guideline levels.

Secondary confinement is required to remain functional (following DBAs) to the extent that the guidelines in DOE Order O 420.1,[18] Section 4.1.1.2, Design Requirements, are not violated. The risk evaluation guidelines developed in this safety analysis report were used in the absence of definitive criteria in DOE Order 6430.1A[20] and DOE safety analysis orders or guidance documents for evaluation of secondary confinement. As stated above, the MEI (exclusive use area) and noninvolved worker unmitigated consequences were found to be well below the selected risk evaluation guidelines, including accidents whose frequency is <1E-06/yr, and as such, secondary confinement is not required. However, existing Design Class II and IIIA secondary confinement SSCs, while not <u>required</u> to mitigate the consequences of an accident from exceeding the risk evaluation guidelines, support the second layer of the WIPP defense-in-depth philosophy. A TSR AC is derived in Chapter 6 to ensure that these secondary confinement defense-in-depth SSCs are operating as required for each WIPP mode of operation as specified in Table 6-2.

As discussed in the accident scenarios in Section 5.2.3, there is no credible physical mechanism by which the <u>**operational**</u> accidents analyzed in the WHB or the underground will also disable the respective ventilation or HEPA filtration systems. No releases are postulated requiring ventilation or HEPA filtration for the DBE and DBT scenarios. If waste container breach occurs in the WHB during a credible operational accident (CH2, CH3, CH4), the release to the outside environment is mitigated by the permanently installed continuously on-line two-stage HEPA filter. For credible accident scenarios in the underground (CH9), shift of the underground ventilation system may occur manually (it is assumed that the CMR operator will be notified or be aware of the accident and actuate the shift to filtration), or automatically.

With regard to DBE and DBT scenarios, no release scenarios are expected to be initiated during the DBE or DBT, primarily due to the DBE/DBT design of the WHB structure including tornado doors and specific waste handling equipment such as the WHB 6-ton bridge crane and waste hoist. As such, the WHB ventilation and filtration systems are not required to mitigate the consequences of the DBE or DBT scenarios.

Based on criteria in Chapter 3, Section 3.1.3.2, the factors that lead to designation of a component as Safety Significant are:

- SSCs whose preventive or mitigative function is necessary to keep hazardous material exposure to the noninvolved worker below on-site risk evaluation guidelines,

- SSCs that prevent acute worker fatality or serious injury from hazardous material release that is outside the protection of standard industrial practice, OSHA regulation, or mine safety regulation (MSHA) (e.g. potentially explosive waste containers).

As concluded from the WIPP SAR Section 5.2, Accident Analysis, none of the analyzed scenarios (note: <u>all</u> scenarios are analyzed without regard for occurrence frequency) resulted in noninvolved worker consequences exceeding the on-site risk evaluation guidelines. Therefore, there are no SSCs that are considered Safety Significant due to the need to prevent or mitigate noninvolved worker consequence.

The HAZOP identified two potential scenarios related to WIPP waste handling operations, that could result in worker fatality: (1) potentially explosive waste containers, and (2) waste hoist failure while transporting personnel. With regard to explosive waste containers, SAR Section 5.2.3.1 evaluates such scenarios as beyond extremely unlikely. These events are effectively controlled through rigorous application of the preventive function provided by the WAC administrative control, and as such, preventive or mitigative SSCs are not evaluated or required.

With regard to the waste hoist failure scenario, the consequences involving waste hoist failure while transporting waste containers were evaluated in SAR Chapter 5. Based on the analysis, Safety SSCs are not applicable for that scenario. Personnel **and** waste containers will not be transported simultaneously. Failure of the waste hoist while transporting personnel does not constitute a process related accident involving radioactive materials and as such is considered a standard industrial hazard associated with standard mining operations. Hoisting operations are required to comply with the requirements of 30 CFR 57 and the New Mexico Safety Code for all Mines. As such, Safety Significant SSCs are not designated for failure of the waste hoist while transporting personnel.

Specific SSCs that fulfill a defense-in-depth safety function are: (1) the waste handling equipment such as the WHB 6-ton TRUDOCK bridge crane, adjustable center of gravity lift fixture (ACGLF), electric forklifts, facility pallets (including tie-downs and stretch wrap), waste-hoist, underground transporter, the Loron/BRUDI attachments, and (2) WIPP confinement SSCs including waste containers, Waste Handling Building (WHB) and underground structure, and WHB and underground HVAC and filtration systems. With regard to waste handling equipment, in each instance their reliability and functionality are important to the prevention of damage to the waste containers (first layer of defense in depth). As such, their designation as defense-in-depth SSCs ensures that they are designed, maintained, and operated to prevent failure resulting in an accident. WIPP confinement SSCs (WHB and underground HVAC and filtration systems, and WHB and underground structure) support the second layer of defense in depth. All other WIPP SSCs are considered as balance of plant.

Table 1.3-8 provides a summary of: (1) the preventive and mitigative defense-in-depth safety functions for each accident analyzed quantitatively in Chapter 5 of the SAR, and (2) the safety features that fulfill those safety functions, and whether they are fulfilled by preventive and mitigative SSCs or administrative controls.

DOE-STD-3009-94, requires that for Safety (Safety Class or Safety Significant) SSCs, a SAR define the SSC safety function and functional requirements, performance requirements (system evaluation), and controls (TSRs). Since Safety SSCs are not defined for WIPP, these requirements are not applicable to the WIPP SAR.

Specific WIPP SSCs are classified as Defense-in-Depth SSCs, based on the above functional classification results. Rather than the WIPP SAR specify functional requirements and performance criteria for those defense-in-depth SSCs, the applicable System Design Descriptions (SDDs) describe their intended safety functions, and specify the requirements for design, operation, maintenance, testing, and calibration.

As discussed in detail in SAR Chapter 6, based on application of the criteria in DOE Order 5480.22 for the selection of safety and operational limits, and the fact that Safety Class and Safety Significant SSCs are not selected for WIPP, TSR Safety Limits (SLs), Limiting Conditions for Operation (LCOs), and Surveillance Requirements are not required. TSR ACs assigned for features discussed above that play a role in supporting the WIPP defense-in-depth approach are derived in SAR Chapter 6. Table 6-1 provides a summary of defense-in-depth safety features, applicable TSR controls, and implementing WIPP documents.

Based on the fact that TSR Operational Limits and Surveillance Requirements are not defined for WIPP, operability definitions for Defense-in-Depth SSCs are not required in the SAR. SSCs are required in the TSR to be operated as required during each facility mode as described in Table 6-2, to support the overall WIPP defense-in-depth strategy.

It is therefore concluded from the hazards and accident analyses in this SAR that the design basis of the WIPP CH TRU waste handling system is adequate in response to postulated range of CH TRU normal operations and accident conditions for the facility.

1.3.2.2.6 Evaluation of Human Factors

A systematic inquiry of the importance to safety of reliable, correct, and effective human-machine interactions, considering the mission of the WIPP facility and the physical nature of the radioactive wastes that it will receive was conducted. The specific human errors that can contribute to accidental releases of hazardous materials were evaluated as an integral part of each hypothesized accident. Based on the analysis of those accidents and the discussion below, it can be concluded that the WIPP waste acceptance criteria for transuranic wastes, facility design, and operational controls provide high confidence that all potential releases can be contained with passive safety features that eliminate the need for human actions requiring sophisticated human-machine interfaces.

To provide additional support for the conclusion that no detailed human factor evaluation of human-machine interfaces is required, a scoping assessment of the effectiveness of the human-machine interfaces that support important design functions of the Table 4.1-1 Design Class II and IIIA systems was performed. It can be seen that most of the Design Class II and IIIA WIPP systems and equipment do not require human actions to initiate or sustain their function relative to the release of radiological or nonradiological waste materials. In most cases these functions are accomplished with automatic passive mechanisms designed to provide containment for the waste materials.

Functions allocated to automatic passive mechanisms or automatic active systems may be influenced by human error during maintenance. However, using the graded approach, human-machine interfaces for maintenance activities at WIPP are judged to be adequate because they are deliberate, and there is ample opportunity to discover errors and correct them with no adverse safety consequences.

The ability of the staff to accomplish their responsibilities in potential accident environments was evaluated. The limited magnitude of the hazard and the lack of dispersal driving forces provide very high confidence that the staffing and training presented in those sections will enable the staff to perform their responsibilities in potential accident environments.

The magnitude of hazardous materials that can be involved in an accident leading to a release is very limited. The radioactive material is delivered to the site in closed containers, and the waste handling operations are designed to maintain that integrity throughout the entire process required to safely emplace those containers in the site's underground waste disposal rooms. Inventory limits on individual containers ensure that heat generated by radioactive decay can be easily dissipated by passive mechanisms. Finally, only a limited number of waste containers have the possibility of being breached as a result of any one accident initiating event. As a result, the consequences of unmitigated releases from all accidents hypothesized in Chapter 5, including those initiated by human error, do not exceed the risk evaluation guidelines.

The facility has no complex system requirements to maintain an acceptable level of risk. The facility is designed to minimize the presence and impact of other energy sources that could provide the heat or driving force to disperse hazardous materials. **When something unusual happens during normal operations, such as support systems becoming unavailable, waste handling can be simply stopped and personnel evacuated until an acceptable operating condition is reestablished.**

Should an initiating event occur that breaches the waste containers, the plant design permits the immediate cessation of activity and isolation of the area where the breach occurs. Once isolation is achieved, there is no driving force within the waste or waste handling area that could result in a release of the waste material. Consequently, sufficient time is available to thoroughly plan and prepare for the remediation process prior to initiating decontamination and recovery actions.

Human factors considered in this SAR are limited to that time necessary to properly emplace the transuranic waste designated for disposal at WIPP. The operations will be straightforward, proceduralized, and consistent. Moreover, they will continue for only the period of time needed to complete the disposal process. Once a panel is filled and closed off, the natural properties of the salt and the location of the mine combine to provide passive isolation of the waste from the environment. The potential for human intrusion after the facility closure is beyond the scope of the human factors evaluation considered here.

1.3.2.2.7 Defense in Depth

In spite of the foregoing favorable safety characteristics of the WIPP, a defense-in-depth safety philosophy is employed in establishing the safety commitments and objectives of the WIPP.

The WIPP defense-in-depth safety approach provides layers of defense against release of radiological and nonradiological hazardous materials to the environment. The WIPP approach provides three layers of defense against releases. Each successive layer provides an additional measure of the combined defense strategy. These layers are defined as follows:

1) The ultimate safety objective of the first, or primary layer of WIPP defense in depth is **<u>accident prevention</u>**. The reduction of risk (as the product of frequency and consequence) to both workers and the public from WIPP CH TRU waste handling and emplacement operations is primarily achieved by reducing the frequency of occurrence of postulated abnormal events or accidents. The conservative design of the facility's SSCs, with operations conducted by trained/qualified personnel to the standards set forth in approved procedures, provides the first layer.

   The occurrence frequency for each postulated deviation as identified in the HAZOP, and in Table 1.3-3 for each deviation considered for quantitative accident analysis is primarily derived from process inherent events, equipment failure and human error. To reduce the frequency of equipment failure, the facility design, fabrication, and construction were undertaken in accordance with applicable codes and standards, based on the design classification of SSCs established in Chapter 4. Extensive pre-operational tests were conducted to verify SSCs perform their design function. This is followed up presently by in-service and pre-operational checks and inspections, and preventive maintenance and quality assurance programs. The WIPP employs configuration management change control and modification retest to ensure quality throughout facility life. For hazards associated with underground operations, a substantial array of ground control planning and practices, support systems, instrumentation, monitoring, and evaluation exist to reduce the frequency of potential underground accidents. Technical Safety Requirement (TSR) Administrative Controls (ACs) are assigned in Chapter 6 and required in the WIPP TSR Document (Attachment 1 to the SAR) to ensure that the high level of design is maintained throughout the facility lifetime.

Additionally, as identified in the HAZOP, accident prevention for process inherent events such as spontaneous ignition, is achieved administratively through the WAC (as discussed in detail in Section 5.1.2.2) which restricts waste elements (such as the presence of pyrophorics) which may be initiating events for accidents. In addition, the following provide administrative controls to prevent the risk from postulated accidents from being unacceptable: (1) WAC limits on the radionuclide and fissile content of each waste container, (2) waste container integrity provisions ensure the robustness reflected in the waste container accident release analyses, and (3) criticality safety is a designed in-storage and handling configuration that ensures (in conjunction with waste characteristics) that active criticality control is not required.

Prevention of human error as an initiating event is achieved by the extensive training and qualification programs, operational procedures, and conduct of operations programs. TSR ACs are derived in Chapter 6 and required in the WIPP TSR Document (Attachment 1 to the SAR) to ensure that these programs are maintained, and operations continue to be conducted with highly qualified and trained personnel using current approved procedures.

2) The second layer of defense in depth provides protection against anticipated and unlikely operational events that might occur in spite of the protection afforded by the first layer of defense. The second defense layer is characterized by detection and protection systems, and controls that: (1) indicate component, system, or process performance degradation created by compromises of the first layer, and (2) provide adequate mitigation and accommodation of the consequences of those operational accidents which may occur.

   Specific mitigative features are identified in Appendix C for each postulated deviation as identified in the HAZOP, and in Table 1.3-3 for each deviation considered for quantitative accident analysis. In general, the WHB and underground radiation monitoring systems and HEPA filtration systems, and the WIPP emergency management program provide this layer of defense in depth. In addition, the WIPP Human Factors Evaluation, determined that well established policies and procedures are in place ensuring normal and emergency procedures are implemented, adequate directions have been provided to shift personnel concerning actions to be taken in a potential accident environment, and adequate procedures are available for follow up response. TSR ACs are assigned in Chapter 6 and required in the WIPP TSR Document (Attachment 1 to the SAR) supporting the second level of defense in depth. Programs supporting defense in depth as required by the TSRs, are discussed in detail in Chapters 7, 8, and 9.

3) The third layer of defense in depth supplements the first two layers by providing protection against extremely unlikely operational, natural phenomenon, and external events. These events represent extreme cases of failures and are analyzed in Chapter 5 using conservative assumptions and calculations to assess the radiological and nonradiological effects of such accidents on the public to verify that a conservative design bases have been established. These accidents include sustained waste container internal fire, waste hoist failure, and roof fall in the underground.

TSR ACs assigned for features discussed above that are of major significance to the WIPP defense-in-depth approach are derived in Chapter 6.

1.3.2.3 Analysis of Beyond the Design Basis

1.3.2.3.1 Operational Events

An evaluation of operational accidents "beyond" the derivative design basis accident (BDBA) is conducted to provide perspective of the residual risk associated with the operation of the facility. As discussed in DOE-STD-3009-94, beyond DBAs are simply those accidents with more severe conditions or equipment failure. The operational scenarios analyzed in this section as "beyond the design basis" take into consideration the effect of the WIPP Waste Acceptance Criteria Pu-239 Equivalent Activity, and Thermal Power Criteria on the assumed accident scenario material at risk (MAR) and accident consequences of the most credible accident sequences. Based on the analyses in Section 5.2.3, the operational accident scenarios involving potential consequences to the noninvolved worker, MEI, and immediate worker, whose frequency is greater than 1E-06/yr are: (1) CH2, Crane Failure in the Waste Handling Building (WHB), (2) CH3, Puncture of Waste Containers in the Waste Handling Building, (3) CH4, Drum Drop in WHB, and (4) CH9, Drum Drop in the Underground.

The source term MAR developed in Section 5.2.3 is based on the waste container inventory derived in Section 5.1.2.1.2. The analyses assumed that based on the data in Appendix A, that: (1) one waste container contains a maximum radionuclide inventory, and (2) the remaining waste containers contain an average radionuclide inventory of 8 PE-Ci (Table A-1 lowest bin upper cutoff). The 8 PE-Ci average bounds 86 percent of the volume for all waste forms, including the predominant heterogeneous, uncategorized metal, and combustible waste forms, and bounds over 96 percent of the volume of uncategorized metals, chosen in Section 5.2.1.1 as the waste form for waste container breach/impact analyses. For accident scenarios which involve single waste containers, it was conservatively assumed that the waste container contains a maximum radionuclide inventory.

As discussed in Section 5.1.2.1.2, the WIPP WAC Thermal Power TRUPACT-II requirements, limit the decay heat from all CH-TRU waste to 40 watts per TRUPACT-II. Using the Pu-238 "heat source" distribution in Table A-4 of Appendix A, calculations indicate that the maximum total PE-Ci for a shipment of Pu-238 waste is approximately 1,117 PE-Ci. The analyses of beyond the design basis considers the effect, and thus the residual risk, on the accident consequences evaluated for CH2, CH3, CH4, and CH9 of a hypothetical TRUPACT-II shipment of **untreated** (not solidified or vitrified) Pu-238 waste with each drum at 80 PE-Ci. Receipt of fourteen drums each at 80 PE-Ci is plausible, considering the above thermal wattage limit PE-Ci equivalent of 1,117 PE-Ci (14 drums x 80 PE-Ci approximately equals 1,117 PE-Ci). However, based on the data presented in Table A-5 of Appendix A, as a result of the conditional likelihood of receiving such a shipment, the on-site and off-site risk evaluation guidelines for the extremely unlikely range are used for the consequence evaluation.

As shown in Appendix E Tables E-13, E-14, E-23, E-24, E-29, E-30, E-43, and E-44, the analysis of CH2, CH3, CH4, and CH9 with each damaged drum at 80 PE-Ci, indicates that the highest immediate worker consequences are obtained from CH3 and CH9. The radiological consequences of CH3 are discussed here assuming that each drum involved in the scenario is at 80 PE-Ci. The same assumptions regarding waste form combustible and noncombustible composition, damage ratio, airborne release fraction, and respirable fraction are assumed. Substitution of these values into the consequence calculations for CH3, indicate doses of approximately 12 rem (120 mSv) to the noninvolved worker (12% of the 100 rem (1 Sv) on-site risk evaluation guideline for the extremely unlikely range), and 1.4 rem (14 mSv) (6% of 25 rem (250 mSv) off-site risk evaluation guideline for the extremely unlikely range) to the MEI. The noninvolved worker and MEI doses therefore remain well within the risk evaluation guidelines. The estimated dose to an immediate worker for the CH3 beyond design basis scenario approaches (70 rem [700 mSv]), but does not exceed the on-site risk evaluation guideline of 100 rem (1.0 Sv) for the extremely unlikely range (Table E-62).

Thus, no significant risk is incurred to the immediate worker, noninvolved worker, or MEI considering the beyond design basis most credible operational accident scenarios above involving a maximally loaded TRUPACT-II shipment of untreated Pu-238 heat source waste, with each drum at 80 PE-Ci.

1.3.2.3.2 Natural Phenomenon

As discussed in Section 3.4.3 of DOE-STD-3009, natural phenomenon beyond design basis accidents are defined by a frequency of occurrence less than that assumed for the DBA. Since the DBT is defined with a 10^6 yr return period, and the DBE as a 10^3 yr return period, the most credible beyond DBA natural phenomenon event is an earthquake with a vertical ground acceleration of greater than 0.1 g (considered extremely unlikely).

For the evaluation of beyond the design basis earthquake, DBE SSCs: (1) the WHB structure, and (2) WHB 6-ton bridge crane, are assumed to fail resulting in a release of radioactive material. It is assumed that the bridge crane fails while removing a load from a TRUPACT II (CH2). The WHB structure is also assumed to fail resulting in some damage to the seven facility pallets (196 drums or 28 SWBs) of waste that may be stored in the CH Bay for a period of up to 5 days awaiting transfer to the underground. It is conservatively assumed that one-half of the drums in storage are breached by the falling WHB structure debris, with an DR equivalent to that from the heights associated with drops from the third layer of the waste stack (DR=0.025). This equivalent to 14 times the consequences of the CH2 accident (0.31 rem [3.1 mSv]) or 4.3 rem (43 mSv) to the MEI.

Combining this with the MEI consequences of CH2 (0.3 rem [3 mSv]), the total MEI (exclusive use area) consequence from the postulated beyond DBE is 4.6 rem (460 mSv) (20% of 25 rem [250 mSv] off-site risk evaluation guideline for the extremely unlikely range). The combined consequences to the noninvolved worker are 41 rem (410 mSv) (41% of the 100 rem [1.0 Sv] on-site guideline). Therefore, the radiological risk associated with a greater than 0.1 g earthquake is considered acceptable.

1.3.2.4 Assessment of WIPP Waste Acceptance Criteria (WAC)

1.3.2.4.1 WAC Pu-239 Equivalent Activity Operations and Safety Requirement

Based on the beyond design basis accident analysis results in Section 5.2.4.2 (using conservative assumptions, and in conjunction with elimination of the WAC Revision 4.0,[29] Immobilization Criteria), the estimated radiological consequences for CH3, Puncture in the Waste Handling Building, to the immediate worker, approach the on-site accident risk evaluation guidelines. Therefore, the 80 PE-Ci for drums and 130 PE-Ci for SWBs derived in Section 5.1.2.1.2, are established as the WAC[11] Pu-239 Equivalent Activity Operations and Safety maximum allowable waste container radionuclide inventories for untreated CH TRU waste. The establishment of the 80 and 130 PE-Ci values, provides an defense-in-depth based approach to ensure that the estimated immediate worker accident consequences from untreated CH TRU waste remain acceptable.

Waste containers exceeding these values must be overpacked or treated (solidified, or vitrified) prior to acceptance at WIPP. Such a defense-in-depth approach, focuses on the prevention of potential higher dose consequences to the immediate worker from high PE-Ci untreated waste containers by reducing: (1) the conditional likelihood of waste container breach, and the damage ratio (DR) term of the source term equation (Equation 5-1) for overpacked containers (drums overpacked in SWBs or ten-drum overpacks), and (2) the combined airborne release fraction (ARF) and respirable fraction (RF) for solidified or vitrified waste containers. The CH1 and CH7 sustained internal waste container fire scenarios were evaluated in Section 5.2.3 to be beyond extremely unlikely. Therefore, for the evaluation of solidification, vitrification, and overpacking options, these scenarios are not evaluated.

The WIPP WAC Thermal Power TRUPACT-II requirements, limit the decay heat from all CH-TRU waste to 40 watts per TRUPACT-II. Using the Pu-238 "heat source" distribution in Table A-4 of Appendix A, calculations indicate that the maximum total PE-Ci for a TRUPACT-II shipment of Pu-238 waste is approximately 1,117 PE-Ci.

The acceptability of the WAC Pu-239 Equivalent Activity Operations and Safety maximum allowable waste container radionuclide inventory of 1,100 PE-Ci for overpacked and 1,800 PE-Ci for solidified/vitrified waste, established in Section 5.1.2.1.2 is verified by evaluating the most credible worst-case accident scenarios involving the largest potential consequences for each scenario of interest to the noninvolved worker, MEI, and immediate worker.

However, the consequences of accident scenarios CH2 and CH3 are evaluated in Appendix E (Tables E-9, E-10, E-11, E-12, E-19, E-20, E-21, E-22, E-57, E-58, E-59, and E-60) assuming that the accidents involve highly loaded (1,100 PE-Ci) overpacked (untreated waste within a 55-gallon (208 L) drum overpacked within a SWB or TDOP), and (1,800 PE-Ci) solidified/vitrified waste containers. The consequences of CH2 and CH3 for solidified/vitrified waste, are discussed here due to the differences in breaching mechanisms, and the release fractions identified in Section 5.2.1.1. It is conservatively assumed that seven solidified waste containers are breached as a result of crane failure (CH2), and two are breached as a result of puncture (CH3), with one drum in each scenario at 1,800 PE-Ci. As discussed in Section 5.2.1.1, the damage ratio for CH2 scenario is conservatively assumed to be the same as for untreated waste (DR = 1E-02), and for CH3, DR = 0.01. The ARF x RF for solids that undergo brittle fracture (e.g. aggregate, glass) due to crush-impact forces is given by Equation 5-1 of DOE-HDBK-3010-94.[30] Applying this equation for solidified waste forms to the drop of waste container from heights equal to or less than 3 meters (5 ft $<$ h $\leq$ 10 ft), the calculated ARF x RF = 1.64E-05.

Comparing this factor with that obtained for contaminated noncombustible materials which are subjected to impact and breach of the waste container for solids that do not undergo brittle fracture (Section 5.2.1.1), solidification offers a two order magnitude reduction in respirable airborne radioactive material for the bounding scenarios analyzed in this SAR.

Substitution of these values into the consequence calculations for CH2 and CH3 (Tables E-9, E-11, E-19, E-21, E-57, and E-59), indicate worst-case consequences to the immediate worker for CH3, and are thus summarized here. The doses to the immediate worker (2.1 rem [21 mSv]), noninvolved worker (0.25 rem [2.5 mSv]), and MEI (0.03 rem [0.3 mSv]), are well within the risk evaluation guidelines (for the extremely unlikely range) despite the higher PE-Ci loading. Based on the data presented in Table A-5 of Appendix A, as a result of the conditional likelihood of receiving such a shipment, the risk evaluation guidelines for the extremely unlikely range are used for the consequence evaluation. Therefore, although a higher PE-Ci limit is allowed, the effects of vitrifying, or solidifying waste containers results in a significant reduction in the release of respirable airborne radioactivity and thus risk to the receptors of concern.

To determine the acceptability of overpacking a drum of untreated waste within a SWB or TDOP, the radiological consequences of CH2 and CH3 are again evaluated assuming that multiple drums are breached, one in each scenario at 1,100 PE-Ci (Tables E-10, E-12, E-20, E-22 E-58, and E-60). As discussed in Section 5.2.1.1, the DR for overpacked noncombustible solids (drum within a SWB or

TDOP) for drops less than 10 ft (3 m) is 2.5E-04, and the DR for punctures of heavy waste containers (overpacked noncombustible solids, drum within a SWB or TDOP) is 1E-02. CH3 therefore results in a worst-case source term and as such, the consequences of CH3 are analyzed here. The ARF and RF for noncombustible solids are 1E-03 and 1.0 respectively. Substitution of these values into the consequence calculations for CH3, indicate doses of approximately 9 rem (90 mSv) to the noninvolved worker, 1 rem (10 mSv) to the MEI, and 77 rem (770 mSv) to the immediate worker. The MEI, noninvolved worker, and immediate worker doses therefore remain well within the risk evaluation guidelines (for the extremely unlikely range). Based on the data presented in Table A-5 of Appendix A, as a result of the conditional likelihood of receiving such a shipment, the risk evaluation guidelines for the extremely unlikely range are used for the consequence evaluation.

The WAC Pu-239 Equivalent Activity Operations and Safety limits defined above, when analyzed in conjunction with conservative safety analysis assumptions, and existing stored waste information: (1) provides a reasonable degree of assurance that the safety envelop of the facility has been defined, and (2) ensures that the risk to immediate workers, noninvolved workers, and the MEI remain well within the risk evaluation guidelines.

1.3.2.4.2 WAC Revision 4.0 Immobilization Criteria

Section 3.3.1.6 of WAC Rev.4[29] stated that immobilization will minimize the quantity of radioactive material that is available for dispersion or inhalation in event of the failure of a waste package.

The types of accidents of SAR concern involve contaminated combustible and non-combustible material packaged in robust containers (drums and standard waste boxes), that are opened and/or fail due to drops and/or punctures. The release fractions for drops and/or punctures of drums used in the SAR analyses for the case of surface contamination on solid, noncombustible surfaces are obtained from DOE-HDBK-3010-94.[30] Section 5.1, page 5-4 of DOE-HDBK-3010-94 states, “the airborne release fractions and respirable fractions for these types of accidents are based on reasoned judgement that suspension under these circumstances will be bounded by suspension postulated for debris impacting powders in cans.”

Therefore, in conjunction with the use of conservative waste container radionuclide inventories and damage ratios for heterogeneous or uncategorized metals, conservatism is provided in the calculation of potential radiological consequences from untreated CH TRU waste to the MEI, noninvolved worker, and immediate worker. The estimated consequences were found to be within the on-site and off-site accident risk evaluation guidelines for all receptors of concern. As such, based on the accident consequence analysis in this SAR, no additional criteria are required to immobilize **untreated** (not solidified or vitrified) waste forms (up to a maximum allowable value of 80 PE-Ci for drums and 130 PE-Ci for SWBs) to minimize the quantity of radioactivity available for release.

Section 5.0 of DOE-HDBK-3010-94 discusses the difficulty in characterizing the size distribution of deposited radionuclide contamination. The handbook states that for surface contamination of combustible and noncombustible materials, it is not expected that defensible bases exist for assuming an original source respirable fraction, as the WAC Rev. 4 criteria required. Therefore, (1) since the use of 80 PE-Ci for a drum radionuclide inventory and the inherent conservatism in the derivation and use of the bounding release fractions produce acceptable dose consequences to the worker, noninvolved worker, and MEI, and (2) considering the difficulty in characterizing waste particle size distributions for the waste forms identified in the BIR, the elimination of the WAC immobilization criteria for “untreated waste” up to the values of 80 PE-Ci for drums and 130 PE-Ci for SWBs is warranted.

As discussed in the preceding discussion on maximum allowable waste container radionuclide inventories, however, waste containers exceeding these values will be overpacked, solidified, or vitrified (thus immobilized) as a defense-in-depth approach to limiting the consequences of potential accidents. Immobilization is therefore based on a more readily quantifiable variable (PE-Ci) (i.e., it is measurable and verifiable in all waste forms) than on the percentage of respirable particulates.

References for Section 1.3

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2. U.S. Department of Energy, DOE Order 5481.1B, Safety Analysis and Review System, September 23, 1986.

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6. U.S. Department of Energy, DOE Order 5480.23, Nuclear Safety Analysis Reports, April 30, 1992.

7. U.S. Department of Energy, DOE Order 5480.24, Nuclear Criticality Safety, August 12, 1992.

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10. Waste Isolation Pilot Plant General Plant System Design Description (GPDD), Rev. 2, April 1997.

11. WIPP-DOE-069, TRU Waste Acceptance Criteria for the Waste Isolation Pilot Plant, Rev.5, April 1996.

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14. DOE/EIS-0026, Final Environmental Impact Statement, Waste Isolation Pilot Plant, 2 Vols, U.S. Department of Energy, Carlsbad, N.M., 1980.

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17. DOE-WID-96-2176, Revision 0, WIPP Fire Hazards Analysis Report, August, 1996.

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20. DOE Order 6430.1A, General Design Criteria, Division 13, Special Facilities, April 1989.

21. 29 CFR 1910-119, Process Safety Management of Highly Hazardous Chemicals.

22. 10 CFR 835, Occupational Radiation Protection.

23. WP 12-5, WIPP Radiation Safety Manual, March 1997.

24. DOE/WIPP 94-026, Waste Isolation Pilot Plant Land Management Implementation Plan, August 1994.

25. ANSI/ANS-51.1, American National Standards Institute, Standard Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants.

26. 40 CFR 191, Environmental Radiation Protection for Management and Disposal of Spent Nuclear Fuel, High-Lead and Transuranic Wastes, Subpart A, Environmental Standards for Management and Storage.

27. 40 CFR 61, National Emission Standards for Hazardous Air Pollutants, Subpart H, National Emission Standards for Emissions of Radionuclides Other than Radon from Department of Energy Facilities.

28. EEG-59, An Analysis of the Annual Probability of Failure of the Waste Hoist Brake System at the Waste Isolation Pilot Plant, Environmental Evaluation Group, New Mexico, November, 1995.

29. WIPP-DOE-069, Rev. 4, TRU Waste Acceptance Criteria for the Waste Isolation Pilot Plant, December 1991.

30. DOE-HDBK-3010-94, Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities, December 1994.

Table 1.3-1, Consultation and Cooperation (WACC) Agreement/SAR Correlation 1 of 5

WACC Topic		SAR Section	
Chapter 1 -	**Introduction and General Description**		
1.1	Location	1.1	Facility Background and Mission
1.2	Mission	1.1	Facility Background and Mission
1.3	Organization	1.4	Organizations
1.4	Facilities - both surface and underground	1.2.1	Facility Design
1.5	Operations - including retrieval	1.2.2	Retrieval operations deleted. Disposal-phase operations are discussed with no intent to retrieve.
1.6	Research and Development programs	Deleted - SAR only addresses disposal phase	
Chapter 2 - Site Characteristics			
2.1	Geography and Demography	2.1	Geography and Demography of the Area Around the WIPP Facility.
2.2	Nearby Industrial, Transportation and Military Facilities	2.2	Nearby Industrial, Transportation and Military Facilities
2.3	Meteorology	2.5	Meteorology
2.4	Surface Hydrology	Deleted per CAO direction.	
2.5	Subsurface Hydrology	Deleted per CAO direction.	
2.6	Regional Geology	Deleted per CAO direction.	
2.7	Site Geology	Deleted per CAO direction.	
2.8	Vibratory Ground Motion	2.8	Vibratory Ground Motion
2.9	Surface Faulting	Deleted per CAO direction.	
2.10	Stability of Subsurface Materials and Foundations	Deleted per CAO direction.	
2.11	Slope Stability	2.5.2.5	Topography

Table 1.3-1, Consultation and Cooperation (WACC) Agreement/SAR Correlation 2 of 5

WACC Topic		SAR Section	
Chapter 3 -	**Principal Design Criteria**		
3.1	Definition of Mission	1.1	Facility Background and Mission
	Waste Characterization	5.1.2	CH Waste Characterization
	Repository Functions	3.1	General Design Criteria
	Storage Capacities	3.1.1	TRU Waste Criteria
	Retrievability	Deleted	
	By-Products	3.1.2	Facility By-Products
3.2	Structural and Mechanical Design	3.2	Structural Design Criteria
3.3	Safety Protection Criteria		
	Confinement	3.3.1	Confinement Requirements
	Handling	3.1	General Design Criteria
	Emplacement	3.1	General Design Criteria
	Retrieval	Deleted	
	Fire	3.3.2	Fire Protection
	Explosion	3.3.2	Fire Protection
	Radiological	3.3.3	Radiological Protection
	Criticality	3.3.3.4	Nuclear Criticality Safety
	Mine Safety	3.3.4	Industrial and Mining Safety
3.4	Design Classification	3.1.3	Design Classification of Structures, Systems, and Components
3.5	Decommissioning	3.1.4	Decontamination and Decommissioning
	Decontamination	3.1.4	Decontamination and Decommissioning
	Backfilling	Deleted	
	Sealing	3.1.4	Decontamination and Decommissioning
	Record Maintenance	3.1.4	Decontamination and Decommissioning
	Site Markers	3.1.4	Decontamination and Decommissioning

Table 1.3-1, Consultation and Cooperation (WACC) Agreement/SAR Correlation 3 of 5

WACC Topic		SAR Section	
Chapter 4 - Plant Design			
4.1	Location Details	4.1	Summary Description
4.2	Surface Facilities	4.2.1	Surface Facilities
	Waste Building Handling	4.2.1.1	Waste Handling Building
	Support Functions	4.2.1.2 4.2.1.3 4.2.1.4 4.2.1.5	Exhaust Filter Building Water Pumphouse Support Building Support Structures
4.3	Shafts and Subsurface Facilities	4.2.2 4.2.3	Shaft and Hoist Facilities Subsurface Facilities
	Shafts	4.2.2	Shaft and Hoist Facilities
	Storage	4.2.3	Subsurface Facilities
	Experimental Areas	4.2.3	Subsurface Facilities
4.4	Service and Utility systems	4.3 4.4 4.5 4.6 4.7	Process Description Confinement Systems Safety Support Systems Utility and Auxiliary Systems Radioactive Waste (Radwaste) and Hazardous Waste Management
	Ventilation	4.4.1 4.4.2	Confinement Ventilation Systems
	Electrical	4.6.1	Electrical System
	Fire Protection	4.5.1	Fire Protection System
	Waste Water	4.6.3 4.6.4 4.7	Domestic Water System Sewage Treatment System Radioactive Waste (Radwaste) and Hazardous Waste Management
	Salt Handling	4.3.5	Underground Mining Operations
	Radwaste	4.7	Radioactive Waste (Radwaste) and Hazardous Waste Management
	Transportation	2.2.7	Land Transportation
	Alarms	4.5.2	Plant Monitoring and Communications
	Maintenance	8.3.5	Maintenance Program

Table 1.3-1, Consultation and Cooperation (WACC) Agreement/SAR Correlation 4 of 5

WACC Topic		SAR Section	
	Compressed Air	4.6.2	Compressed Air
	Underground Fuel	4.2.3.1	General Design
4.5	Emplacement and Retrieval	4.3	Retrieval Deleted
4.6	Underground Excavation Equipment	Deleted -	Standard Industrial (MSHA) Hazard
Chapter 5 -	**Process Description**		
5.1	Contact-handled (CH) waste handling	4.3.1	CH TRU Waste Handling System
5.2	Remote-handled (RH) waste handling	4.3.2	RH TRU Waste Handling System
5.3	Experimental handling	Deleted - SAR only addresses disposal phase	
5.4	Plant Generated Radwaste	4.7	Radioactive Waste (Radwaste) and Hazardous Waste Management
5.5	General process		
	Instrumentation	4.5.2	Plant Monitoring and Communications
	Criticality Safety	5.1.4	Nuclear Criticality
	Waste Logging	4.3.4	WIPP Waste Information System
5.6	Underground excavation	4.3.5	Underground Mining Operations
5.7	Control room	4.5.2.1	Central Monitoring System
5.8	Analytical Sampling	7.1.4.2.1 7.2.4	Effluent Sampling/Monitoring and Environmental Monitoring Environmental Monitoring
5.9	Retrievability of All Waste Forms	Deleted	
Chapter 6 -	**Radiation Protection**		
6.1	As low as reasonably achievable (ALARA)	7.1.2 7.2.3.1	ALARA Policy and Program ALARA Policy
6.2	Radiation Sources	7.1.3.1.3.2	Direct Radiation Sources
6.3	Radiation protection	7.1.3	Radiological Exposure Control
6.4	On-site dose assessment	7.1.4.1 7.2.2.2	On-site Dose Assessment On-site Exposure Assessment
6.5	Radiological control program	7.1.1	Radiological Control Program and Organization
6.6	Off-site dose assessment	7.1.4.2 7.2.2.1	Off-site Dose Assessment Off-site Exposure Assessment

Table 1.3-1, Consultation and Cooperation (WACC) Agreement/SAR Correlation 5 of 5

WACC Topic		SAR Section	
Chapter 7 -	**Accident Analysis**		
7.1	Accident classifications	5.2	CH TRU Accident Analysis
7.2	Source terms and analytical methods	5.2	CH TRU Accident Analysis
7.3	Accident descriptions and actual analyses	5.2	CH TRU Accident Analysis
Chapter 8 -	**Long Term Waste Isolation Assessment**	5.5	Long-Term Waste Isolation Assessment
8.1	Identification of potential communication modes	5.5	Long-Term Waste Isolation Assessment
8.2	Modeling methods	5.5	Long-Term Waste Isolation Assessment
8.3	Consequence analyses	5.5	Long-Term Waste Isolation Assessment
Chapter 9 -	**Conduct of Operations**		
9.1	Organizational structure	8.1.3	Organizational Structure, Responsibilities, and Interfaces
9.2	Acceptance tests	8.3.3	Initial Test Program
9.3	Training	8.2.4	Training Program
9.4	Operating procedures	8.2.3	Procedures Program
9.5	Security		Deleted
9.6	Emergencies	8.5	Emergency Preparedness Program
Chapter 10 -	**Operating Limits and Controls**		
10.1	Design limits	Chapter 3	
10.2	Operating limits and surveillance requirements	6.4	Derivation of WIPP TSRs
10.3	Design features	Not Required by 5480.22	
10.4	Administrative controls	6.4.5	Administrative Controls
10.5	Guidelines for the operating organization	6.4.5	Administrative Controls
Chapter 11 -	**Quality Assurance**	Chapter 9 - Quality Assurance	

Table 1.3-2, DOE Order 5480.23/SAR Correlation 1 of 1

DOE Order 5480.23 Topic	SAR Section
Chapter 1 - Executive Summary	Chapter 1 - Executive Summary
Chapter 2 - Applicable Statutes, Rules, and Departmental Orders	Chapter 1 - Executive Summary
Chapter 3 - Site Characteristics	Chapter 2 - Site Characteristics
Chapter 4 - Facility Description and Operation	Chapter 4 - Facility Design and Operation
Chapter 5 - Hazards Analysis and Classification of the Facility	Chapter 5 - Hazards and Accident Analysis
Chapter 6 - Principal Health and Safety Criteria	Chapter 3 - Principal Design and Safety Criteria
Chapter 7 - Radioactive and Hazardous Material Waste Management	Chapter 4 - Facility Design and Operation
Chapter 8 - Inadvertent Criticality Protection	Chapter 5 - Hazards and Accident Analysis
Chapter 9 - Radiation Protection	Chapter 7 - Radiological and Hazardous Material Protection
Chapter 10 - Hazardous Material Protection	Chapter 7 - Radiological and Hazardous Material Protection
Chapter 11 - Analysis of Normal, Abnormal, and Accident Conditions	Chapter 5 - Hazards and Accident Analysis
Chapter 12 - Management, Organization, Institutional Safety	Chapter 8 - Institutional Programs
Chapter 13 - Procedures and Training	Chapter 8 - Institutional Programs
Chapter 14 - Human factors	Chapter 4 - Facility Design and Operation
Chapter 15 - Initial Testing, In service Surveillance, Maintenance	Chapter 8 - Institutional Programs
Chapter 16 - Technical Safety Requirements	Chapter 6 - Derivation of Technical Safety Requirements
Chapter 17 - Operational Safety	Chapter 8 - Institutional Programs
Chapter 18 - Quality Assurance	Chapter 9 - Quality Assurance
Chapter 19 - Emergency Preparedness	Chapter 8 - Institutional Programs
Chapter 20 - Decontamination and Decommissioning	Chapter 10 - Decontamination and Decommissioning

Table 1.3-3, HAZOP Accident Scenario Ranking **Page 1 of 3**

Accident	Scenario	# Node	Deviation	Consequence	Qualitative Consequence Ranking (Table 5.1-6)	Qualitative Frequency Ranking (Table 5.1-5)	Risk	Prevention/Mitigation
CH1	Fire/spontaneous ignition	07 TRUPACT II internal condition	Fire in TRUPACT II	Minor radioactive materials released	**3**	**3**	**9**	Prevention: Type A container, Waste container integrity, QA, Reinstall ICV lid, Building Construction, Stable drum history, TRUPACT II integrity, Vented drums, WAC criteria. Mitigation: Reinstall ICV lid, WHB HEPA filtration and fire suppression systems, Emergency response plan and teams.
CH2	Crane failure/breach	08 Transfer of payload from TRUDOCK to facility pallet	Failure of lifting equipment	Negligible radioactive materials released	**2**	**3**	**6**	Prevention: Type A container, Crane fail safe design, QA, Operator training & qualification, PM program, Procedures, Stretch wrapping, WAC criteria, Hoisting **&** rigging practices, two operators, pre-op checks, waste container integrity. Mitigation: Building Exhaust HEPA filtered, Emergency response plan and teams.
CH2	Crane failure/breach	08 Transfer of payload from TRUDOCK to facility pallet	Failure to secure load	Negligible radioactive materials released	**2**	**3**	**6**	Prevention: Type A container, Fail safe design, QA, Operator training & qualification, Preoperational checks on equipment, PM program, Procedures, Stretch wrapping, WAC criteria, Hoisting & rigging practices, Two operators, Waste container integrity. Mitigation: Building Exhaust HEPA filtered, Emergency Response Plan and teams .
CH3	Fork lift mishap/puncture	09 Transfer facility pallet to conveyance car	Fork lift improper engagement of load	Negligible radioactive materials released	**2**	**3**	**6**	Prevention: Forklift design, QA, Adequate lighting, Operator training & qualification, Pre-op checks, PM program, Procedures, Spotters, WAC criteria, Type A container, Drum integrity, Waste container integrity. Mitigation: Building Exhaust HEPA filtered, Emergency response plan and teams.
CH4	Fork lift mishap/breach	09 Transfer facility pallet to conveyance car	Moving accident	Negligible radioactive materials released	**2**	**3**	**6**	Prevention: Type A container, Operator training & qualification, PM program, Stretch wrapping, Spotters, Tie-down strapping, WAC criteria, Procedures, Pre-op checks, QA, Drum integrity, Waste container integrity. Mitigation: Building Exhaust HEPA Filtered, Emergency Response Plan and Teams.

Table 1.3-3, HAZOP Accident Scenario Ranking

Accident	Scenario	# Node	Deviation	Consequence	Qualitative Consequence Ranking (Table 5.1-6)	Qualitative Frequency Ranking (Table 5.1-5)	Risk	Prevention/Mitigation
CH4	Fork lift mishap/breach	09 Transfer facility pallet to conveyance car	Mislocation on the conveyance car	Negligible radioactive materials released	**2**	**3**	**6**	Prevention: Type A container, QA, Air lock doors interlocked, Local alarms, Operator training & qualification, Restricted access, Robust doors & walls, Stretch wrapping, Spotters, WAC criteria, Procedures, Tie-down strapping, Waste container integrity, PM program, Pre-op checks. Mitigation: HEPA filtration, Emergency response plan and teams.
CH4	Car/breach	10 Transfer conveyance car load onto the waste cage	Moving accident	Negligible radioactive materials released	**2**	**3**	**6**	Prevention: Type A container, QA, Operator training & qualification, Procedures, Stretch wrapping, Spotters, Strapped containers, WAC criteria, Waste container integrity, PM program, Pre-op checks. Mitigation: HEPA filtration, Emergency response plan and teams.
CH5	Hoist failure/breach	11 Waste hoist	Waste hoist drop	Minor radioactive materials released	**3**	**1**	**3**	Prevention: Brake testing, Cable NDT exams, Acoustics exam for failed parts, Control system has elevation check mechanisms, Four independent valve failures required to fail brakes, Brakes checked with full power, Catch gear, Cage fails up, Maintenance procedures & program, Mine rescue equipment, MSHA inspections, Preoperational checks, Qualified personnel, Redundant brakes & controls, Sump under shaft, Six hoist ropes each capable of holding load, inspections, Training and qualification, Weekly inspections, annual vendor inspection, visual inspection of structural steel assemblies, QA. Mitigation: HEPA, Emergency response plan and teams.
CH6	Seismic	15 Natural events	Seismic event	Negligible radioactive materials released	**2**	**1**	**2**	Prevention: Drum integrity, DBE qualified Class II and IIIA SSCs, TRUPACT II integrity, WAC criteria, Type A containers, QA. Mitigation: Shutdown procedure, Emergency response plan and teams.

Table 1.3-3, HAZOP Accident Scenario Ranking **Page 3 of 3**

Accident	Scenario	# Node	Deviation	Consequence	Qualitative Consequence Ranking (Table 5.1-6)	Qualitative Frequency Ranking (Table 5.1-5)	Risk	Prevention/Mitigation
CH7	Spontaneous ignition	27 Drum fire	Drum fire	Minor radioactive materials released	**3**	**3**	**9**	Prevention: Type A container, Waste container integrity, Reinstall ICV lid, Building Construction, Stable drum history, TRUPACT II integrity, Vented drums, WAC criteria. Mitigation: HEPA filtration, , Emergency response plan and teams.
CH8	Crash/fire/breach	16 External events	Aircraft crashes into WHB	Minor radioactive materials released	**3**	**1**	**3**	Prevention: Flight patterns, Remote location. Mitigation: Emergency response plan and teams.
CH9	Fork lift mishap/breach	23 Life of facility	Floor distortion	Negligible radioactive materials released	**2**	**3**	**6**	Prevention: Drift inspections, Floor surveys, MSHA inspections, Forklift design, Type A containers, Procedures, Training. Mitigation: Ventilation flow, Emergency response plan and teams, HEPA filtration.
CH10	Tornado	15 Natural events	Tornado	Negligible radioactive materials released	**2**	**2**	**4**	Prevention: CMR monitors weather conditions, DBT qualified Design Class II and IIIA SSCs, Drum integrity, Procedural guidance for personnel protection, TRUPACT II integrity, WAC criteria, Type A containers. Mitigation: Emergency response plan and teams.
CH11	Roof fall/breach	22 Storage room	Roof collapse during emplacement	Negligible radioactive materials released	**2**	**3**	**6**	Prevention: Inspections & assessments, Ground control, Mine instrumented and monitored, MSHA inspections, Predictive monitoring, Pre-emplacement checks, Type A containers, WAC, procedures, training. Mitigation: Emergency response plan and teams, HEPA filtration.
CH11	Roof fall	23 Life of facility	Roof collapse in life of facility area	Negligible radioactive materials released	**2**	**2**	**4**	Prevention: MSHA inspections, Shift inspections, WAC criteria, Instrumentation and monitoring, Ground control, Bi-monthly visual and instrument inspections, Procedures, Training. Mitigation: Ventilation during emplacement, HEPA filtration, Emergency response plan and teams.

NOTE: Accidents CH5, CH6, CH8, and CH11 were retained in the safety analysis due to being an external event, a natural event, or an event of significant interest.

Table 1.3-4, Summary of Noninvolved Worker and MEI Estimated Radiological Dose and Comparison to Guidelines Page 1 of 1

Accident	Unmitigated Release Freq/yr [1, 2]	On-site /Off-site Guidelines (rem)	Type of Release	Receptor Dose (CEDE-rem)			Receptor Dose % of Guidelines [(Dose/Guidelines)*100]		
				100 m (Noninvolved Worker)	Exclusive Use Area Boundary (MEI)	Site Boundary	100 m (Noninvolved Worker)	Exclusive Use Area Boundary (MEI)	Site Boundary
CH2 Crane Failure in WHB	Extremely Unlikely	100/25	Drums/mitigated	2.7E-06	3.1E-07	2.1E-08	<1%	<1%	<1%
			Drums/unmitigated	2.7E+00	3.1E-01	2.1E-02	2.7%	1.2%	<1%
			SWBs/mitigated	1.1E-06	1.3E-07	8.5E-09	<1%	<1%	<1%
			SWBs/unmitigated	1.1E+00	1.3E-01	8.5E-03	1.1%	<1%	<1%
CH3 Puncture in WHB	Extremely Unlikely	100/25	Drums/mitigated	3.8E-06	4.4E-07	3.0E-08	<1%	<1%	<1%
			Drums/unmitigated	3.8E+00	4.4E-01	3.0E-02	3.8%	1.8%	<1%
			SWBs/mitigated	1.3E-06	1.6E-07	1.1E-08	<1%	<1%	<1%
			SWBs/unmitigated	1.3E+00	1.6E-01	1.1E-02	1.3%	<1%	<1%
CH4 Drop in WHB	Extremely Unlikely	100/25	Drums/mitigated	8.6E-07	1.0E-07	6.8E-09	<1%	<1%	<1%
			Drums/unmitigated	8.6E-01	1.0E-01	6.8E-03	<1%	<1%	<1%
			SWBs/mitigated	1.3E-07	1.6E-08	1.1E-09	<1%	<1%	<1%
			SWBs/unmitigated	1.3E-01	1.6E-02	1.1E-03	<1%	<1%	<1%
CH9 Drop in U/G	Extremely Unlikely	100/25	Drums/mitigated	2.7E-06	4.4E-07	2.1E-08	<1%	<1%	<1%
			Drums/unmitigated	2.7E+00	4.4E-01	2.1E-02	2.7%	1.8%	<1%
			SWBs/mitigated	1.1E-06	1.8E-07	8.4E-09	<1%	<1%	<1%
			SWBs/unmitigated	1.1E+00	1.8E-01	8.4E-03	1.1%	<1%	<1%

Notes: (1) Listed accidents are those whose unmitigated frequency, as derived in Appendix D, is $>10^{-6}$/yr. The consequences of beyond extremely unlikely accidents may be found in the respective accident scenario.

(2) The unmitigated release frequency is as derived from the event tree (Appendix D) for the associated scenario, and includes: (a) the likelihood of the initiating event, (b) the conditional likelihood of waste container damage/failure as derived from test data, and (c) the conditional likelihood of the worst-case CI from Table A-5 of Appendix A.

100 rem = 1 Sv

Table 1.3-5, Summary of Immediate Worker Estimated Radiological Dose and Comparison to Guidelines[1] Page 1 of 1

Accident	No-Mitigation Release Freq/yr[2]	Type of Release	Noninvolved Worker Guidelines (rem)	Receptor Dose (CEDE-rem)	Receptor Dose % of Guidelines
CH2 Crane Failure in WHB	Extremely Unlikely	Drums/no-mitigation	100	1.1E+01	11.0%
		SWBs/no-mitigation	100	4.5E+00	4.5%
CH3 Puncture in WHB	Extremely Unlikely	Drums/no-mitigation	100	3.2E+01	32.0%
		SWBs/no-mitigation	100	1.1E+01	11.0%
CH4 Drop in WHB	Extremely Unlikely	Drums/no-mitigation	100	3.6E+00	3.6%
		SWBs/no-mitigation	100	5.6E-01	<1.0%
CH9 Drop in U/G	Extremely Unlikely	Drums/no-mitigation	100	2.2E+01	22.0%
		SWBs/no-mitigation	100	8.8E+00	8.8%

Notes: (1) Listed accidents are those whose no-mitigation frequency, as derived in Appendix D, is $>10^{-6}$/yr. The consequences of beyond extremely unlikely accidents may be found in the respective accident scenario.

(2) The no-mitigation release frequency is as derived from the event tree (Appendix D) for the associated scenario, and includes: (a) the likelihood of the initiating event, (b) the conditional likelihood of waste container damage/failure as derived from test data, and (c) the conditional likelihood of the worst-case CI from Table A-5 of Appendix A.

1 REM = .01 Sv

Table 1.3-6, Summary of Noninvolved Worker and MEI Nonradiological Concentrations and Comparison to Guidelines Page 1 of 2

Accident	Unmitigated Release Freq./yr [1]	Type of Release	Compound	Concentrations (mg/m³)			% of Guidelines	% of Guidelines
				100 m (Noninvolved Worker)	Exclusive Use Area (MEI)	On-site/Off-site Guidelines (mg/m³) (Table 5.2-2)	100 m (Noninvolved Worker)	Exclusive Use Area (MEI)
CH2 Crane Failure in WHB	Unlikely	Drums/unmitigated	Methylene Chloride	7.3E+00	8.6E-01	21,000/870	<1.0%	<1.0%
			Carbon Tetrachloride	1.4E+01	1.6E+00	1,917/63	< 1.0%	2.50%
			Chloroform	7.10E-01	8.3E-02	5,000/50	< 1.0%	<1.0%
			1,1,2,2-Tetrachloroethane	3.70E-01	4.34E-02	1,505/35	<1.0%	<1.0%
		SWBs/unmitigated	Methylene Chloride	4.2E+00	4.9E-01	21,000/870	<1.0%	<1.0%
			Chloroform	4.1E-01	4.7E-02	5,000/50	<1.0%	<1.0%
			1,1,2,2-Tetrachloroethane	2.12E-01	2.5E-02	1,505/35	<1.0%	<1.0%
			Carbon Tetrachloride	7.7E+00	9.0E-01	1,917/63	<1.0%	1.40%
CH3 Puncture in WHB	Unlikely	Drums/unmitigated	Methylene Chloride	4.2E+00	4.9E-01	21,000/870	<1.0%	<1.0%
			Carbon Tetrachloride	7.8E+00	9.0E-01	1,917/63	<1.0%	1.40%
			Chloroform	4.10E-01	4.7E-02	5,000/50	<1.0%	<1.0%
			1,1,2,2-Tetrachloroethane	2.10E-01	2.5E-02	1,505/35	<1.0%	<1.0%

Table 1.3-6, Summary of Noninvolved Worker and MEI Nonradiological Concentrations and Comparison to Guidelines Page 2 of 2

Accident	Unmitigated Release Freq./yr [1]	Type of Release	Compound	Concentrations (mg/m^3) 100 m (Noninvolved Worker)	Concentrations (mg/m^3) Exclusive Use Area (MEI)	On-site/Off-site Guidelines (mg/m^3) (Table 5.2-2)	% of Guidelines 100 m (Noninvolved Worker)	% of Guidelines Exclusive Use Area (MEI)
		SWBs/unmitigated	Methylene Chloride	8.4E+00	9.8E-01	21,00/870	<1.0%	<1.0%
			Chloroform	8.1E-01	9.5E-02	5,000/50	<1.0%	<1.0%
			1,1,2,2-Tetrachloroethane	4.2E-01	4.9E-02	1,505/35	<1.0%	< 1.0%
			Carbon Tetrachloride	1.6E+01	1.8E+00	1.917/63	<1.0%	2.9%
CH4 Drop in WHB	Unlikely	Consequences same as CH3	-	-	-	-	-	-
CH9 Drop in U/G	Unlikely	Drums/unmitigated	Methylene Chloride	7.3E+00	1.2E+00	21,000/870	<1.0%	<1.0%
			Chloroform	7.1E-01	1.2E-01	5,000/50	<1.0%	<1.0%
			1,1,2,2-Tetrachloroethane	3.7E-01	6.1E-02	1,505/35	<1.0%	<1.0%
			Carbon Tetrachloride	1.36E+01	2.2E+00	1,917/63	<1.0%	3.5%
		SWBs/unmitigated	Methylene Chloride	4.2E+00	6.9E-01	21,000/870	<1.0%	<1.0%
			Chloroform	4.1E-01	6.7E-02	5,000/50	<1.0%	<1.0%
			1,1,2,2-Tetrachloroethane	2.1E-01	3.5E-02	1,505/35	<1.0%	<1.0%
			Carbon Tetrachloride	7.7E+00	1.3E+00	1,917/63	<1.0%	2.1%

NOTE: (1) No credit is taken for mitigation of solid, liquid chemicals or VOCs by HEPA filtration. The unmitigated release frequency is as derived from the event tree (Appendix D) for the associated scenario, and includes: (a) the likelihood of the initiating event, and (b) the conditional likelihood of waste container damage/failure as derived from test data.

Table 1.3-7, Summary of Immediate Worker Estimated Nonradiological Dose and Comparison to Guidelines **Page 1 of 1**

Accident	No-mitigation Freq/yr	Compound	Noninvolved Worker Guidelines (mg/m^3)	Drum Concentration (mg/m^3)	Drum % 0f Guidelines	SWB Concentration (mg/m^3)	SWB % of Guidelines
CH2	Unlikely	Methylene Chloride	21,000	5.49E+00	<1.0%	3.14E+00	<1.0%
		Chloroform	5,000	5.30E-01	<1.0%	3.03E-01	<1.0%
		Carbon Tet	1,917	1.01E+01	<1.0%	5.79E+00	<1.0%
		1,1,2,2-Tetrachlor.	1,505	2.78E-01	<1.0%	1.58E-01	<1.0%
CH3 Puncture in WHB	Unlikely	Methylene Chloride	21,000	3.14E+00	<1.0%	6.27E+00	<1.0%
		Chloroform	5,000	3.03E-01	<1.0%	6.06E-01	<1.0%
		Carbon Tet	1,917	5.79E+00	<1.0%	1.16E+01	<1.0%
		1,1,2,2-Tetrachlor.	1,505	1.59E-01	<1.0%	3.16E-01	<1.0%
CH4	Unlikely	Same as CH3		Same as CH3		Same as CH3	
CH9	Unlikely	Methylene Chloride	21,000	5.99E+01	<1.0%	3.42E+01	<1.0%
		Chloroform	5,000	5.78E+00	<1.0%	3.30E+00	<1.0%
		Carbon Tet	1,917	1.11E+02	5.8%	6.31E+01	3.3%
		1,1,2,2-Tetrachlor.	1,505	3.03E+00	<1.0%	1.73E+00	<1.0%

NOTE: (1) No credit is taken for mitigation of solid, liquid chemicals or VOCs by HEPA filtration. The unmitigated release frequency is as derived from the event tree (Appendix D) for the associated scenario, and includes: (a) the likelihood of the initiating event, and (b) the conditional likelihood of waste container damage/failure as derived from test data.

Table 1.3-8, Summary of Defense-In-Depth Functions and Defense-in-Depth Features Important to Accident Scenarios

Accident	Defense-In-Depth Function	Defense-in-Depth Feature	Type of Feature (SSC or AC)	Type of TSR Control
CH1 Spontaneous Ignition in WHB	• Primary Confinement	• Vented DOT Type A Waste Container or Equivalent	SSC (Passive)	Design Feature/AC 5.9.12
	• Secondary Confinement	• Waste Handling Building Structure (WHB) • WHB CH HVAC System • WHB HEPA Filters	SSC (Passive) SSC (Active) SSC (Passive)	Design Feature/AC 5.1 Design Feature/AC 5.1 Design Feature/AC 5.1
	• Limitations on waste container radionuclide and fissile inventory and waste characteristics	• WIPP Waste Acceptance Criteria	AC	AC 5.9.12
	• Provide facility emergency response to the event (notification, evacuation, direct response)	• WIPP Emergency Management Program	AC	AC 5.9.8
CH2 Crane Failure in WHB	• Primary Confinement	• Vented DOT Type A Waste Container or Equivalent	SSC (Passive)	Design Feature/AC 5.9.12
	• Secondary Confinement	• Waste Handling Building Structure (WHB) • WHB CH HVAC System • WHB HEPA Filters	SSC (Passive) SSC (Active) SSC (Passive)	Design Feature/AC 5.1 Design Feature/AC 5.1 Design Feature/AC 5.1
	• TRUDOCK Crane designed to prevent failure resulting in a dropped load • Adjustable Center of Gravity Lift Fixture (ACGLF) designed to prevent load from swinging	• TRUDOCK Crane Design, ACGLF Design • Configuration Control • Quality Assurance	SSC (Active) AC AC	Design Feature/AC 5.1 AC 5.9.1/5.9.13 AC 5.9.4
	• TRUDOCK Crane maintained to prevent failure resulting in a dropped load • Adjustable Center of Gravity Lift Fixture maintained to prevent load from swinging	• Preventative Maintenance	AC	AC 5.9.3
	• TRUDOCK Crane operated to prevent failure resulting in a dropped load • Adjustable Center of Gravity Lift Fixture designed to prevent load from swinging	• Pre-op Checks/Inspections (Conduct of Ops) • Operator Training and Qualifications • Waste Handling Procedures • Hoisting and Rigging Practices • Operations performed with spotter present • Document Control	AC AC AC AC AC AC	AC 5.1/5.9.7 AC 5.9.6/5.4 AC 5.9.5 AC 5.9.6 AC 5.9.6 AC 5.9.2
	• Limitations on waste container radionuclide and fissile inventory and waste characteristics	• WIPP Waste Acceptance Criteria	AC	AC 5.9.12
	• Provide facility emergency response to the event (notification, evacuation, direct response)	• WIPP Emergency Management Program	AC	AC 5.9.8

Table 1.3-8, Summary of Defense-In-Depth Functions and Defense-in-Depth Features Important to Accident Scenarios Page 2 of 7

Accident	Defense-In-Depth Function	Defense-in-Depth Feature	Type of Feature (SSC or AC)	Type of TSR Control
CH3 Puncture in WHB	• Primary Confinement	• Vented DOT Type A Waste Container or Equivalent	SSC (Passive)	Design Feature/AC 5.9.12
	• Secondary Confinement	• Waste Handling Building Structure (WHB)	SSC (Passive)	Design Feature/AC 5.1
		• WHB CH HVAC System	SSC (Active)	Design Feature/AC 5.1
		• WHB HEPA Filters	SSC (Passive)	Design Feature/AC 5.1
	• Waste Handling Equipment (Forklift and Attachment Design, and Facility Pallet) designed to prevent failure resulting in a punctured waste container	• Forklift and Attachments Design, Facility Pallet Design	SSC (Active)	Design Feature/AC 5.1
		• Configuration Control	AC	AC 5.9.1/5.9.13
		• Quality Assurance	AC	AC 5.9.4
	• Waste Handling Equipment maintained to prevent failure resulting in a punctured waste container	• Preventative Maintenance	AC	AC 5.9.3
	• Waste Handling Equipment operated to prevent failure resulting in a punctured waste container	• Pre-op Checks/Inspections (Conduct of Ops)	AC	AC 5.1/5.9.7
		• Operator Training and Qualifications		
		• Waste Handling Procedures	AC	AC 5.9.6/5.4
		• Hoisting and Rigging Practices	AC	AC 5.9.5
		• Operations performed with spotter present	AC	AC 5.9.6
		• Document Control	AC	AC 5.9.6
	• Limitations on waste container radionuclide and fissile inventory and waste characteristics		AC	AC 5.9.2
		• WIPP Waste Acceptance Criteria	AC	AC 5.9.12
	• Provide facility emergency response to the event (notification, evacuation, direct response)	• WIPP Emergency Management Program	AC	AC 5.9.8

Table 1.3-8, Summary of Defense-In-Depth Functions and Defense-in-Depth Features Important to Accident Scenarios Page 3 of 7

Accident	Defense-In-Depth Function	Defense-in-Depth Feature	Type of Feature (SSC or AC)	Type of TSR Control
CH4 Drop in WHB	• Primary Confinement	• Vented DOT Type A Waste Container or Equivalent	SSC (Passive)	Design Feature/AC 5.9.12
	• Secondary Confinement	• Waste Handling Building Structure (WHB)	SSC (Passive)	Design Feature/AC 5.1
		• WHB CH HVAC System	SSC (Active)	Design Feature/AC 5.1
		• WHB HEPA Filters	SSC (Passive)	Design Feature/AC 5.1
	• Waste Handling Equipment (Forklift and Attachments, Facility Pallet) designed to prevent failure resulting in a dropped waste container	• Forklift and Attachments Design, Facility Pallet Design	SSC (Active)	Design Feature/AC 5.1
		• Configuration Control	AC	AC 5.9.1/5.9.13
		• Quality Assurance	AC	AC 5.9.4
	• Waste Handling Equipment maintained to prevent failure resulting in a dropped waste container	• Preventative Maintenance	AC	AC 5.9.3
	• Waste Handling Equipment operated to prevent failure resulting in a dropped waste container	• Pre-op Checks/Inspections (Conduct of Ops)	AC	AC 5.1/5.9.7
		• Operator Training and Qualifications	AC	AC 5.9.6/5.4
		• Waste Handling Procedures	AC	AC 5.9.5
		• Hoisting and Rigging Practices	AC	AC 5.9.6
		• Operations performed with spotter present	AC	AC 5.9.6
		• Document Control	AC	AC 5.9.2
	• Limitations on waste container radionuclide and fissile inventory and waste characteristics	• WIPP Waste Acceptance Criteria	AC	AC 5.9.12
	• Provide facility emergency response to the event (notification, evacuation, direct response)	• WIPP Emergency Management Program	AC	AC 5.9.8

Table 1.3-8, Summary of Defense-In-Depth Functions and Defense-in-Depth Features Important to Accident Scenarios Page 4 of 7

Accident	Defense-In-Depth Function	Defense-in-Depth Feature	Type of Feature (SSC or AC)	Type of TSR Control
CH5 Waste Hoist Failure	• Primary Confinement	• Vented DOT Type A Waste Container or Equivalent	SSC (Passive)	Design Feature/AC 5.9.12
	• Secondary Confinement	• Underground Ventilation Exhaust System	SSC (Active)	Design Feature/AC 5.1
		• Underground Ventilation Exhaust HEPA Filters	SSC (Passive)	Design Feature/AC 5.1
		• Central Monitoring System (for actuation of underground shift to filtration only)	SSC (Active)	Design Feature/AC 5.1
	• Waste Hoist System designed to prevent failure resulting in an uncontrolled movement of the hoist	• Waste Hoist and Brake System Design	SSC (Active)	Design Feature/AC 5.1
		• Configuration Control	AC	AC 5.9.1/5.9.13
		• Quality Assurance	AC	AC 5.9.4
	• Waste Hoist System maintained to prevent failure resulting in an uncontrolled movement of the hoist	• Preventative Maintenance	AC	AC 5.9.3
	• Waste Hoist System operated to prevent failure resulting in an uncontrolled movement of the hoist	• Pre-op Checks/Inspections (Conduct of Ops)	AC	AC 5.9.7
		• Operator Training and Qualifications	AC	AC 5.9.6/5.4
		• Waste Handling Procedures	AC	AC 5.9.5
		• Hoisting and Rigging Practices	AC	AC 5.9.6
		• Operations performed with spotter present	AC	AC 5.9.6
		• Document Control	AC	AC 5.9.2
	• Limitations on waste container radionuclide and fissile inventory and waste characteristics	• WIPP Waste Acceptance Criteria	AC	AC 5.9.12
	• Provide facility emergency response to the event (notification, evacuation, direct response)	• WIPP Emergency Management Program	AC	AC 5.9.8
CH6 DBE	• Primary Confinement	• Vented DOT Type A Waste Container or Equivalent	SSC (Passive)	Design Feature/AC 5.9.12
	• WHB structure (includes structure and structural components) designed and maintained to prevent failure during a DBE resulting in waste container breach	• Waste Handling Building DBE design	SSC (Passive)	Design Feature/AC 5.1
		• Configuration Control	AC	AC 5.9.1/5.9.13
		• Quality Assurance	AC	AC 5.9.4
		• Preventative Maintenance	AC	AC 5.9.3
	• WHB 6-ton bridge crane and waste hoist designed and maintained to prevent failure during a DBE resulting in waste container breach	• Waste Handling Building 6-ton bridge crane and waste hoist DBE design	SSC (Passive)	AC 5.1
		• Configuration Control	AC	AC 5.9.1/5.9.13
		• Quality Assurance	AC	AC 5.9.4
		• Preventative Maintenance	AC	AC 5.9.3
	• Limitations on waste container radionuclide and fissile inventory and waste characteristics	• WIPP Waste Acceptance Criteria	AC	AC 5.9.12
	• Provide facility emergency response to the event (notification, evacuation, direct response)	• WIPP Emergency Management Program	AC	AC 5.9.8

Table 1.3-8, Summary of Defense-In-Depth Functions and Defense-in-Depth Features Important to Accident Scenarios **Page 5 of 7**

Accident	Defense-In-Depth Function	Defense-in-Depth Feature	Type of Feature (SSC or AC)	Type of TSR Control
CH 7 Spontaneous Ignition in U/G	• Primary Confinement	• Vented DOT Type A Waste Container	SSC (Passive)	Design Feature/AC 5.9.12
	• Secondary Confinement	• Underground Ventilation Exhaust System	SSC (Active)	Design Feature/AC 5.1
		• Underground Ventilation Exhaust HEPA Filters	SSC (Passive)	Design Feature/AC 5.1
		• Radiation Monitoring System (active waste disposal room exit alpha CAM for underground shift to filtration)	SSC (Active)	Design Feature/AC 5.1
		• Central Monitoring System (for actuation of underground shift to filtration only)	SSC (Active)	Design Feature/AC 5.1
	• Limitations on waste container radionuclide and fissile inventory and waste characteristics	• WIPP Waste Acceptance Criteria	AC	AC 5.9.12
	• Provide facility emergency response to the event (notification, evacuation, direct response)	• WIPP Emergency Management Program	AC	AC 5.9.8

Table 1.3-8, Summary of Defense-In-Depth Functions and Defense-in-Depth Features Important to Accident Scenarios Page 6 of 7

Accident	Defense-In-Depth Function	Defense-in-Depth Feature	Type of Feature (SSC or AC)	Type of TSR Control
CH9 Drop in U/G	• Primary Confinement	• Vented DOT Type A Waste Container or Equivalent	SSC (Passive)	Design Feature/AC 5.9.12
	• Secondary Confinement	• Underground Ventilation Exhaust System • Underground Ventilation Exhaust HEPA Filters • Radiation Monitoring System (active waste disposal room exit alpha CAM for underground shift to filtration) • Central Monitoring System (for actuation of underground shift to filtration only)	SSC (Active) SSC (Passive) SSC (Active) SSC (Active)	Design Feature/AC 5.1 Design Feature/AC 5.1 Design Feature/AC 5.1 Design Feature/AC 5.1
	• Waste Handling Equipment (Forklift and Attachments, Facility Pallet) designed to prevent failure resulting in a dropped waste container	• Forklift and Attachments Design, Facility Pallet Design • Configuration Control • Quality Assurance	SSC (Active) AC AC	Design Feature/AC 5.1 AC 5.9.1/5.9.13 AC 5.9.4
	• Waste Handling Equipment maintained to prevent failure resulting in a dropped waste container	• Preventative Maintenance	AC	AC 5.9.3
	• Waste Handling Equipment operated to prevent failure resulting in a dropped waste container	• Pre-op Checks/Inspections (Conduct of Ops) • Operator Training and Qualifications • Waste Handling Procedures • Hoisting and Rigging Practices • Operations performed with spotter present • Document Control	AC AC AC AC AC AC	AC 5.9.7 AC 5.9.6/5.4 AC 5.9.5 AC 5.9.6 AC 5.9.6 AC 5.9.2
	• Limitations on waste container radionuclide and fissile inventory and waste characteristics	• WIPP Waste Acceptance Criteria	AC	AC 5.9.12
	• Provide facility emergency response to the event (notification, evacuation, direct response)	• WIPP Emergency Management Program	AC	AC 5.9.8

Table 1.3-8, Summary of Defense-In-Depth Functions and Defense-in-Depth Features Important to Accident Scenarios Page 7 of 7

Accident	Defense-In-Depth Function	Defense-in-Depth Feature	Type of Feature (SSC or AC)	Type of TSR Control
CH10 DBT	• WHB structure (includes structure and structural components) designed and maintained to prevent failure during a DBT resulting in waste container breach	• Waste Handling Building DBT design • Configuration Control • Quality Assurance • Preventative Maintenance	SSC (Passive) AC AC AC	Design Feature/AC 5.1 AC 5.9.1/5.9.13 AC 5.9.4 AC 5.9.3
	• Limitations on waste container radionuclide and fissile inventory and waste characteristics	• WIPP Waste Acceptance Criteria	AC	AC 5.9.12
	• Provide facility emergency response to the event (notification, evacuation, direct response)	• WIPP Emergency Management Program	AC	AC 5.9.8
CH11 Roof Fall	• Primary Confinement	• Vented DOT Type A Waste Container or Equivalent	SSC (Passive)	Design Feature/AC 5.9.12
	• Secondary Confinement	• Underground Ventilation Exhaust System • Underground Ventilation Exhaust HEPA Filters • Radiation Monitoring System (active waste disposal room exit alpha CAM for underground shift to filtration) • Central Monitoring System (for actuation of underground shift to filtration only)	SSC (Active) SSC (Passive) SSC (Active) SSC (Active)	Design Feature/AC 5.1 Design Feature/AC 5.1 Design Feature/AC 5.1 Design Feature/AC 5.1
	• Underground disposal areas designed to prevent failure resulting in a breached waste container	• Underground Disposal Area Design • Configuration Control • Quality Assurance	SSC (Passive) SSC (Passive) AC	Design Feature/AC 5.1 AC 5.9.1/5.9.13 AC 5.9.4
	• Underground disposal areas maintained to prevent failure resulting in a breached waste container	• Ground Control/Inspections and Assessments • Geomechanical Monitoring	AC AC	AC 5.9.14 AC 5.9.14
	• Limitations on waste container radionuclide and fissile inventory and waste characteristics	• WIPP Waste Acceptance Criteria	AC	AC 5.9.12
	• Provide facility emergency response to the event (notification, evacuation, direct response)	• WIPP Emergency Management	AC	AC 5.9.8

1.4 Organizations

The overall responsibility for the design, construction, operation, and decommissioning of the WIPP rests solely with the DOE. Within the DOE, the Assistant Secretary for Environmental Restoration and Waste Management (EM) is responsible for implementing the radioactive waste disposal policy. In 1993, the DOE Carlsbad Area Office (CAO) was created to be directly responsible for the WIPP Project. The CAO reports programmatically to the DOE-EM and administratively to the DOE-AL.

During the construction phase, DOE-AL contracted with the following organizations to participate in the WIPP Project:

- Sandia National Laboratories (SNL), Department of Waste Management Technology, Albuquerque, New Mexico, to serve as the Scientific Advisor

- Bechtel National Incorporated, Advanced Technology Division, San Francisco, California, to serve as the Architect/Engineer

- Westinghouse Electric Corporation, Waste Isolation Division, Carlsbad, New Mexico, to serve first as the Technical Support Contractor (1978-1985) and later as the Management and Operating Contractor (1985-present)

  NOTE: The U.S. Army Corps of Engineers was the construction manager under provisions of an Interagency Agreement prior to transfer of this responsibility to the Management and Operating Contractor (MOC).

SNL, as the Scientific Advisor, has been responsible for developing the conceptual design of the WIPP facility, and performing the site selection and characterization studies. SNL is also responsible for completing the performance assessment of the WIPP facility in compliance with 40 CFR 191 Subparts B and C.[1]

Bechtel, the Architect/Engineer, was responsible for developing the detailed design of the facility, including construction bid package development and design related geotechnical explorations. Bechtel engaged the services of Rockwell International as consultant for the design of special waste handling equipment.

As the Technical Support Contractor (TSC) (from 1978-1985), Westinghouse was responsible for providing general management and procurement support. In this role, Westinghouse performed technical reviews of the design, prepared the Safety Analysis Report, supported preparation of the Final Environmental Impact Statement, and provided support in operational planning and quality assurance. In 1985, the DOE-AL contracted with Westinghouse to provide management and operating services as the MOC. In this capacity, Westinghouse is responsible for general management and operating services, including operational safety, engineering management, quality assurance and control, project control, construction management, and environmental services. As part of its responsibility as MOC, Westinghouse ensures that all inputs to facility operations are properly reviewed for health, safety, and environmental implications.

The DOE has entered into a formal agreement with the State of New Mexico for the purpose of consultation and cooperation (WACC[2]). This agreement, including its associated working agreement and subsequent modifications, provides a basis for the Governor of New Mexico to exercise the state's right, to comment on and make recommendations regarding the public health and safety aspects of the WIPP Project. The WACC designates key events, sets time frames for review, provides for comments and resolution of comments, and establishes procedures for review of the WIPP Project activities and for resolving conflicts. The WACC agreement also provides a mechanism for conflict resolution.

References for Section 1.4

1. 40 CFR 191, U.S. Environmental Protection Agency, Environmental Radiation Protection for Management and Disposal of Spent Nuclear Fuel, High Level and Transuranic Wastes, Subpart B, Environmental Standards for Disposal, July 1994.

2. Working Agreement for Consultation and Cooperation, signed by the U.S. DOE and the State of New Mexico, July 1981 and subsequent revisions.

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1.5 Statutes, Federal Rules, and DOE Directives Applicable to the Preclosure WIPP CH TRU Waste Operational Safety

Public Law 83-703	Atomic Energy Act of 1954
Public Law 90-148	Clean Air Act
Public Law 91-190	National Environmental Policy Act
Public Law 94-580	Resource Conservation and Recovery Act
Public Law 95-164	Federal Mine Safety and Health Act of 1977
Public Law 96-164	Department of Energy National Security and Military Applications of Nuclear Energy Authorization Act of 1980
Public Law 96-510	Comprehensive Environmental Response, Compensation, and Liability Act
Public Law 102-579	Waste Isolation Pilot Plant Land Withdrawal Act [as amended by Public Law [104-201]
10CFR Part 830	Nuclear Safety Management, April 5, 1994
10CFR Part 835	Occupational Radiation Protection, December 14, 1993
29 CFR Part 1910	Occupational Safety and Health Standards, June 27, 1974
30 CFR Part 57	Safety and Health Standards - Underground Metal and Nonmetal Mines, January 29, 1985
40 CFR Part 61, Subpart H	Subpart H - National Emission Standards for Emissions of Radionuclides Other than Radon from Department of Energy Facilities; 40 CFR Part 61, National Emission Standards for Hazardous Air Pollutants, December 15, 1989
40 CFR Part 191, Subpart A	Subpart A - Environmental Standards for Management and Storage; 40 CFR 191, Environmental Radiation Protection for Management and Disposal of Spent Nuclear Fuel, High-level and Transuranic Radioactive Wastes, November 18, 1985
40 CFR Part 261	Identification and Listing of Hazardous Waste, May 19, 1980
40 CFR Part 262	Standards Applicable to Generators of Hazardous Waste, May 19, 1980
40 CFR Part 264	Standards for Owners and Operators of Hazardous Waste Treatment, Storage, and Disposal Facilities, May 19, 1980
40 CFR Part 265	Interim Status Standards for Owners and Operators of Hazardous Waste Treatment, Storage, and Disposal Facilities, May 19, 1980
40 CFR Part 268	Land Disposal Restrictions, May 19, 1980
40 CFR Part 270	EPA Administered Permit Programs: The Hazardous Waste Permit Program, April 1, 1983
40 CFR Part 280	Technical Standards and Corrective Action Requirements for Owners and Operators of Underground Storage Tanks, September 23, 1988
DOE Order O 414.1	Quality Assurance, November 1998
DOE Order O 420.1	Facility Safety, October 1996
DOE Order O 430.1A	Life-Cycle Asset Management, October 1998
DOE Order 4330.4B	Maintenance Management Program, February 10, 1994
DOE Order 4700.1	Project Management Systems, June 2, 1992 (For reference only, superceded by DOE O 430.1A)
DOE Order 5000.3B	Occurrence Reporting and Processing of Operations Information, January 19, 1993
DOE Order 5400.1	General Environmental Protection Program, June 29, 1990
DOE Order 5400.4	Comprehensive Environmental Response, Compensation, and Liability Act Requirements, June 6, 1989
DOE Order 5400.5	Radiation Protection of the Public and the Environment, January 7, 1993

DOE Order 5440.1E	National Environmental Policy Act Compliance Program, November 10, 1992
DOE Order 5480.4	Environmental Protection, Safety, and Health Protection Standards, January 7, 1993
DOE Order 5480.18B	Nuclear Facility Training Accreditation Program, August 31, 1994
DOE Order 5480.19	Conduct of Operations Requirements for DOE Facilities, May 18, 1992
DOE Order 5480.20A	Personnel Selection, Qualification, Training Requirements for DOE Nuclear Facilities, November 15, 1994
DOE Order 5480.21	Unreviewed Safety Questions, May 12, 1994
DOE Order 5480.22	Technical Safety Requirements, September 15, 1992
DOE Order 5480.23	Nuclear Safety Analysis Reports, April 30, 1992
DOE Order 5500.1B	Emergency Management System, April 30, 1991
DOE Order 5500.2B	Emergency Categories, Classes, and Notification and Reporting Requirements, February 27, 1992
DOE Order 5500.3A	Planning and Preparedness for Operational Emergencies, February 27, 1992
DOE Order 5500.3B	Occurrence Reporting and Processing of Operations Information, January 19, 1993
DOE Order 5500.7B	Emergency Operation Records Protection Program, October 23, 1991
DOE Order 5500.10	Emergency Readiness Assurance Program, February 27, 1992
DOE Order 5820.2A	Radioactive Waste Management, September 1988
DOE Order 6430.1A	General Design Criteria, 1989 (For reference only, superceded by DOE O 420.1 and DOE O 430.1A)

Note: Conversion to, and implementation of, selected applicable DOE O series Orders are not required until inclusion into Managing and Operating Contractor contracts. As such, demonstration of compliance with applicable Orders, replacing any listed above, will be included in the appropriate Annual SAR Update when the Orders become effective and are implemented at WIPP.