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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 be the 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.

As a result of successful tests, the EPA and the New Mexico Environmental Department (NMED) authorized operations. WIPP started receiving CH TRU and TRU mixed waste in 1999. WIPP is currently scheduled to receive remote-handled (RH) TRU mixed waste (hereafter referred to as RH TRU waste or RH waste) in the second quarter of FY05.

The disposal phase is currently scheduled to last 35 years,[6, 7] and will consist of receipt, handling, and emplacing TRU waste in the repository for disposal, and will end when the design capacity of the planned 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, post closure environmental monitoring, 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 repository functions as designed, and the potential for future, inadvertent human intrusion is reduced to a level that renders such intrusion unlikely.

This Preliminary Safety Analysis Report (PSAR) documents the safety analyses that develop and evaluate the adequacy of the WIPP RH TRU safety basis 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 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 5.3, 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 1979.

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

3. U.S. Department of Energy, 55 FR 256892, Record of Decision, Waste Isolation Pilot Plant, June 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, The National Transuranic Waste Management Plan, U.S. Department of Energy, Carlsbad Field Office.

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

9. EPA (U.S. Environmental Protection Agency), 1998. Criteria for the Certification and Re-certification 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 1998, Radiation Protection Division, Washington, D.C.

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1.2 Facility Overview

1.2.1 Facility Location

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 16 sections of land set aside for the WIPP includes an area of 10,240 acres (4144 hectares). The WIPP is located in an area of low population density with fewer 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 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[1], no mineral resource development is allowed within the WIPP Site Boundary (with the exception of existing leases).

1.2.2 Facility Design

The WIPP is designed to receive and handle a maximum of 10,000 ft^3/yr (283 m^3/yr) RH TRU waste. The WIPP facility is designed to have a total 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 containers which are placed inside shielded road casks then transported to the WIPP facility. 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[2] as a total volume. In addition, the WIPP Land Withdrawal Act of 1992[1] limits the total RH TRU activity to 5.1 E 06 Curies.

RH TRU wastes will be disposed in the 100 acre (40.5 hectares) disposal area on a horizon located 2,150 ft (655 m) 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 placement of CH and RH 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 functional areas: surface structures, shafts, and subsurface structures as shown in Figure 1.2-2. The WIPP surface structures (Figure 1.2-3a) 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 current design of the RH TRU waste handling area includes the following; a RH Bay for cask receiving and preparation; the Cask Unloading Room (CUR) where the 72B cask is prepared for and lowered into the Transfer Cell and where the waste drums are removed from the 10-160B cask and lifted into the Hot Cell; the Hot Cell where radiological surveys on each drum and identity verification of each drum is performed before being placed into facility canisters (max of three drums per canister) and where the facility canisters are lowered into a shielded insert in the Transfer Cell; and the Transfer Cell where the canister in a 72B cask or the facility canister in a shielded insert are transferred (raised) into the facility cask. During the lift from the 72B cask, radiological surveys and identity verification as well as a physical inspection is performed on the 72B canister; the facility Cask Loading room where the facility cask is loaded with either a 72B canister or a facility canister and then positioned on the waste hoist conveyance for transfer to the underground.

The vertical shafts extending from the surface to the underground horizon (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 m) 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 surrounding water-bearing formation. The waste shaft is located between the CH TRU and RH TRU areas in the WHB. It is nominally 19 feet (5.8 m) 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 (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 ft (10 m) wide, 13 ft (4 m) high, and 300 ft (91.5 m) long. The RH waste canisters are placed in 14 ft (4.3 m) long horizontal bore holes on 30 in (0.8 m) centers in the walls (ribs) of the disposal rooms. The disposal rooms are separated by pillars of salt 100 ft (30.5 m) wide and 300 ft (91.5 m) long. Panel entries at the end of each of these disposal rooms are also 33 ft (10 m) wide and 13 ft (4 m) high and will be used for waste disposal, except for the first 200 ft (61 m) from the main entries which are 22 ft (6.7 m) wide by 14 ft (4.3 m) high. This first 200 ft (61 m) will be used for installation of panel closure systems.

1.2.3 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. A pictorial view of the 72B RH TRU waste handling process is shown in Figure 4.3-1, while the 10-160B waste handling process is shown in Figure 4.3-2.

RH TRU waste will be shipped to the WIPP in Nuclear Regulatory Commission (NRC) certified shipping packages (72B and 10-160B road casks). The RH waste handling process begins when the truck arrives at the WIPP gate. After the RH TRU road cask is surveyed for contamination and shipping documentation confirmed, the loaded road cask trailer is staged in the parking lot adjacent to the RH entrance to the WHB.

The loaded trailer is moved into the WHB RH bay. Impact limiter(s) are removed before the 72B and/or 10-160B road cask is transferred to their respective road cask transfer car. The outer containment vessel (OCV) lid of the 72B road cask is removed or the bolts loosened on the primary lid of the 10-160B road cask and initial waste handling activities are performed before the road cask is transported to the CUR. The CUR crane lifts the loaded 72B road cask from the road cask transfer car and lowers it into the Transfer Cell onto the shuttle car. In the Transfer Cell, the inner containment vessel lid of the 72B road cask is removed, the identity of the waste cannister is confirmed and remote radiological surveys are performed. The 10-160B road cask is moved to the CUR where the payload of ten 55-gal drums of RH waste is lifted into the Hot Cell by the Hot Cell 15-ton crane. Radiological surveys are performed on each drum and the identity of each drum is confirmed. Three RH waste drums are loaded into a WIPP facility canister which is lowered into the shielded insert, installed on the shuttle car, located in the Transfer Cell. The grapple hoist in the Facility Cask Loading Room lifts the waste canister into the facility cask. The loaded facility cask is moved into the waste shaft's hoist cage for transfer to the disposal horizon.

At the disposal horizon, the facility cask is transported by a forklift into the waste disposal room. In the disposal room, the waste canister is removed from the facility cask, emplaced in a horizontal borehole and then a shield plug is installed in the borehole. Details of the RH waste operations are provided in Section 4.3.

The RH waste, consisting of radiologically hazardous and chemically hazardous material, received for placement in the WIPP facility must conform with the RH Waste Acceptance Criteria (WAC). Draft criteria have been prepared (RH Draft WAC [5]) and are currently being reviewed. These criteria will be formalized prior to receipt of RH waste. The purpose of the RH WAC is to summarize the waste acceptance criteria that RH-TRU waste must meet before it can be transported to, managed, and disposed of at the WIPP. These criteria serve as the DOE's primary directive for ensuring that TRU waste is managed and disposed of in a manner that protects worker and public health and safety and the environment.

The operational philosophy at the WIPP facility is to start radiologically clean and stay radiologically clean. As a canister is removed from the 72B road cask or drums removed from the 10-160B road cask, contamination surveys, damage inspections, and identity verifications are performed. If any identity discrepancies are found and/or any levels of radiation, contamination, or significant damage in excess of acceptance criteria are found, actions will be taken in accordance with approved procedures. Also, any local area of contamination may be decontaminated prior to continuation of the waste handling process.

References for Section 1.2

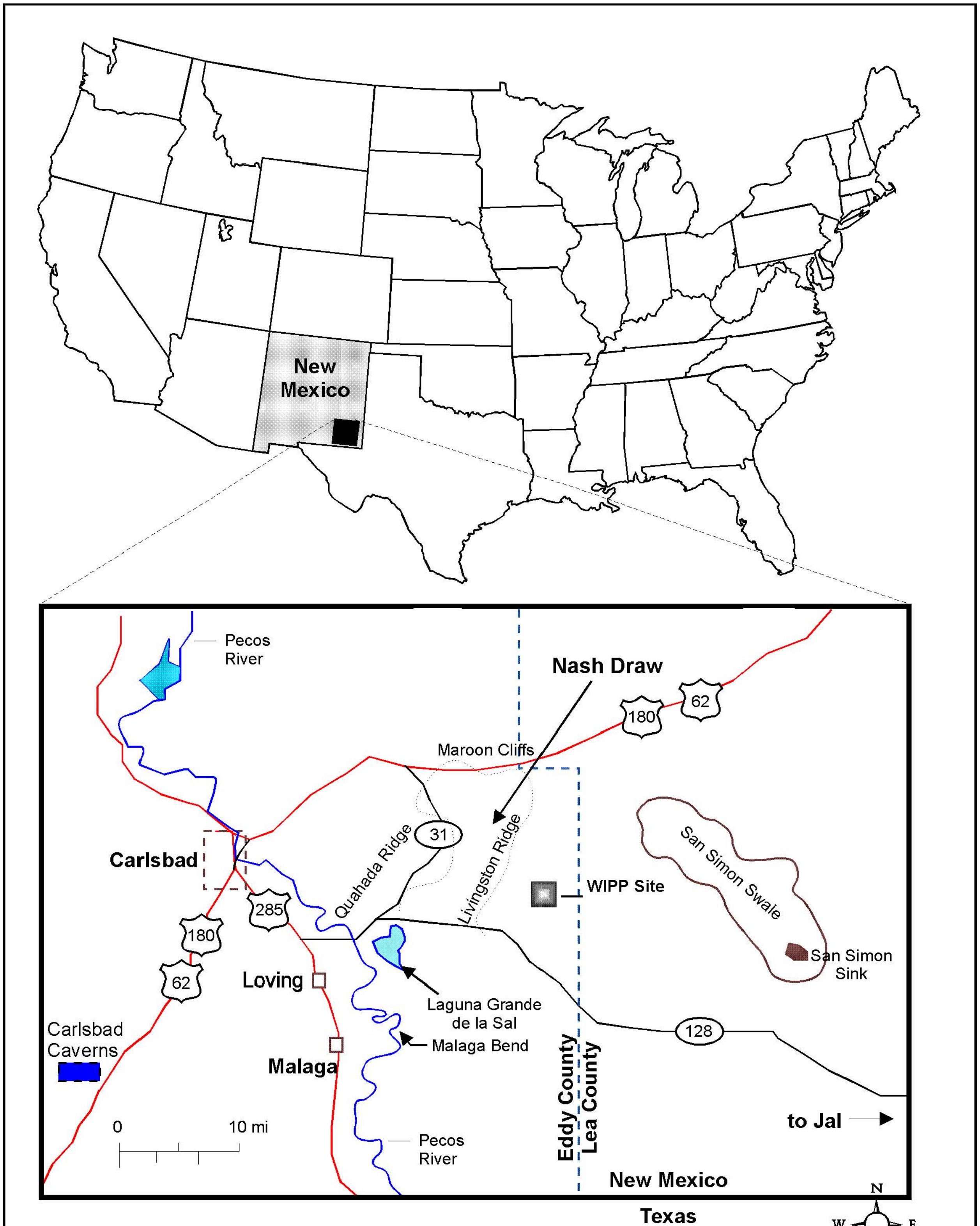
1. Public Law 102-579, Waste Isolation Pilot Plant Land Withdrawal Act, October 1992 [as amended by Public Law 104-201].

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

3. DOE/NTP-96-1204, The National Transuranic Waste Management Plan, U. S. Department of Energy, Carlsbad Field Office.

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. DOE/WIPP-Draft-3123, Remote-Handled Waste Acceptance Criteria for the Waste Isolation Pilot Plant.



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

Figure 1.2-1, WIPP Location in Southeastern New Mexico

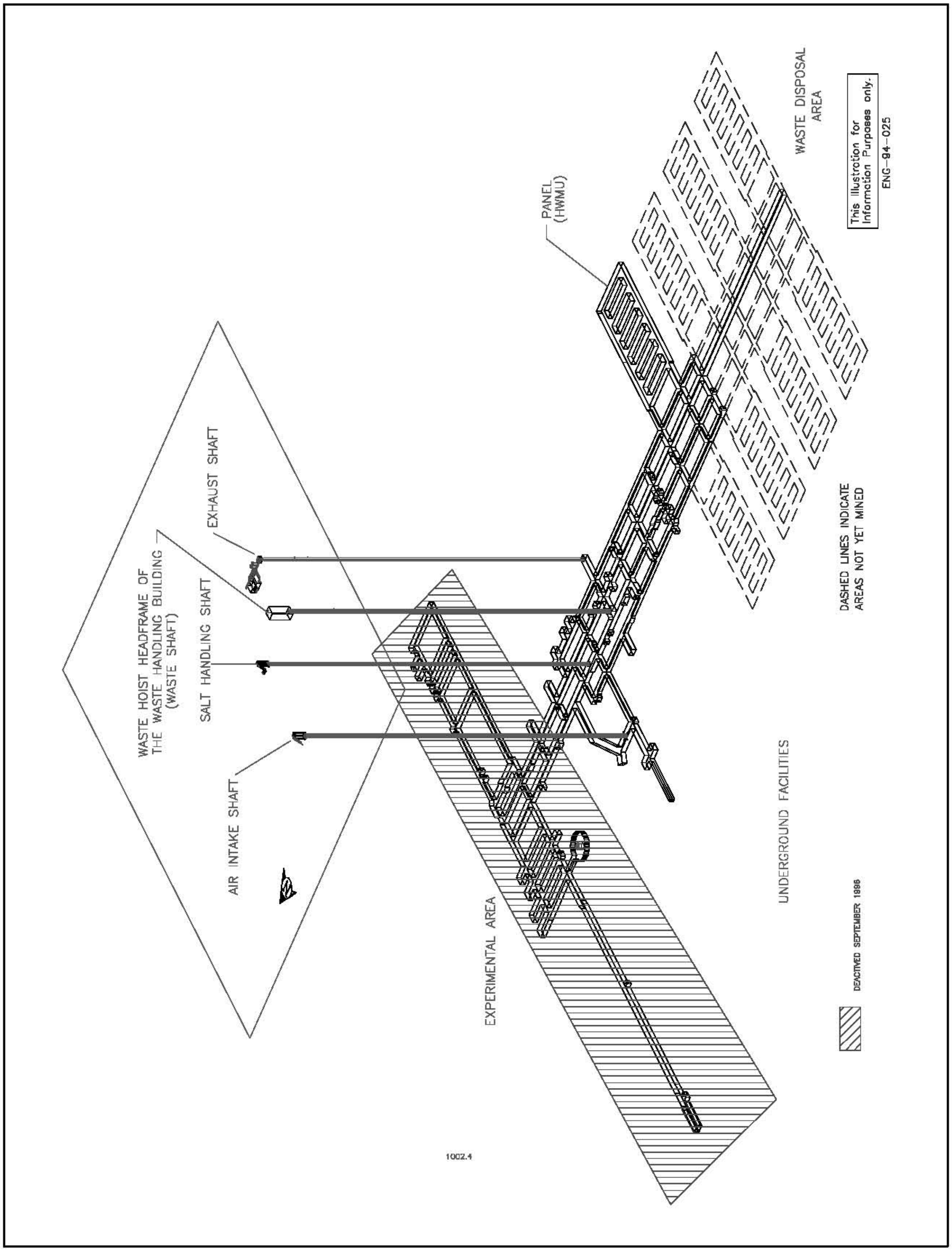


Figure 1.2-2, Spatial View of the WIPP Facility

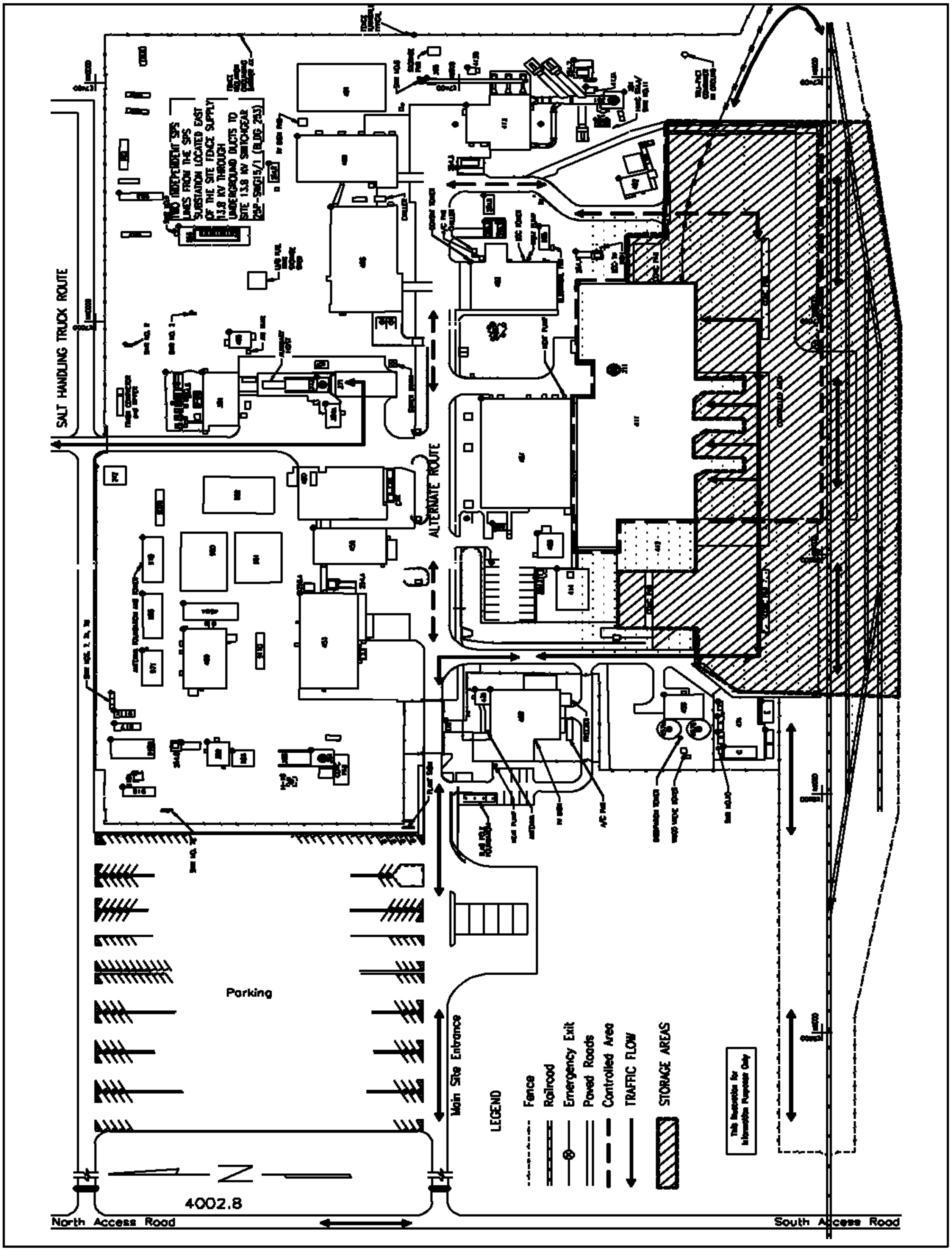


Figure 1.2-3a, WIPP Surface Structures

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

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

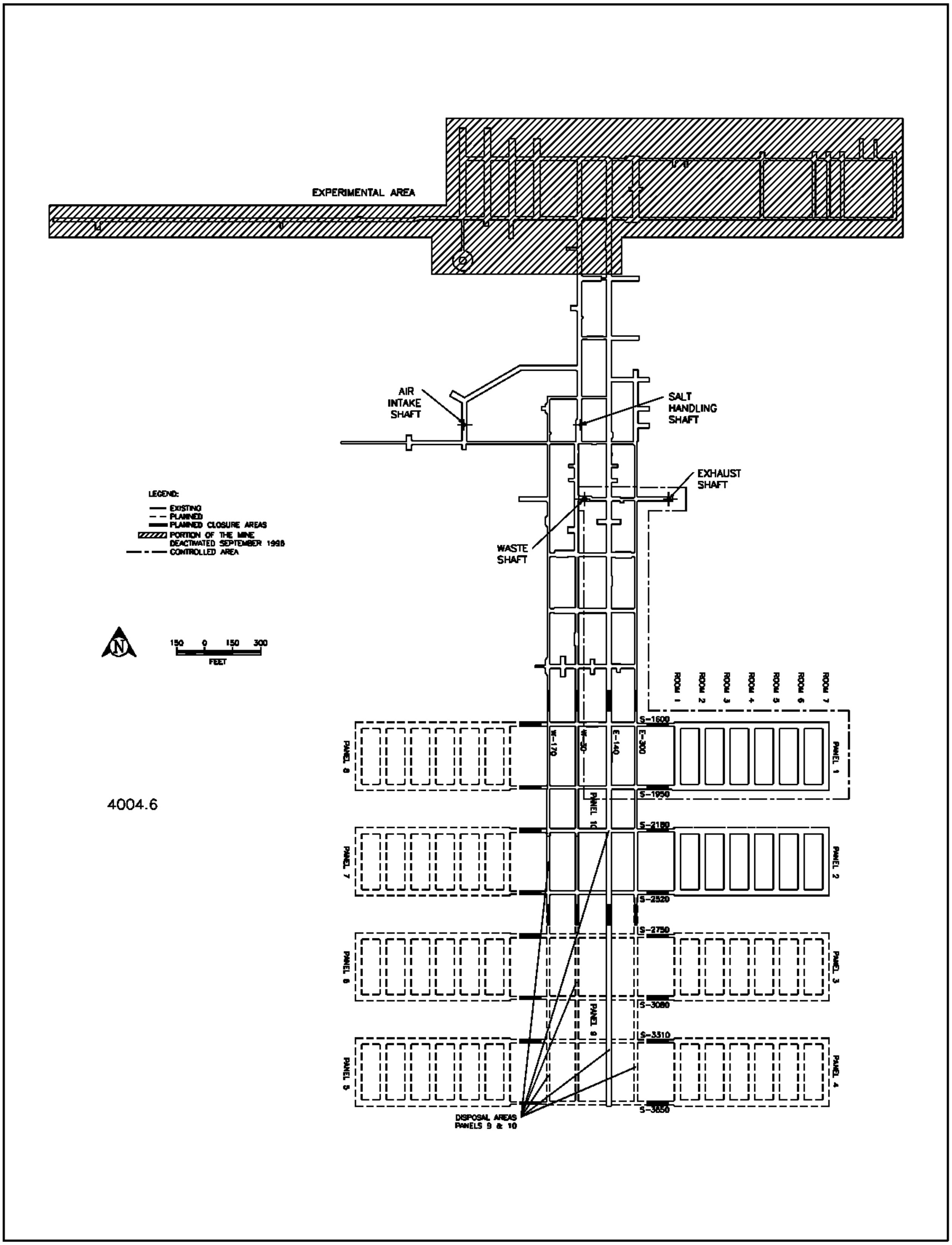

Figure 1.2-4, Underground Subsurface Areas

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1.3 Safety Analysis Overview and Conclusions

1.3.1 Safety Analysis Report Strategy and Approach

The WIPP RH PSAR is prepared to satisfy 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. The initial draft was written to ensure compliance with the requirements of DOE Orders 5480.21, Unreviewed Safety Questions,[2] 5480.22, Technical Safety Requirements,[3] 5480.23, Nuclear Safety Analysis Reports,[4] and 420.1, Facility Safety.[5] This draft of the RH PSAR is prepared to comply with the methodology and requirements of 10 CFR 830, Nuclear Safety Management [6](including Parts 830.203, Unreviewed Safety Question Process, 830.204, Documented Safety Analysis, 830.205, Technical Safety Requirements, and 830.206, Preliminary Documented Safety Analysis),[6] and its implementing standards DOE-STD-1027-92[7] and DOE-STD-3009-94[8]. A "Preliminary" SAR generally refers to a facility in the design, construction, or preoperational stage. This PSAR represents a statement and commitment by the DOE that the WIPP can be operated safely and at acceptable risk.

In accordance with the requirements of 10 CFR 830.204 [6], the SAR documents the safety analyses that develop and evaluate the adequacy of the safety bases. The safety bases are defined by 10 CFR 830.3, Definitions,[6] as: "The documented safety analysis and hazard controls that provide reasonable assurance that a DOE nuclear facility can be operated safely in a manner that protects workers, the public, and the environment."

This PSAR establishes and evaluates the adequacy of the WIPP RH 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 RH structures, systems, or components (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 preliminary Technical Safety Requirements (PTSRs), and (4) the management, conduct of operations, and institutional dimensions of safety assurance.

1.3.1.1 Facility Hazard Classification

The hazard classification 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.[7] 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 classification was defined as the maximum radiological contents of a single RH waste container as derived in Chapter 5. The WIPP Facility is classified as a Hazard Category 2 facility based on this single waste canister inventory in comparison to the threshold quantities provided in Table A-1 of DOE-STD-1027-92.[7]

1.3.1.2 Design and Operation Descriptions

The System Design Descriptions[9] (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.

The Woking Agreement for Consultation and Cooperation (WACC Agreement)[1] 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).[10]

The systematic evaluation of the human factors[11] associated with the design and operation of the WIPP to meet the requirements of DOE Order 5480.23[4] (10 CFR 830.204 [6]) and DOE-STD-3009-94[8] is discussed in Chapter 4. The evaluation determined that policies and procedures 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.

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 WIPP CCA.[10] Chapter 2 provides a detailed description of the site characteristics.

1.3.1.3 RH Waste Handling Hazard Analysis

The WIPP RH TRU waste handling processes were qualitatively evaluated in two Hazard and Operability Studies (HAZOPs),[12, 13] one for each type road cask (Summarized in Appendix C). This systematic approach to hazard analysis was conducted by leaders knowledgeable in the HAZOP methodology and consisted of personnel from various disciplines familiar with the design and operation of the RH TRU handling processes (HAZOP Team). The HAZOP Teams identified deviations from the intended design and operation of the RH waste handling systems that could: (1) result in process slowdown or shutdown, (2) result in worker injury or fatality, and (3) result in the release of radiological and non-radiological materials from waste containers.

Both HAZOP Teams 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), non-involved worker, and immediate worker. Based on this ranking approach HAZOP[12, 13] deviations whose combined hazard rank were identified to be of moderate or high risk (Table 5.1-10) were selected for <u>quantitative</u> analysis in Section 5.2 to: (1) verify and document the basis for the qualitative frequency and consequence assignments in the HAZOP,[12, 13] and (2) identify the need for safety (safety-class or safety-significant) SSCs and PTSRs.

The HAZOPs[12, 13] replace previous hazards analyses in existing documentation including the Final Environmental Impact Statement (FEIS),[14] and the Final Supplement Environmental Impact Statement (SEIS),[15] for the purposes of identifying initiating events for quantitative accident analysis in Section 5.2. These documents were reviewed to ensure that all hazards associated with RH TRU waste handling were identified in the HAZOPs.[12, 13]

The HAZOP Team concluded that:

- Safeguards currently exist at the WIPP to <u>prevent or reduce the frequency</u> of postulated accidents from occurring. Identified safeguards include facility and equipment design, procedures, training, preventative maintenance and inspection, and administrative controls including the RH Waste Acceptance Criteria (WAC)[16] (Table 5.1-10, and Appendix C).

- Mitigation exists to reduce the consequences of any postulated accident to acceptable levels. Identified mitigation includes confinement/ventilation systems and associated HEPA filtration systems (Table 5.1-10, and Appendix C).

Based on the results of the HAZOP,[12, 13] operational events are binned into three accident categories (fire, explosion, and breach of waste canister/drum). Since breach of waste canisters 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 for surface and underground areas of the facility. Natural initiating events including seismic and tornado are also evaluated.

Since the performance of the HAZOPs, periodic updates of the WIPP Fire Hazards Analysis Report (FHA)[24] have been performed to meet the requirements of DOE O 420.1.[5] The updated FHA confirms the previous evaluation that the frequency of a room or structural fire, as an accident in the WHB resulting in a direct release of radioactive material from the waste containers engulfed in the fire, is beyond extremely unlikely (<1E-06/yr). The updated FHA confirmed also that due to the limited combustible fire loading of the WHB waste processing rooms and the WHB design features, worse case fire accidents will not thermally challenge waste container integrity.

1.3.1.4 Defense in Depth

The WIPP defense-in-depth provides three layers of defense which include conservative design of the facility's SSCs, protection against anticipated operational occurrences and unlikely events, and passive controls that may be on line continuously or automatically/manually activated.

The objective of the first layer of WIPP defense-in-depth is **accident prevention**. The reduction of risk to both workers and the public from WIPP RH TRU waste handling and emplacement operations is primarily achieved by reducing the frequency of occurrence of postulated accidents. The conservative design of the facility's SSCs, with operations conducted by personnel trained and qualified 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,[12, 13] and in Table 5.1-10 for each deviation considered for quantitative accident analysis.

Additionally, accident prevention for process inherent events such as spontaneous ignition fire, is achieved administratively through the RH WAC[16] which restricts hazardous waste elements (such as the presence of pyrophorics) which may be initiating events for accidents. The following provide administrative controls (ACs) to prevent the risk from postulated accidents from being unacceptable: (1) RH WAC limits on the radionuclide and fissile content of each waste canister/drum; (2) RH WAC limits on hazardous waste such as non-radionuclide pyrophorics, explosives, and compressed gases, (3) waste canister/drum integrity provisions ensure the robustness reflected in the waste canister accident release analyses, and (4) criticality safety is a designed in-storage and handling configuration that ensures 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. PTSR ACs are derived in Chapter 6 and required in the WIPP PTSR Document (Attachment 1 to the PSAR) to ensure that these programs are maintained, and operations continue to be conducted with highly qualified and trained personnel using current approved procedures.

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 WHB and underground radiation monitoring systems, the HEPA filtration systems, and the WIPP emergency management program [25] provide this layer of defense-in-depth.

The third layer of defense-in-depth supplements the first two layers by providing protection against extremely unlikely operational, natural phenomena, 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 non-radiological effects of such accidents on the maximally exposed individual (MEI), non-involved worker, and immediate worker to verify that a conservative design basis has been established. These accidents include waste canister/drum fire and waste hoist failure.

1.3.1.5 Waste Acceptance Criteria

The waste accepted for placement in the WIPP facility must conform with the RH WAC[16] unless an exception to the RH WAC[16] 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 RH WAC Operations and Safety Requirements. A PTSR AC for Waste Characteristics require that the safety analysis criteria be incorporated into the RH WAC.[16]

The RH WAC[16] establishes minimum criteria that the waste must meet, and limits that cannot be exceeded in order to ensure that TRU waste is managed and disposed of in a manner that protects worker and public health and safety and the environment. The following waste is unacceptable for management at the WIPP facility:

- Ignitable, reactive, and corrosive waste
- Liquid wastes (all waste must meet the RH WAC [16] criteria regarding residual liquid content)
- Compressed gases
- Incompatible waste (waste must be compatible with backfill, seal and panel closure materials, canister, road cask, facility cask, and as well as with other waste)
- Headspace-gas VOC concentrations resulting in average annual emissions not protective of human health and the environment
- Wastes with EPA codes not listed on Hazardous Waste Facility Permit, Table II.C [26].
- Waste with equal to or more than 50 ppm (50 mg/L) polychlorinated biphenyls (PCB)

The WIPP facility will not accept waste that exhibits the characteristics of ignitability, reactivity, or corrosiveness.

Estimates of the radiological waste canister inventory for safety analysis calculations were obtained by using the radionuclide inventory by final waste form, stored waste volume, and waste site included on the June, 1996 query of the WIPP Transuranic Waste Baseline Inventory Report (BIR)[27] database.

This PSAR has evaluated a reasonable range of Container Inventories (CIs) for "untreated" (not solidified, vitrified, or overpacked) RH 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, the potential dose consequences due to inhalation by the non-involved worker, the immediate workers, and the MEI from operational accidents with frequencies greater than 1E-06/yr are within the risk evaluation guidelines in Section 5.2.2.

The adequacy of the WIPP facility design and operational administrative controls is evaluated, based on the accident results in Section 5.2.

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[8] and its Appendix A 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."

As described in Section 5.1.2, the maximum plutonium-239 equivalent Curies (PE-Ci) radionuclide inventory for a 72B canister loaded with non-containerized waste (direct loaded) is 80 PE-Ci. The maximum radionuclide inventory for a 72B canister loaded with waste contained in three 55-gallon drums is 240 PE-Ci. The maximum radionuclide inventory for a 10-160B road cask containing ten 55-gallon drums of waste is 20 PE-Ci.

The adequacy of these assumptions and the WIPP RH TRU facility design basis are evaluated in detail based on the accident results in Section 5.2.3. Receipt of waste for disposal at WIPP that does not meet the applicable Operations and Safety Requirements of the RH WAC[16] will first require the performance of an Unreviewed Safety Question Determination (USQD) in accordance with the requirements of 10 CFR 830.203, Unreviewed Safety Question Process.[6]

1.3.1.6 Nuclear Criticality

The intent of a criticality safety program is to prevent the accumulation of fissile and fissionable material and neutron moderating or reflecting materials in quantities and configurations that could result in an accidental nuclear criticality.

To ensure adequate margins of criticality safety for adherence to DOE O 420.1,[5] the WIPP facility was designed so that during each operation involving fissile material K_{eff} does not exceed a value of 0.947 (at the 95 percent probability level) for the most reactive set of conditions considered credibly possible. The calculation of K_{eff} includes the effect of neutron interaction and reflection between fissile elements and dimensional variations resulting from fabrication tolerances and changes due to corrosion and mechanical distortion. As discussed below, these calculations indicate the combination of conditions enabling the K_{eff} limit of 0.947 to be exceeded for the RH waste forms handled at the WIPP facility is incredible.

The WIPP nuclear criticality program elements consist of mass limits control, TRU waste disposal configuration control, and analytical verification of subcriticality.

Mass Limits Control

The WIPP RH WAC[16] limits the fissile or fissionable radionuclide content of RH TRU waste, including allowance for measurement errors, to 325 Fissile-Gram Equivalent (FGE) for a RH waste canister.

TRU Waste Disposal Configuration Control

In addition to the mass limits control, geometry controls are required for the emplacement and/or in-transit handling disposal configurations. Canisters will be stored in horizontal positions in the walls of the Underground disposal rooms with an analyzed minimum center-to-center spacing of 30 in (76 cm).[17]

RH TRU Nuclear Criticality Safety Analysis

In compliance with DOE O 420.1,[5] a criticality analysis[17] was performed to ensure that no credible criticality accident could occur at the WIPP. The analysis was based on the mass limit control and geometry control, with additional conservative assumptions in terms of; isotopic content, density and configuration modeling, moderation, and reflection. Further, for the RH waste analysis, it was assumed that the waste package storage array is infinite in both horizontal directions.

The results of the WIPP RH TRU criticality analysis[17] indicate that, for each of the conditions analyzed, the calculated effective multiplication factor, K_{eff}, is less than 0.95 including uncertainties at 95 percent probability at 95 percent confidence level. Accordingly, no credible criticality hazard exists at the WIPP for RH TRU operations.

DOE Order 420.1[5] requires additional analysis of nuclear criticality safety. The WIPP RH TRU criticality analysis [17] was examined for compliance with the order and all the applicable requirements for the order in performance of criticality analysis were complied with within the analysis.

The WIPP nuclear criticality safety program elements were reviewed to ensure compliance with the six mandatory American Nuclear Society ANSI/ANS nuclear criticality safety standards as the Order requires. The six mandatory standards are: ANSI/ANS-8.1,[18] 8.3,[19] 8.5,[20] 8.7,[21] 8.15,[22] and 8.19.[23]

The WIPP nuclear criticality safety program elements are found to be in compliance with the requirements of ANSI/ANS-8.1, Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors,[18] and ANSI/ANS-8.15, Nuclear Criticality Control of Special Actinide Elements,[22] in regard to: mass control, geometry control, and performance of criticality analyses.

The criticality-related administrative control provisions were determined to be in compliance with ANSI/ANS-8.19, Administrative Practices for Nuclear Criticality Safety.[23]

Since it has been established by analyses[17] that a criticality accident is beyond extremely unlikely (frequency $\leq$ 1 E-06/yr) at the WIPP, ANSI/ANS-8.3,[19] a Criticality Accident Alarm System, is not applicable as called for in the Order.

The two facility-specific standards, ANSI/ANS-8.5, Use of Borosilicate-Glass Raschig Rings as a Neutron Absorber in Solutions of Fissile Material,[20] and ANSI/ANS-8.7, Guide for Nuclear Criticality Safety in the Storage of Fissile Materials,[21] are not applicable to the WIPP.

The existing WIPP nuclear criticality safety program elements are in compliance with the DOE Order 420.1[5] mandatory criticality safety standards.

1.3.1.7 Atmospheric Dispersion

The meteorological conditions used to evaluate both radiological and non-radiological doses are based on Nuclear Regulatory Guide (NRG) 1.145,[28] "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants." NRG 1.145 [28] provides an NRC acceptable methodology to determine site-specific atmospheric dispersion coefficient (χ/Q'). χ/Q', is a ratio of the air concentration, χ, to the release rate, Q', and is used to determine the dose consequences for a receptor based on the quantity released (i.e., the source term), atmospheric conditions, and the distance to the receptor of interest. This methodology was used to develop the atmospheric dispersion coefficients to assess accidental releases from the WIPP Underground exhaust shaft and the WHB exhaust vent. Section 5.2.1.1 provides a more detailed explanation of atmospheric dispersion at WIPP. The model used is a straight line Gaussian plume which is appropriate to WIPP due to the terrain around the property protection area.

1.3.1.8 Significant Hazards

The accident analyses utilize currently available Rules, DOE Orders, standards and guidance as documented in DOE-STD-3009-94[8] and DOE-STD-1027-92[7], for determination of safety of the public, worker, and the environment. This PSAR 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 RH TRU waste operations conducted. The accidents selected for quantitative analysis are considered "Derivative Design Basis Accidents," (DBAs) as defined in DOE Standard 3009-94[8]. The 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 following accidents were selected for analysis (the accidents identified with RH are for 72B waste operations, while those with the NC identifier are for 10-160B waste operations):

1. Operational Events

   Fires

   RH1 Fire in the Underground

   RH2 Fire in the WHB

   RH5 Fire followed by explosion in the Underground

   NC1 Fire in the Hot Cell

   NC2 Fire in the Underground

   NC5 Explosion followed by fire in the Hot Cell

   NC6 Fire followed by explosion in the Underground

   Waste Container Breaches

   RH3 Loss of containment in the WHB

   RH4-A Loss of containment in the Underground (waste hoist)

   RH4-B Loss of containment in the Underground (waste transport & emplacement)

   NC3 Loss of confinement in the WHB. This scenario is divided into sub-parts NC3-A, NC3-B, NC3-C, NC3-D, NC3-E, NC3-F, NC3-G, NC3-H, and NC3-I

   NC4 Loss of confinement in the Transfer Cell or Underground

2. Natural Events

RH6 Seismic event

RH7 Tornado event

NC7 Seismic Event

NC8 Tornado event

3. External Events

RH8 Aircraft Crash (applicable to both 72B and 10-160B operations)

It should be noted that accidents NC3-I and NC5 occurred in the Hot Cell and were initiated by the arc of the robotic electric welder that was to be used to weld the lid to a facility canister. The facility canister was redesigned, after the 10-160B HAZOP, so that the lid is mechanically attached to the facility canister and the welder was removed from service. As a result, accidents NC3-I and NC5 were not evaluated.

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

For the purposes of establishing safety SSCs, the consequences of these accidents are analyzed to a non-involved worker conservatively assumed to be 328 ft (100 m) from each release point, to the MEI located at the WIPP Exclusive Use Area boundary, and to the immediate worker located in the immediate vicinity of the accident. 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. In the RH waste handling process, there are no immediate workers present in the Hot Cell or Transfer Cell. There is no immediate worker present in the Cask Unloading Room (CUR) when a 10-160B cask is processed.

1.3.2 Off-site and On-site Risk Evaluation Guidelines

DOE Standard 3009-94[8] 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, identified DBAs whose frequencies 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 5.2.4.

Guidelines do not exist for the frequency range of beyond extremely unlikely (frequency ≤ 1E-06/yr). The consequences of accidents in that range are conservatively evaluated against the guidelines for the extremely unlikely range for the sole purpose of evaluating the risk associated with facility operations.

1.3.2.1 Radiological Evaluation Guidelines

Off-site radiological dose criteria for accident analyses have been well established by national standards through the licensing process of nuclear facilities regulated by the NRC. These criteria are based on the probabilities of occurrence of the accidents or events hypothesized for the accident analysis. For nuclear power plants, the operational accidents or events are classified as Plant Conditions (PC) in accordance with the estimated frequency of occurrence. ANSI/ANS-51.1[29] provides frequency based radiological dose values, recognized by the NRC, which are used by nuclear power plants, those values have adopted by the WIPP to compare accidental releases from postulated events to dose limits based on estimated frequency of occurrence. Table 1.3-1 summarizes the risk evaluation guidelines for the assessment of off-site radiological exposures.

The same approach is used for the on-site risk evaluation guidelines as for the off-site (public) dose (Table 1.3-2). The on-site risk evaluation guidelines are greater than those for the public by assuming 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 health effects noticeable to exposed individuals at frequencies greater than 1E-4 event per year and would not result in any acute life-threatening effects.

For accidents with an estimated frequency between 1E-1 and 1E-2 event per year (anticipated) the limit is 5 rem (50 mSv) based on the allowable yearly worker exposure limits cited in 10 CFR 835.[30] For the estimated frequency range of 1 E-2 to 1 E-4 event per year (unlikely), the threshold is 25 rem (250 mSv) for the same reason the NRC provided in 10 CFR 100[31] for using it for design basis reactor accident calculations (i.e., value at which no significant health effects result).

Accidents with an estimated frequency range of 1E-4 to 1E-6 event per year (extremely unlikely) have a limit of 100 rem (1 Sv). The DOE Emergency Management Guide for Hazards Assessment[32] uses 100 rem (1 Sv) whole body exposure as a threshold for early severe effects. It also acknowledges that early severe effects would not actually be experienced for a 50-year dose of 100 rem (1 Sv) due to alpha emitters.

1.3.2.2 Radiological Evaluations

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, non-involved 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: the (1) MEI located at the WIPP Exclusive Use Area boundary and off-site public located at the site 16 section boundary, (2) non-involved worker located at 328 ft (100 m) from each release point, and (3) immediate worker located within the immediate vicinity of the accident.

Atmospheric transport is the only significant release and exposure pathway during normal operations and accident conditions during the disposal phase. Based on the site characteristics information in Chapter 2, surface water and groundwater transport from normal or accidental releases of radioactive material is not considered likely. Human exposure pathways from the airborne radioactive material include inhalation, air immersion, ingestion, and ground-shine. Radiological dose consequences are calculated assuming the inhalation pathway in CEDE.

External (ground-shine and air immersion) and ingestion dose calculations are not performed due to their minimal contribution to the Total Effective Dose Equivalent (TEDE). Section A.3 in Appendix A of DOE-STD-3009-94[8] states that the airborne pathway is of primary interest in the non-reactor nuclear facilities, therefore CEDE will be reported as the dose consequences for each accident evaluated. The calculated dose in CEDE is then compared to the non-involved worker and MEI radiological risk evaluation guidelines discussed in Section 5.2.2.

In evaluating hypothetical accidents, the safety analysis assumptions 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[8] and bound the full range of possible scenarios, and 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 PTSRs assigned will provide for the protection of the public, the worker, and the environment.

For accidents with an estimated frequency between 1E-2 to 1E-4 event per year (unlikely), the MEI limit is 6.5 rem (65 mSv) and the noninvolved worker limit is 25 rem (250 mSv). Accidents with an estimated frequency range of 1 E-4 to 1 E-6 event per year (extremely unlikely) have a MEI limit of 25 rem (250 mSv), while the non-involved worker limit is 100 rem (1 Sv). Since no current guidelines exist for immediate workers, the non-involved worker limit of 100 rem (1 Sv) is used for the immediate worker limits for all frequencies.

The quantitative frequency analysis (in Section 5.2.3) for each accident produced accidents in the three ranges, Unlikely, Extremely Unlikely, and Beyond Extremely Unlikely.

Additional quantitative frequency analyses in the form of event/fault tree analyses (Appendix D) 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 non-involved worker and MEI). Specific accidents evaluated in this manner were: RH3, RH4A, RH4B, RH6, RH7, NC1, NC3 (A-G), and NC5. With the exceptions of RH4B, RH6, NC1, and NC3(A - F), the event tree/fault tree analyses indicated that the no-mitigation frequency of the identified accidents occurring are beyond extremely unlikely (frequency ≤ 1E-06/yr).

Based on the RH accident source term and release mechanism analyses presented in Section 5.2.3 for accident scenarios with a frequency greater than 1E-06/yr (RH2, RH4-B, RH6, NC1, NC3 (A - F), NC4, NC7, and NC8), the calculated worst-case no-mitigation accident consequences to the non-involved worker, the MEI, and immediate worker were found to be below the accident risk evaluation guidelines for the unlikely range 25 rem (250 mSv) for the non-involved worker and 6.5 rem (65 mSv) for the MEI. The highest consequences to the non-involved worker are obtained from NC-1, with an estimated 8.2 rem (82 mSv) approximately 8 percent of 100 rem (1 Sv) guideline and 0.65 rem (.65 mSv) to the MEI approximately 3 percent of 25 rem (250 mSv) guideline. The highest consequences to the immediate worker are obtained from RH4-B, with an estimated 5.4 rem (54 mSv), approximately 5 percent of 100 rem guideline.

The MEI no-mitigation consequences for all accidents analyzed, regardless of frequency, were found to be below 25 rem (250 mSv) risk evaluation guideline. The worst-case for the 10-160B analysis calculated dose to an immediate worker is from NC3-G and NC3-H with an estimated 4.16 rem (41.6 mSv), which is below the on-site risk evaluation guidelines for the unlikely range (6.5 rem).

The consequences to the immediate worker from RH4-B are well within the non-involved worker evaluation guidelines. Therefore, no specific additional worker protection, engineering, or ACs other than those already qualitatively identified as providing defense-in-depth for the immediate worker, are needed.

For scenarios with a frequency less than 1E-06/yr (RH1, RH3, RH4-A, RH5, RH7, NC3-G, and NC3-H), the highest consequences are in RH3, which occurs in the Transfer Cell (no immediate worker present), with an estimated 65.8 rem (658 mSv) to the non-involved worker (approximately 66 percent of 100 rem (1 Sv) guideline), and 5.2 rem (52 mSv) to the MEI (approximately 21 percent of 25 rem (250 mSv) guideline. The non-involved worker consequences (65.8 rem) is below the guideline (100 rem) for selection of Safety Significant SSCs. The Transfer Cell safety features are passive therefore TSR controls are not necessary. The highest dose consequences to the immediate worker occurs during scenario RH4-A with a 116 rem (1.16 Sv) dose, 116 percent of 100 rem guideline.

For protection of the immediate worker, the waste hoist brake system is designated Safety Significant and assigned in Attachment 1, Preliminary Technical Safety Requirements. The risk associated with the potential exposure to the immediate worker from RH4-A is deemed acceptable for the following reasons:

- 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.

1.3.2.3 Non-radiological Evaluation Guidelines

DOE orders do not contain a unique set of approved non-radiological risk evaluation guidelines. The WIPP non-radiological risk guidelines are based on Emergency Response Planning Guidelines (ERPG) published by the American Industrial Hygiene Association (AIHA). ERPGs are estimates of concentration ranges for specific chemicals above which acute (< 1 hour) exposure would be expected to lead to adverse health effects of increasing severity. The EPRG-1 values represents a concentration that would have little or no health effects, while EPRG -3 values have the most severe health effects.

The definitions of ERPGs are:

ERPG-1 The maximum concentration in air below which it is believed nearly all individuals could be exposed for up to one hour without experiencing other than mild transient adverse health effects or perceiving a clearly defined objectionable odor.

ERPG-2 The maximum concentration in air below which it is believed nearly all individuals could be exposed for up to one hour without experiencing or developing irreversible or other serious health effects or symptoms that could impair their abilities to take protective action.

ERPG-3 The maximum concentration in air below which it is believed nearly all individuals could be exposed for up to one hour without experiencing or developing life-threatening health effects.

ERPGs have been developed for approximately 100 chemicals and do not exist for some of the chemicals found in TRU mixed waste. Chemicals without established ERPG values will use Temporary Emergency Exposure Limits (TEELs) developed by the DOE Emergency Management Advisory Committee's Subcommittee on Consequence Assessment and Protective Action (SCAPA), Revision 18, Table 4. SCAPA developed TEELs to allow for the preliminary identification of hazardous or potentially hazardous situations for emergency planning even when ERPGs were not available. The TEEL is an interim parameter meant to approximate an ERPG so that emergency planning and preparedness activities can be conducted. Whenever an ERPG is developed for a new chemical, the ERPG replaces the TEEL. The definitions of TEELs are:

TEEL-0 The threshold concentration below which most people will experience no appreciable risk of health effects;

TEEL-1 The maximum concentration in air below which it is believed nearly all individuals could be exposed without experiencing other than mild transient adverse health effects or perceiving a clearly defined objectionable odor.

TEEL-2 The maximum concentration in air below which it is believed nearly all individuals could be exposed without experiencing or developing irreversible or other serious health effects or symptoms that could impair their abilities to take protective action;

TEEL-3 The maximum concentration in air below which it is believed nearly all individuals could be exposed without experiencing or developing life-threatening health effects.

The following TEEL values will be used for those chemicals in TRU waste that do not have an ERPG value.

ERPG-1	TEEL-1
ERPG-2	TEEL-2
ERPG-3	TEEL-3

1.3.2.4 Non-radiological Evaluations

Hazardous waste, as defined in 40 CFR 261, Subparts C and D,[33] often occurs as co-contaminants with TRU waste from defense-related operations, resulting in "TRU mixed waste." The BIR[27] estimates the quantities of RCRA regulated TRU waste to be shipped from each generator site. The most common hazardous constituents in the TRU mixed waste consist of the following: (1) metals such as beryllium, cadmium, lead, mercury (2) solidified sludges; (3) cemented laboratory liquids, and waste from decontamination and decommissioning activities; (4) asbestos; (5) polychlorinated biphenyls (PCBs); (6) halogenated organic solvents such as methylene chloride; Tetrachloroethylene; trichloroethylene; carbon tetrachloride; 1,1,1-trichloroethane, and 1,1,2-trichloro-1,2,2-trifluoroethane; (7) non-halogenated organic solvents such as xylene, methanol, and butyl alcohol. The solvents are referred to as volatile organic compounds (VOCs).

The assumptions used in the analysis for determining the amount of hazardous non-radiological chemicals released to the environment and the extent of exposure to the MEI, non-involved worker, and immediate worker are provided in Section 5.2. Chemical exposures in milligrams per cubic meter (mg/m^3) were calculated for the MEI, off-site public, the non-involved worker, and the immediate worker. Atmospheric dispersion of hazardous chemicals was performed using NRG 1.145,[28] which is described in Section 1.3.16.

For accidents with an estimated frequency between 1E-1 and 1E-2 event per year (anticipated), the MEI limit and the non-involved worker limit is ERPG-1. For the estimated frequency range of 1E-2 to 1E-4 event per year (unlikely), the MEI limit is ERPG-1 and the noninvolved worker limit is ERPG-2. Accidents with an estimated frequency range of 1E-4 to 1E-6 event per year (extremely unlikely) have a MEI limit of ERPG-2 while the non-involved worker limit is ERPG-3. Since no current guidelines exist for immediate workers, EPRG-3 is used for the immediate worker limits for all frequencies.

Based on the RH accident analyses presented in Section 5.2.3, for accident scenarios with a frequency greater than 1E-06/yr, only accident scenario NC1, drum fire in the Hot Cell, required using the guidelines for all the substances listed on Table 5.2-2. Loss of confinement (breach and puncture) accidents and natural phenomena (seismic and tornado) accidents cause the release of VOCs; methylene chloride, carbon tetrachloride, chloroform, and 1,1,2,2-tetrachloroethane. There will not be a buildup of hydrogen gas due to the vent filters installed on the waste drums and on the 72B waste canister. The facility canister is not vented through a filter, but is constructed so as not to be air tight which will prevent a hydrogen build up.

NC1 accident consequences to the non-involved worker and the MEI were found to be less than 1% of evaluation guidelines for the extremely unlikely range (ERPG-2). There are no immediate worker consequences for this accident because all work in the Hot Cell is performed remotely and the Hot Cell has its own ventilation system..

The MEI non-radiological consequences for all accidents analyzed in Section 5.2 and shown on Table 5.2-4a, were 1 percent or less of their respective guidelines. The non-involved worker worst-case consequences for the four VOCs was for carbon tetrachloride which occurred during NC3-D. The non-involved worker carbon tetrachloride consequences during NC3-D were 12 percent of the ERPG-1 guidelines.

The VOCs contained in the RH waste and any hydrogen generated in the RH waste will escape from the waste containers (canisters and 55-gallon drums) and will be rapidly diffused and diluted by the high underground ventilation flow, approximately 35,000 cubic feet per minute. Therefore, the VOCs and hydrogen will have minimal, if any, impact on the repository.

1.3.2.5 Preventive and Mitigative Features

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 non-radiological consequences to the MEI and non-involved 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.

The accident analyses indicate that safety (safety-class or safety-significant) SSCs are not required for the WIPP to mitigate any MEI or non-involved worker accident radiological and non-radiological consequence resulting from RH waste operations 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,[5] 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 orders or guidance documents for evaluation of secondary confinement. As previously stated, the MEI and non-involved worker unmitigated consequences were found to be below the selected risk evaluation guidelines, including accidents whose frequency is $\leq$1E-06/yr, and as such, secondary confinement is not required. However, existing Design Class II and IIIA secondary confinement SSCs, while not required 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 PTSR 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 **operational** accidents analyzed in the WHB or the underground will disable the respective ventilation or HEPA filtration systems. No releases are postulated requiring ventilation or HEPA filtration for the Design Basis Earthquake (DBE) and Design Basis Tornado (DBT) scenarios. If waste container breach occurs in the WHB during a credible operational accident, the release to the outside environment is mitigated by the permanently installed continuously on-line two-stage HEPA filter. For accident scenarios in the underground, shift to HEPA filtration of the underground ventilation exhaust 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 releases 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 6.25-ton grapple hoist and the RH Bay 140/25-ton crane. 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 non-involved 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 MSHA regulation.

As concluded from WIPP RH PSAR Section 5.2, none of the analyzed scenarios resulted in non-involved 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 non-involved worker consequence resulting from RH waste operations.

With regard to the waste hoist failure scenario (RH4A), the consequences involving waste hoist failure while transporting a loaded facility cask was evaluated in Chapter 5. The waste hoist will not be used to simultaneously transport personnel **and** a loaded facility cask 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[34] and the New Mexico Safety Codes for all Mines.[35] For protection of the immediate worker, the waste hoist brake system is designated Safety Significant and specific ACs are derived in Chapter 6 and in Attachment 1, Technical Safety Requirements.

Table 6.1 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 ACs (TSRs).

Specific WIPP SSCs are classified as defense-in-depth SSCs, based on the above functional classification results and accident impacts. Rather than the WIPP PSAR specify functional requirements and performance criteria for those defense-in-depth SSCs, the applicable SDDs[9] describe their intended safety functions, and specify the requirements for design, operation, maintenance, testing, and calibration.

As discussed in detail in Chapter 6, based on application of the criteria in 10 CFR 830.205 [6] for the selection of safety and operational limits, and the fact that Safety Class and Safety Significant SSCs (the waste hoist is the only Safety Significant SSC) are not selected for WIPP RH waste operations, PTSR Safety Limits (SLs), Limiting Conditions for Operation (LCOs), and Surveillance Requirements are not required. PTSR ACs assigned for features that play a role in supporting the WIPP defense-in-depth approach are derived in SAR Chapter 6. 10 CFR 830.205 [6] and its implementation guide allow coverage of Safety Significant SSCs through AC. Table 6-1 provides a summary of defense-in-depth safety features and applicable PTSR controls.

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 PSAR. SSCs are required in the PTSR 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 concluded from the hazards and accident analyses in this PSAR that the design basis of the WIPP RH TRU waste handling system is adequate in response to postulated range of RH TRU normal operations and accident conditions for the facility.

1.3.2.6 Technical Safety Requirements

PTSRs are developed based on the requirements provided in 10 CFR 830.205 [6], Technical Safety Requirements. 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 PTSR ACs are established as follows:

- To maintain the design, quality, testability, inspect ability, maintainability, and accessibility of the facility, PTSR 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 accident initiating event.

- 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, PTSR ACs are required relating to: (1) waste characteristics (WAC), (2) waste canister integrity, (3) criticality safety, (4) fire protection, and (5) waste handling PE-Ci limits. The PTSR AC for waste characteristics limits the radionuclide content of each waste canister, restricts the fissile content of the canister, and restricts the presence of waste characteristics unacceptable for management at the WIPP facility. Canister 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. Waste handling PE-Ci controls limit the radionuclide content of a road cask that can be handled during normal operations.

Supporting the second and third layers of defense-in-depth, WIPP PTSR ACs are identified which establish programs for radiation protection and emergency management. Basic elements and requirements defined for TSR AC programs are enforced by the associated implementing WIPP procedures.

1.3.3 Safety Analysis Conclusions

1.3.3.1 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 to the off-site evaluation guidelines and the immediate worker and non-involved worker to the on-site risk evaluation guidelines, are identified.

The predicted RH waste (radioactive/chemical) to be received in a waste container at the WIPP was conservatively estimated based on data, as shown in the BIR[27], from the generating sites, process knowledge, and limiting criteria provided in the RH WAC.[16] These estimates provided bounding container inventories used in the determination of potential consequences from postulated accidents.

Hazards associated with the facility RH processes were evaluated through two systematic hazard analysis processes, a 72B HAZOP and a 10-160B HAZOP. The analyses encompassed waste receipt, handling and disposal of RH TRU waste in the WIPP. Each hazards analysis involved a multi-step process which included: 1) identification of the potential hazards associated with the RH TRU waste handling processes, 2) characterization of the waste expected at the WIPP, and 3) a hazard evaluation in the form of a HAZOP.[12, 13] These multi-step processes provided comprehensive examinations of the potential hazards which may require quantitative evaluation in the accident analysis.

The major hazard associated with the RH TRU waste handling process is associated with the radiological and non-radiological hazardous materials within the waste container. Hazards associated with mining operations are considered standard industrial hazards governed by OSHA and MSHA regulations and are considered only when they may be an initiating event leading to the accidental release of radiological or non-radiological 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 canisters. Therefore, for the purposes of establishing an inventory of radiological and non-radiological material, only that material contained in the waste containers was considered.

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 5.1-10.

Review of the WIPP Land Management Plan[36] 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. The location of the MEI is at the "closest point of public access," or the DOE "exclusive use area" boundary which is consistent with guidance for the implementation of 40 CFR 191,[37] Subpart A. Calculations are performed in Appendix E for a member of the public at the site boundary for reference purposes.

Although prevailing winds are from the southeast 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 m) and the closest distance to the exclusive use area boundary from the WHB lies southeast at approximately 1150 ft (350 m) (Figure 5.2-2).

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

A summary of the non-involved worker and MEI radiological and toxicological consequences of analyzed accidents and comparison to risk evaluation guidelines is presented in Tables 5.2-3a, 5.2-3b, 5.2-4a, and 5.2-4b. Off-site (MEI) risk evaluation radiological guidelines are based on ANSI/ANS-51.1,[29] also used by the NRC, which compares dose consequences, due to accidental releases from postulated events, to the estimated frequency of occurrence. Non-involved worker radiological dose consequences are compared to on-site risk evaluation guidelines developed from available supporting DOE and ANSI guidance. The guidelines for chemical exposure are those provided in DOE O 151.1[32] and its guidance documents.

However, on-site risk evaluation guidelines are greater than those for the public as DOE-CBFO 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 regard 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 HAZOPs,[12, 13] of operational initiating events is based on the evaluation of process events (leaks), 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 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.

As required by DOE-STD-3009-94[8], a graded approach is used to achieve the objectives of analysis of accidents. The level of analytical effort is primarily a function of magnitude of the hazard, but also takes into account system complexity, and the degree to which detailed modeling can be meaningfully supported by system definition. For non-reactor nuclear facilities, such as WIPP, the Standard does not present an expectation of or requirement for probabilistic/quantitative risk assessment.

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 non-involved 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, 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 HAZOPs[12, 13]) in keeping worker dose from accidents ALARA. 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.

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

As required by the 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.[37] Paragraph 191.03(b) which 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 mrem (0.25 mSv) to the whole body and 75 mrem (0.75 mSv) to any critical organ. Also, paragraph 61.92 of 40 CFR 61 Subpart H[38] 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 (0.10 mSv).

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[30] external contamination limits. To ensure compliance, the containers are surveyed prior to release from the generator sites and again as the road casks are opened at the WIPP. Since radioactive material remains in the waste containers unless an accident occurs, emissions to the ambient air during normal WIPP waste disposal operation will be below measurable levels and for all practical purposes will not occur. A WIPP analysis[39] demonstrates, through dispersion modeling, that off-site radiological emission consequence to the public and environment resulting from normal waste disposal operations (without taking credit for any mitigation systems; i.e., HEPA filtration) will be minimal. WIPP management anticipates that 40 CFR 191, Subpart A[37] compliance sampling will confirm the dispersion modeling. WIPP hazard analysis demonstrates that EPA emission standards will not be exceeded unless waste containers are breached in a waste handling accident or in another off-normal event and facility mitigation systems fail. Also, the public is expected to receive a negligible dose during normal operations. As a result of the above information, it may be concluded that the WIPP will be operated in compliance with the release standards of 40 CFR 191 Subpart A[37] and 40 CFR 61 Subpart H.[38] Effluent sampling will be conducted to demonstrate compliance with the annual release limits in those standards.

The EPA implementation guidance for 40 CFR 191, Subpart A (EPA 402-R-97-001, Section 2.3 [40]) states "DOE must examine radiation doses to the public due to both actual normal operations and any unplanned or accidental release which occur during the reporting period." Further, EPA 402-R-97-001, Section 2.1[40] states, "Section 191.03(b) states that management and storage of transuranic waste at DOE facilities shall be conducted to provide reasonable assurance that the annual radiation dose to any member of the public in the general environment resulting from discharges of radioactive material and direct radiation from such management and storage shall not exceed specified limits." As shown in this SAR, only certain types of accidents have the capability of producing a dose to the public. The DOE has implemented a program that provides reasonable assurance that the radiation dose resulting from WIPP discharges to any member of the public in the general environment will not exceed 25 mrem (0.25 mSv) to the whole body and 75 mrem (0.75 mSv) to any critical organ (DOE/WIPP-00-3121 [41]).

The following discussion provides a comparison of the calculated dose consequences to the release standards. As the provisions of 40 CFR 191 Part A [37] 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 of 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 WHB and underground HEPA filtration systems, **the worst-case <u>mitigated</u> accident doses to the maximally exposed individual for all accidents analyzed, regardless of occurrence frequency, will be much less than the <u>annual</u> release limits imposed by 40 CFR 191 Subpart A [37] and 40 CFR 61, Subpart H [38].**

DOE will provide EPA with regularly scheduled reports summarizing the results of compliance sampling and dose calculations. As specified in the WIPP Land Withdrawal Act, reporting will be every two years, the Biennial Environmental Compliance Report (BECR) shall be the documentation in which the DOE provides data to EPA demonstrating compliance with 40 CFR 191, Subpart A. [37] Additional reporting information for Subpart A is documented in DOE/WIPP-00-3121.[41]

1.3.4 Analysis of Beyond the Design Basis Accidents

1.3.4.1 Operational Events

An evaluation of 72-B cask and 10-160B cask operational accidents "beyond" 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[8], BDBAs are simply those operational accidents with more severe conditions or equipment failure. Based on the analyses in Section 5.2.3, the operational accident scenario involving potential consequences to the non-involved worker, MEI, and immediate worker, whose frequency is less than 1E-06/yr is RH5, Fire followed by Explosion. A 10-160B accident was not selected for BDBA analysis because the radionuclide inventory for the 72B canister bounds that of a facility canister loaded with drums from a 10-160B road cask.

The source term MAR developed in Section 5.2.3 is based on the 72-B waste canister inventory derived in Section 5.1.2.1.2. The analyses assumed that based on the data in Appendix A, that the maximum radionuclide inventory in a 72-B waste canister is 80 PE-Ci for direct loaded waste and 240 PE-Ci for double contained waste. The on-site and off-site risk evaluation guidelines for the extremely unlikely range are used for the consequence evaluation even though the frequency of the BDBA scenarios is beyond extremely unlikely.

The worst case radiological consequences of RH5 are discussed here assuming that waste canister involved in the scenario is at 80 PE-Ci. The same assumptions regarding waste form combustible and noncombustible composition, damage ratio, airborne release fraction (median value instead of bounding), and respirable fraction are assumed. Substitution of these values into the consequence calculations for RH5, indicate doses of approximately 0.6 rem (6 mSv) to the noninvolved worker individual (less than one percent of the 100 rem noninvolved worker risk evaluation guideline for the extremely unlikely range), and 0.05 rem (.5 mSv) (less than one percent of 25 rem MEI risk evaluation guideline for the extremely unlikely range) to the MEI. The noninvolved worker and MEI doses are below their respective risk evaluation guidelines. The estimated 5.4 rem (54 mSv) dose to the immediate worker for the RH5 beyond design basis scenario (Appendix E, Table E-14) does not exceed the noninvolved worker risk evaluation guideline of 100 rem (1 Sv) for the extremely unlikely range. Therefore, no specific additional worker protection engineering or administrative controls are identified and the risk associated with this potential exposure is deemed acceptable.

1.3.4.2 Natural Phenomena

As discussed in Section 3.4.3 of DOE-STD-3009-94[8], natural phenomenon BDBAs are defined by a frequency of occurrence less than that assumed for the DBA. Since the DBT is defined with a 1,000,000 year return period, and the DBE with a 1,000 year return period, the most credible BDBA natural phenomenon event is an earthquake with a vertical ground acceleration of greater than 0.1 g (considered extremely unlikely). DBE SSCs: (1) the WHB structure, and (2) WHB 140/25-ton bridge crane, the CUR 25-ton crane, the Hot Cell crane, and the Facility Cask Loading Room grapple hoist, are assumed to fail resulting in a release of radioactive material.

The source term MAR developed in Section 5.2.3 is based on the 10-160B road cask inventory derived in Section 5.1.2.1.2. The analyses assumed that based on the data in Appendix A, that the maximum radionuclide inventory in a 10-160B road cask is 20 PE-Ci.

It is assumed that the WHB structure fails resulting in the Hot Cell roof collapsing into the Hot Cell resulting in damage to ten 10-160B RH waste drums, with a radionuclide inventory of 20 PE-Ci, awaiting placement in facility canisters and a partially loaded facility canister. The partially loaded facility canister contains two drums from two different 10-160B road casks is in the loading station. Each of the two drums in the facility canister contain the maximum radionuclide inventory of a 10-160B road cask. The total Hot Cell inventory is 60 PE-Ci. It is conservatively assumed that all of the drums and the partially loaded facility canister are breached by the falling Hot Cell roof debris and the Hot Cell crane.

The beyond DBE is basically the same accident as described for NC3-F, with the same MAR, waste form combustible and noncombustible composition, airborne release fraction, and respirable fraction. Using the NC3-F values and a factor of 10 increase in the damage ratio, the consequence calculations for beyond DBE indicate doses of approximately 24.7 rem (247 mSv) to the non-involved worker (approximately 25 percent of the 100 rem non-involved worker risk evaluation guideline for the extremely unlikely range), and 1.9 rem (19 mSv) (approximately 7.6 percent of 25 rem MEI risk evaluation guideline for the extremely unlikely range) to the MEI. The non-involved worker and MEI doses are below the risk evaluation guidelines, respectively. There is no postulated dose to the immediate worker since the event occurs in the Hot Cell which would not be occupied during 10-160B RH waste handling operations or when RH waste is stored there. Therefore, the radiological risk associated with a greater than 0.1 g earthquake is considered acceptable.

References for Section 1.3

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19. ANSI/ANS-8.3, Criticality Accident Alarm System.

20. ANSI/ANS-8.5, Use of Borosilicate-Glass Raschig Rings as a Neutron Absorber in Solutions of Fissile Material.

21. ANSI/ANS-8.7, Guide for Nuclear Criticality Safety in the Storage of Fissile Materials.

22. ANSI/ANS-8.15, Nuclear Criticality Control of Special Actinide Elements.

23. ANSI/ANS-8.19, Administrative Practices for Nuclear Criticality Safety.

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28. Nuclear Regulatory Commission Regulatory Guide 1.145, Rev 1, Atmospheric Dispersion Models for the Potential Accident Consequence Assessments at Nuclear Power Plants, United States Nuclear Regulatory Commission, Washington, DC, November 1982.

29. ANSI/ANS-51.1-1983, American National Standards Institute, Standard Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants.

30. 10 CFR 835, Occupational Radiation Protection.

31. 10 CFR 100, Reactor Site Criteria.

32. DOE O 151.1A, Comprehensive Emergency Management System, November 2000.

33. 40 CFR 261, Identification and Listing of Hazardous Waste.

34. 30 CFR 57, Safety and Health Standards - Underground Metal and Nonmetal Mines, 8th edition, 1994.

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Table 1.3-1 MEI Risk Evaluation Guidelines

Description	Estimated Annual Frequency of Occurrence	Description	Radiological Guidelines	Nonradiological Guidelines
Normal operations	$1 \geq f > 10^{-1}$			
Anticipated	$10^{-1} \geq f \geq 10^{-2}$	Incidents that may occur several times during the lifetime of the facility. (Incidents that commonly occur)	≤ 2.5 rem (25 mSv)	ERPG-1
Unlikely	$10^{-2} \geq f > 10^{-4}$	Accidents that are not anticipated to occur during the lifetime of the facility. Natural phenomena of this class include: Uniform Building Code-level earthquake, 100-year flood, maximum wind gust, etc.	≤ 6.5 rem (65 mSv)	ERPG-1
Extremely Unlikely	$10^{-4} \geq f > 10^{-6}$	Accidents that will probably not occur during the life cycle of the facility.	≤ 25 rem (250 mSv)	ERPG-2
Beyond Extremely Unlikely	$10^{-6} \geq f$	All other accidents.	No Guidelines	No Guidelines

Table 1.3-2 Noninvolved Worker Risk Evaluation Guidelines

Description	Estimated Annual Frequency of Occurrence	Description	Radiological Guidelines	Nonradiological Guidelines
Normal operations	$1 \geq f > 10^{-1}$			
Anticipated	$10^{-1} \geq f \geq 10^{-2}$	Incidents that may occur several times during the lifetime of the facility. (Incidents that commonly occur)	≤ 5 rem (50 mSv)	ERPG-1
Unlikely	$10^{-2} \geq f > 10^{-4}$	Accidents that are not anticipated to occur during the lifetime of the facility. Natural phenomena of this class include: Uniform Building Code-level earthquake, 100-year flood, maximum wind gust, etc.	≤ 25 rem (250 mSv)	ERPG-2
Extremely Unlikely	$10^{-4} \geq f > 10^{-6}$	Accidents that will probably not occur during the life cycle of the facility.	≤ 100 rem (1 Sv)	ERPG-3
Beyond Extremely Unlikely	$10^{-6} \geq f$	All other accidents.	No Guidelines	No Guidelines

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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 was upgraded to a DOE Field Office (CBFO), which 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 (MOC) (1985-2001).

- Washington TRU Solutions (WTS) to serve as MOC (2001 to Present)

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

In 1985, the DOE-AL contracted with Westinghouse to provide management and operating services as the MOC. In that capacity, Westinghouse was responsible for general management and operating services, including operational safety, engineering management, quality assurance and control, project control, construction management, environmental services, and ensured that all inputs to facility operations were properly reviewed for health, safety, and environmental implications.

In 2001, WTS was contracted by DOE-AL to serve as the MOC. WTS is responsible for providing general management and operating services, including operational safety, engineering management, quality assurance and control, project control, construction management, and environmental services. WTS also 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. The Working Agreement for Consultation and Cooperation (WACC[2]) provides detail about the SAR and provides for the Director of EEG to be the representative for the State. 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.

1.5 Safety Analysis Report Organization

The WIPP RH SAR was 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,[2] and the 17 chapter concept of DOE-STD-3009-94.[3] By applying the graded approach concepts as discussed in DOE-STD-3009-94,[3] 10 of the 20 DOE Order 5480.23[2] chapters and 7 of the 17 DOE-STD-3009-94[3] chapters were consolidated into other identified chapters. This resulted in a 10 chapter WIPP RH PSAR format that is similar to the WACC Agreement[1] format. This graded approach consolidation and reformatting is consistent with the discussion in DOE Order 5480.23[2] Attachment 1, Sections 4.f.(1)(c), and 4.f.(3)(d). PSAR chapter titles are renamed to follow selected DOE-STD-3009-94[3] or DOE Order 5480.23[2] 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 1.6-1 provides a correlation between the WACC Agreement SAR Format and Content requirements and the WIPP SAR format, and Table 1.6-2 provides a correlation between the WIPP RH SAR format, the SAR topics required by DOE Order 5480.23,[2] and DOE-STD-3009-94.[3] DOE-STD-3009-94[3] contains the format and content standard for documented safety analysis meeting the requirements of 10 CFR 830.[4]

References for Section 1.5

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

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

3. U.S. Department of Energy, DOE-STD-3009-94, Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports, (Change 1, January 2000)

4. 10 CFR 830, Nuclear Safety Management

Table 1.5-1, 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 CBFO direction.
2.5 Subsurface Hydrology	Deleted per CBFO direction.
2.6 Regional Geology	Deleted per CBFO direction.
2.7 Site Geology	Deleted per CBFO direction.
2.8 Vibratory Ground Motion	2.8 Vibratory Ground Motion
2.9 Surface Faulting	Deleted per CBFO direction.
2.10 Stability of Subsurface Materials and Foundations	Deleted per CBFO direction.
2.11 Slope Stability	2.5.2.5 Topography

Table 1.5-1, 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	RH 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 1.5-1, 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 Exhaust Filter Building 4.2.1.3 Water Pumphouse 4.2.1.4 Support Building 4.2.1.5 Support Structures
4.3 Shafts and Subsurface Facilities	4.2.2 Shaft and Hoist Facilities 4.2.3 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 Process Description 4.4 Confinement Systems 4.5 Safety Support Systems 4.6 Utility and Auxiliary Systems 4.7 Radioactive Waste (Radwaste) and Hazardous Waste Management
Ventilation	4.4.1 Confinement 4.4.2 Ventilation Systems
Electrical	4.6.1 Electrical System
Fire Protection	4.5.1 Fire Protection System
Waste Water	4.6.3 Domestic Water System 4.6.4 Sewage Treatment System 4.7 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 1.5-1, 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	CH SAR	
5.2	Remote-handled (RH) waste handling	4.3.1	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	5.1.5	Prevention of Inadvertent Nuclear Criticality
	Waste Logging	4.3.3	WIPP Waste Information System
5.6	Underground excavation	4.3.4	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 1.5-1, Consultation and Cooperation (WACC) Agreement/SAR Correlation 5 of 5

WACC Topic	SAR Section
Chapter 7 - Accident Analysis	
7.1 Accident classifications	5.2 RH TRU Accident Analysis
7.2 Source terms and analytical methods	5.2 RH TRU Accident Analysis
7.3 Accident descriptions and actual analyses	5.2 RH TRU Accident Analysis
Chapter 8 - Long Term Waste Isolation Assessment	5.3 Long-Term Waste Isolation Assessment
8.1 Identification of potential communication modes	5.3 Long-Term Waste Isolation Assessment
8.2 Modeling methods	5.3 Long-Term Waste Isolation Assessment
8.3 Consequence analyses	5.3 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 1.5-2, DOE Order 5480.23/ 10CFR830.204/ WIPP SAR Correlation **Page 1 of 2**

DOE Order 5480.23 Topics	10CFR830 Documented Safety Analysis DOE-STD-3009-94	WIPP SAR Chapter
Chapter 1 - Executive Summary	Unnumbered Executive Summary	Chapter 1 - Executive Summary
Chapter 3 - Site Characteristics	Chapter 1 - Site Characteristics	Chapter 2 - Site Characteristics
Chapter 4 - Facility Description and Operation	Chapter 2 - Facility Description	Chapter 3 - Principal Design and Safety Criteria Chapter 4 - Facility Design and Operation
Chapter 5 - Hazards Analysis and Classification of the Facility Chapter 11 - Analysis of Normal, Abnormal, and Accident Conditions	Chapter 3 - Hazard and Accident Analysis	Chapter 5 - Hazards and Accident Analysis
Chapter 4 - Facility Description and Operation	Chapter 4 - Safety Structures, Systems, and Components	Chapter 3 - Principal Design and Safety Criteria Chapter 4 - Facility Design and Operation
Chapter 16 - Derivation of Technical Safety Requirements	Chapter 5 - Derivation of Technical Safety Requirements	Chapter 6 - Derivation of Technical Safety Requirements
Chapter 8 - Inadvertent Criticality Protection	Chapter 6 - Prevention of Inadvertent Criticality	Chapter 5 - Hazards and Accident Analysis
Chapter 9 - Radiation Protection Chapter 11 - Analysis of Normal, Abnormal, and Accident Conditions	Chapter 7 - Radiation Protection	Chapter 7 - Radiological and Hazardous Material Protection
Chapter 10 - Hazardous Material Protection Chapter 11 - Analysis of Normal, Abnormal, and Accident Conditions	Chapter 8 - Hazardous Material Protection	Chapter 7 - Radiological and Hazardous Material Protection
Chapter 7 - Radioactive and Hazardous Material Waste Management Chapter 11 - Analysis of Normal, Abnormal, and Accident Conditions	Chapter 9 - Radioactive and Hazardous Waste	Chapter 7 - Radiological and Hazardous Material Protection
Chapter 15 - Initial Testing, In service Surveillance, Maintenance	Chapter 10 - Initial Testing, In-Service Surveillance, Maintenance	Chapter 8 - Institutional Programs

Table 1.5-2, DOE Order 5480.23/ 10CFR830.204/ WIPP SAR Correlation **Page 2 of 2**

DOE Order 5480.23 Topics	10CFR830 Documented Safety Analysis DOE-STD-3009-94	WIPP SAR Chapter
Chapter 17 - Operational Safety	Chapter 11 - Operational Safety	Chapter 8 - Institutional Programs
Chapter 13 - Procedures and Training	Chapter 12 - Procedures and Training	Chapter 8 - Institutional Programs
Chapter 14 - Human factors	Chapter 13 - Human Factors	Chapter 4 - Facility Design and Operation Chapter 5 - Hazards and Accident Analysis
Chapter 18 - Quality Assurance	Chapter 14 - Quality Assurance	Chapter 9 - Quality Assurance
Chapter 19 - Emergency Preparedness	Chapter 15 - Emergency Preparedness Program	Chapter 8 - Institutional Programs
Chapter 20 - Provisions for Decontamination and Decommissioning	Chapter 16 - Provisions for Decontamination and Decommissioning	Chapter 10 - Decontamination and Decommissioning
Chapter 12 - Management, Organization, Institutional Safety Provisions	Chapter 17 - Management, Organization, and Institutional Safety Provisions	Chapter 8 - Institutional Programs

Note 1 - WIPP SAR Chapter 3, Principal Design and Safety Criteria, addresses applicable statues, rules, and Departmental Orders, Safety Criteria, and Design Criteria. Chapter 3 supports the compliance aspects of each SAR chapter.

Note 2 - DOE Order 5480.23, Chapter 2, Applicable Statutes, Rules, and Departmental Orders, and Chapter 6, Principal Health and Safety Criteria, are incorporated into all applicable chapters of DOE-STD-3009-94.

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

Public Law 83-703	Atomic Energy Act of 1954, as amended
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, February 2001
10CFR Part 835	Occupational Radiation Protection, December 1993
29 CFR Part 1910	Occupational Safety and Health Standards, June 1974
30 CFR Part 57	Safety and Health Standards - Underground Metal and Nonmetal Mines, January 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 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 1985
40 CFR Part 261	Identification and Listing of Hazardous Waste, May 1980
40 CFR Part 262	Standards Applicable to Generators of Hazardous Waste, May 1980
40 CFR Part 264	Standards for Owners and Operators of Hazardous Waste Treatment, Storage, and Disposal Facilities, May 1980
40 CFR Part 265	Interim Status Standards for Owners and Operators of Hazardous Waste Treatment, Storage, and Disposal Facilities, May 1980
40 CFR Part 268	Land Disposal Restrictions, May 1980
40 CFR Part 270	EPA Administered Permit Programs: The Hazardous Waste Permit Program, April 1983
40 CFR Part 280	Technical Standards and Corrective Action Requirements for Owners and Operators of Underground Storage Tanks, September 1988
DOE O 151.1A	Comprehensive Emergency Management System
DOE O 232.1A	Occurrence Reporting and Processing of Operations Information.
DOE O 414.1A	Quality Assurance
DOE O 420.1	Facility Safety
DOE O 430.1A	Life-Cycle Asset Management
DOE O 433.1	Maintenance Management Program for DOE Nuclear Facilities
DOE O 435.1	Radioactive Waste Management
DOE O 451.1B	National Environmental Policy Act Compliance Program
DOE Order 5400.1	General Environmental Protection Program
DOE Order 5480.4	Environmental Protection, Safety, and Health Protection Standards,
DOE Order 5480.19	Conduct of Operations Requirements for DOE Facilities
DOE Order 5480.20A	Personnel Selection, Qualification, Training Requirements for DOE Nuclear Facilities
DOE Order 5480.21	Unreviewed Safety Questions
DOE Order 5480.22	Technical Safety Requirements
DOE Order 5480.23	Nuclear Safety Analysis Reports
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.