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Yucca Mountain Site Characterization Project

Identification of Structures, Systems, and Components Important to Safety at the Potential Repository at Yucca Mountain

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Printed October 1991

"Prepared by Yucca Mountain Site Characterization Project (YMSCP) participants as part of the Civilian Radioactive Waste Management Program (CRWM). The YMSCP is managed by the Yucca Mountain Project Office of the U.S. Department of Energy, DOE Field Office, Nevada (DOE/NV). YMSCP work is sponsored by the Office of Geologic Repositories (OGR) of the DOE Office of Civilian Radioactive Waste Management (OCRWM)."

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SAND89-7024
Unlimited Release
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IDENTIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS IMPORTANT TO SAFETY AT THE POTENTIAL REPOSITORY AT YUCCA MOUNTAIN

by

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ABSTRACT

This study recommends which structures, systems, and components of the potential repository at Yucca Mountain are important to safety. The assessment was completed in April, 1990 ana uses the reference repository configuration in the Site Characterization Plan Conceptual Design Report and follows the methodology required at that time by DOE Procedure AP6.10-Q. Failures of repository items during the preclosure period are evaluated to determine the potential offsite radiation doses and associated probabilities. Items are important to safety if, in the event they fail to perform their intended function, an accident could result which causes a dose commitment greater than 0.5 rem to the whole body or any organ of an individual in an unrestricted area. This study recommends that these repository items include the structures that house spent fuel and high-level waste, the associated filtered ventilation exhaust systems, certain waste-handling equipment, the waste containers, the waste treatment building structure, the underground waste transporters, and other items listed in this report.

This work was completed April 1990.

This document was written in response to DIM 60 and was completed under WBS 1.2.1.1, but was prepared under the earlier WBS 1.2.4.6.3. The work was completed in April 1990 and used DOE Administrative Procedures (APs) 6.9Q and 6.10Q, which later were superseded.

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ILLUSTRATIONS

EXECUTIVE SUMMARY

This study recommends which structures, systems and components of the potential repository at Yucca Mountain are important to safety. The assessment follows the method required by DOW Procedure AP-6.10Q (in Appendix A) and uses the Site Characterization Plan Conceptual Design Report (SNL, 1987) as a basis. This work was completed in April, 1990 and AP-6.10Q was superseded at a later date. The term "safety" refers to radiological safety, as described in the definition of "important" in 10CFR60.2 (NRC, 1989a).

In accordance with AP-6.10Q, the repository is divided into compartments, each of which is uniquely characterized by its inventory of radioactive materials, design features, and the operations conducted within it. Compartment locations are assigned to each item on the list of repository items subject to the quality level assignment process (DOE, 1989a).

Credible initiating events that are applicable to the potential Yucca Mountain repository are then identified on the basis of surveys of literature on safety analyses and reviews of previous assessments for the repository. Both external and internal initiation events are considered, including earthquakes, tornadoes and extreme winds, floods, aircraft crashes, fires, loss of electrical power, operator errors, and equipment failures. Event trees are then developed for each compartment to depict the relevant combinations of failures of items that could result form various initiating events. Only items whose failure could affect releases of radioactive material are included in the event trees.

Radiation doses at the nearest boundary of the unrestricted area are calculated for each scenario in the event trees, and an assessment is made as to whether each scenario is credible. To determine which repository items are important to safety, this assessment initially assumes that repository items would not withstand severe credible events unless the items are important to safety, this assessment initially assumes that repository items would not withstand severe credible events unless the items were identified as important to safety and designed and constructed accordingly. Because items that are not important to safety would not

necessarily be designed or constructed to withstand severe credible events, these items could fail as a result of such events.

In accordance with NCR (1989b), when accident analyses take credit for an item/s function to prevent or mitigate the release of radioactivity (or to prevent criticality), the item should be designated as important to safety, placed on a Q-list, and subjected to an appropriate QA program. This study recommends that these repository items include the structures that house spent fuel and high-level waste, the associated filtered ventilation exhaust systems, certain waste-handling equipment, the waste containers, the waste treatment building structure, the underground waste transporters, and other items listed in this report.

The items important to safety should be designed, fabricated, and constructed to more stringent requirements than other items that are not important to safety. For example, items important to safety should be designed and constructed to maintain their safety functions during credible initiating events. These events form the basis for design of items important to safety and are called design basis accidents. This study identifies several references that describe design basis accidents and general methods to select the associated design conditions.

1.0 INTRODUCTION

1.1 Purpose

The purpose of this assessment is to recommend which structures, systems, and components of the potential Yucca Mountain repository are important to safety, as defined in 10CFR60.2 (NRC, 1989a). The assessment will contribute to the development of design criteria and quality assurance (QA) requirements for the repository. The results can also be used in the next phase of repository design.

1.2 Scope

This radiological safety analysis considers failures of repository structures, systems, and components during preclosure operations, including the waste emplacement, caretaking, and decommissioning periods. The assessment follows the method required by DOE Procedure AP-6.10Q (Appendix A) and uses the Site Characterization Plan Conceptual Design Report (SCP-CDR) (SNL, 1987) as a basis. This work was completed in April, 1990 and AP-6.10Q was superseded at a later date. The results of the assessment include the following:

- A list of repository structures, systems, and components important to safety (hereinafter called repository items important to safety)

- A list of repository items not important to safety, which were evaluated as candidate items subject to the quality level assignment process

- The report documentation (including a list of references and sources of information) demonstrating that each step of the DOE procedure has been completed

- Safety recommendations that result from the analysis, including recommended design requirements (for items important to safety), operational limitations, and QA stringency

1.3 Organizational Approach

This report is organized into eight sections, including this introduction. The remaining seven sections are as follows:

- Section 2.0, Bases for the Assessment, which describes the repository facilities, defines "important to safety," and identifies the method and assumptions used in this assessment

- Section 3.0, Development of Potential Accident Scenarios, which describes the repository compartments, identifies potential initiating events, and develops event trees showing potential failures that could result in accidental releases of radioactivity

- Section 4.0, Event Tree Analyses, which evaluates the consequences and probabilities of accident scenarios

- Section 5.0, Identification of Items Important to Safety, which presents the assessments that identify repository items important to safety

- Section 6.0, Safety Recommendations, which describes the recommended design requirements and operational limitations that result from the analyses

- Section 7.0, Conclusions, which summarizes the results of the assessment and identifies areas requiring further evaluation

- Section 8.0, References, which presents a list of references and sources of information

DOE Procedure AP-6.10Q, Identification of Items Important to Safety, is attached as Appendix A.

2.0 BASES FOR THE ASSESSMENT

This section describes the repository facilities, defines the term "important to safety," summarizes the method used to identify items important to safety, and presents the major assumptions used in this study.

2.1 Facility Description

The reference configuration of the repository corresponds to the conceptual design documented in the SCP-CDR (SNL, 1987). This subsection describes the repository facilities and systems as they would appear if the conceptual design were followed. Additional information is included in SNL (1987).

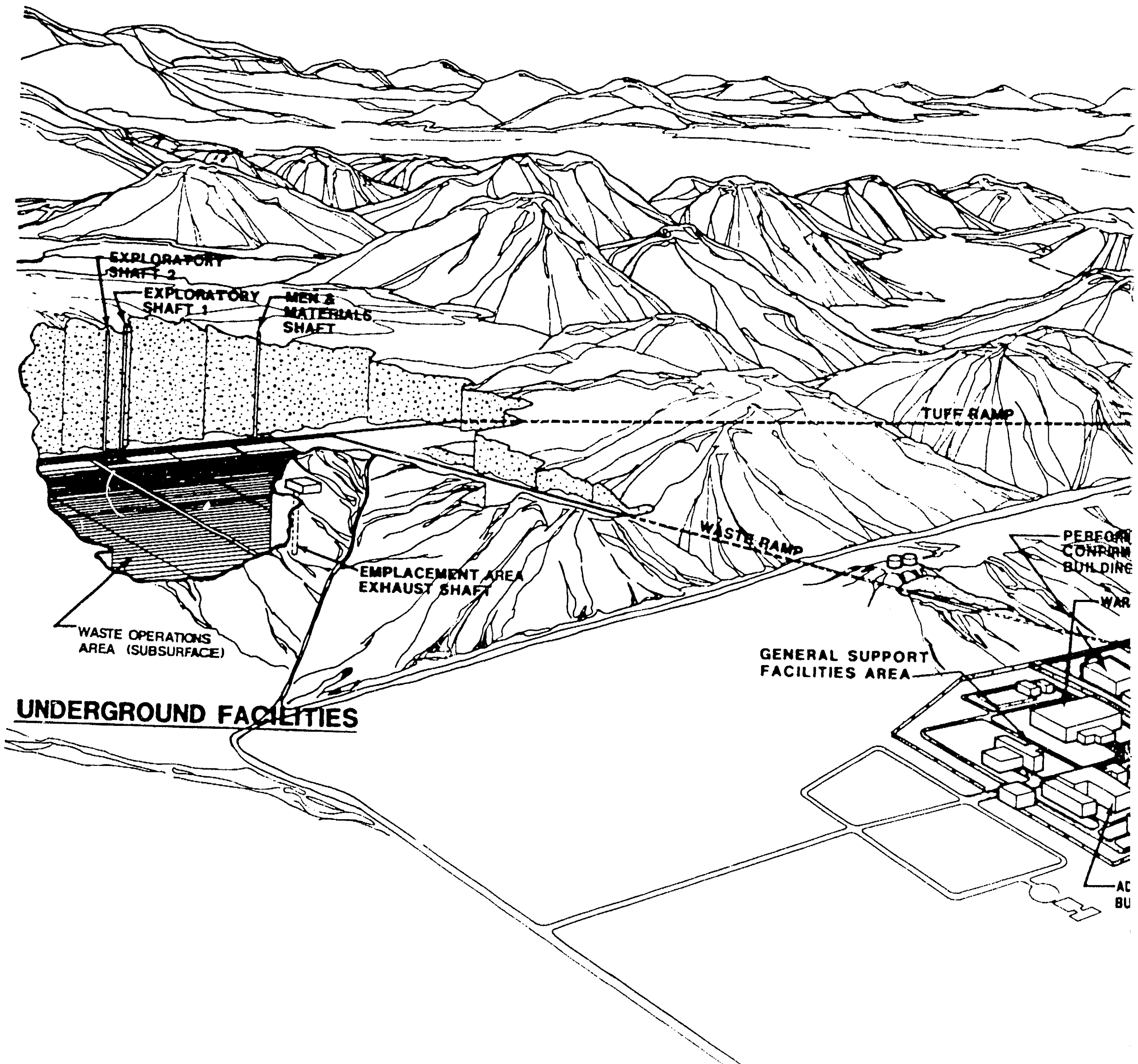
2.1.1 General

The repository design is based on the annual receipt of 3,000 metric tons of uranium (MTU) of spent fuel and 400 MTU of vitrified high-level waste (HLW). The spent fuel will be received as pressurized water reactor (PWR) and boiling water reactor (BWR) fuel assemblies. The fuel pins will be removed from most of the assemblies at the repository and consolidated into containers for disposal. The vitrified HLW will be received in canisters and overpacked in containers for disposal. A total of up to 70,000 MTU of spent fuel and vitrified HLW will be received by rail and truck over the operating life of the facility.

The overall site is composed of surface and underground facilities linked by a combination of shafts and ramps, as shown in the perspective sketch and overall site plan in Figure 2-1. The surface and underground facilities are generally described below.

2.1.2 Surface Facilities

The surface facilities include the facilities at the tops of the shafts and ramps, and the central surface facilities. The central surface facilities are located on gently sloping terrain at the


EXPLORATORY SHAFT 2
EXPLORATORY SHAFT 1
MEN & MATERIALS SHAFT
TUFF RAMP
WASTE RAMP
EMPLACEMENT AREA EXHAUST SHAFT
WASTE OPERATIONS AREA (SUBSURFACE)
GENERAL SUPPORT FACILITIES AREA
UNDERGROUND FACILITIES

TUFF

PERS

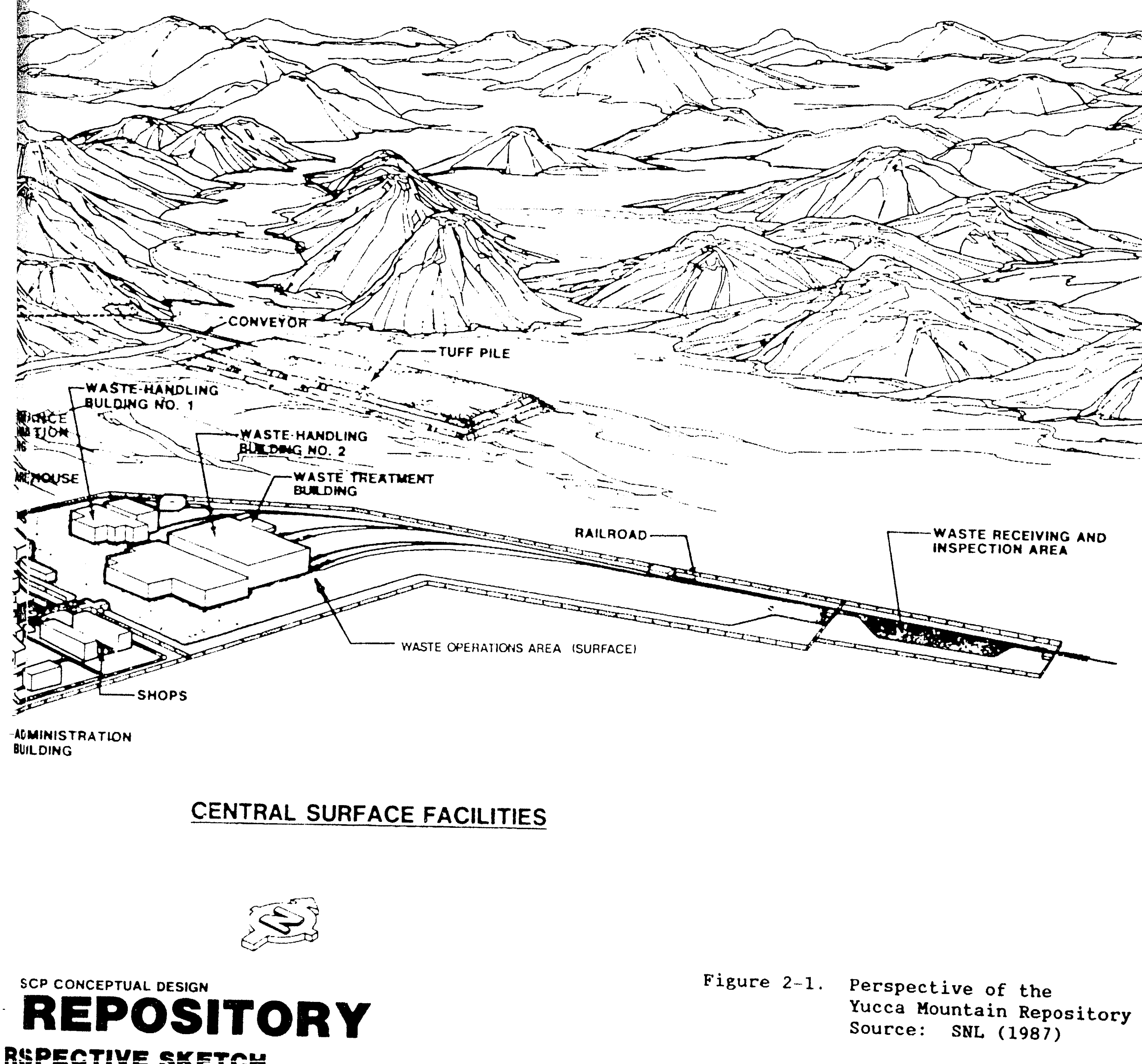


Figure 2-1. Perspective of the Yucca Mountain Repository
Source: SNL (1987)

eastern base of Yucca Mountain. The facilities are divided into three adjacent areas: the waste receiving and inspection area, the waste operations area, and the general support facilities area. Shipping casks are received and inspected in the waste receiving and inspection area prior to transfer into the waste operations area. The waste operations area includes the rail yard and truck parking lot and the surface waste-handling facilities where radioactive materials are stored or handled. The general support facilities area includes administrative facilities and other support facilities where no radioactive materials are present.

The surface facilities that handle and store radioactive materials include five major areas:

- Waste-handling building #1
- Waste-handling building #2
- Waste treatment building
- Performance confirmation building
- Cask transport area

Shipping casks are transported from parking areas in the waste operations area to the receiving bays in one of the waste-handling buildings. A two-stage approach to repository construction requires two waste-handling buildings. For the first several years of operation, only waste-handling building #1 (WHB-1) will be available for waste-handling functions. During this period, construction of waste-handling building #2 (WHB-2), a full-capacity building, will be completed.

WHB-1 is designed to receive and prepare 400 MTU/yr of spent fuel or vitrified HLW for subsurface emplacement. During Stage 1, spent fuel assemblies will be processed without any consolidation. When WHB-2 begins operations during Stage 2, WHB-1 will handle only defense HLW shipments and a relatively small amount of West Valley HLW. WHB-2 will be dedicated to receiving, consolidating, packaging, and handling spent fuel.

The waste-handling buildings contain receiving and shipping bays where casks are removed from their carriers and prepared for unloading, shielded hot cells where spent fuel and HLW are handled and packaged, and other supporting facilities. The cask receiving and shipping bays are steel-framed structures. The hot cells are reinforced concrete structures.

The waste treatment building receives site-generated radioactive wastes, both liquid and solid, and processes these wastes into a solid form suitable for disposal.

The performance confirmation building is located near the portal of the waste-handling ramp. Containers of spent fuel and HLW are removed from emplacement for observation, testing, and evaluation as part of the repository performance confirmation program. Retrieved waste containers are received, sampled, tested, and repackaged in the performance confirmation building for return and emplacement underground.

The cask parking lot includes space for 315 truck trailers and 70 railcars to provide interim storage of up to 6 weeks of waste throughput.

2.1.3 Underground Facilities

The underground facilities are located under the ridgeline of Yucca Mountain. A ramp is used for transporting waste from the surface to the underground disposal area. Another ramp is used to convey mined tuff to the surface. Four vertical shafts (two exploratory shafts, an emplacement exhaust shaft, and a men-and- materials shaft) are located near the northeast boundary of the underground waste emplacement or disposal area. The shafts are used for underground ventilation and for access of personnel, supplies, and equipment.

During repository operations, mining development areas are separated from the underground emplacement areas (where radioactive materials are handled) by partitions. These partitions serve as ventilation and access barriers. No radioactive materials are permitted in the underground mining development area. To prevent leakage of potential radioactive contamination, ventilation air in the emplacement area is maintained at a lower pressure than ventilation in the development area.

Underground development includes mining, installation of ground support (e.g., rock bolts), excavation of emplacement boreholes, installation of utilities, and construction of support facilities, such as shops, warehouses, and offices. In the emplacement area of the underground repository, containers are placed in vertical boreholes in the floor of each emplacement drift. The underground repository is divided by three main drifts - the waste main, the tuff main, and the service main. A perimeter drift encircles the repository. Each emplacement panel is bounded by parallel panel access drifts, the perimeter drift, and one of the main drifts. Parallel emplacement drifts, containing the emplaced waste, run perpendicular to the access drifts and are equally spaced for the length of the panel.

Other areas in the underground facility include emplacement service shops, development service shops, training areas, a decontamination area, and a performance confirmation area.

2.2 Definition of "Important to Safety"

10CFR60 defines "important to safety," with reference to structures, systems, and components, as "those engineered structures, systems, and components essential to the prevention or mitigation of an accident that could result in a radiation dose to the whole body, or any organ, of 0.5 rem or greater at or beyond the nearest boundary of the unrestricted area at any time until the completion of permanent closure" (NRC, 1989a).

NRC (1983a) provides further clarification that "structures, systems, and components are important to safety if, in the event they fail to perform their intended function, an accident could result which causes a dose commitment greater than 0.5 rem to the whole body or any organ of an individual in an unrestricted area." In this report, the terms "radiation dose" and "radiation exposure" are used with the same meaning as "committed dose equivalent."

The items important to safety are to be placed on a Q-list, as required by AP-6.10Q (Appendix A). Additional NRC guidance on the Q-list is provided in NUREG-1347 (NRC, 1989b): "The primary purpose of developing a Q-list is to assure that those structures, systems, and components essential to prevent or mitigate the release of radionuclides to the environment are subject to appropriate quality control." Also, "it is the NRC staff position that those items for which DOE is taking credit in the prevention or mitigation of release of radionuclides should be subject to a 10CFR50, Appendix B (or equivalent) QA program." In addition, "the Q-list should include significant items such as the 'design' to preclude criticality" (i.e., features that prevent criticality), as stated in NRC (1989b).

In summary, the above NRC guidance indicates that when accident analyses take credit for an item's function to prevent or mitigate the release of radioactivity (or to prevent criticality), the item should be designated as important to safety, placed on a Q-list, and subjected to an appropriate QA program.

2.3 Method

This assessment follows the method required by DOE Procedure AP-6.10Q, Identification of Items Important to Safety (see Appendix A). The method involves 13 steps, as described in Table 2-1. The table identifies the subsections of this report in which each step of the assessment is discussed.

TABLE 2-1

PROCEDURAL STEPS FOR
IDENTIFYING ITEMS IMPORTANT TO SAFETY

Step	Description	Subsection
1	Select Documented Design Configuration	2.1
2	Define Compartments	3.1
3	Assign Compartment Locations to Items	3.1
4	Identify and Screen Initiating Events	3.2
5	Develop Event Trees	3.3
6	Estimate Doses	4.1
7	Classify Scenarios as Credible or Not Credible	4.2
8	Identify Credible Scenarios Exceeding the Dose Criterion (Q-Scenarios)	5.1
9	Identify Any Other Scenarios Exceeding Other Criteria (Q-Scenarios)	5.1
10	Eliminate NQ-Scenarios from Further Analysis	5.2
11	Evaluate Q-Scenarios to Identify Items Important to Safety	5.3
12	Construct a List of Items Important to Safety	5.3
13	Iterate the Above Steps for Future Stages of the Design	7.2

Source: Appendix A

In Step 1, a documented design configuration is selected for the repository. This configuration is described in Subsection 2.1.

In Steps 2 and 3, the repository design configuration is separated into compartments. Compartment locations are assigned to each repository structure, system, and component included on the Candidate List of Items and Activities Subject to the Quality Level Assignment Process (DOE, 1989a). These steps are described in Subsection 3.1. (Note that the identification of items important to safety is part of the "quality level assignment process.")

In Step 4, initiating events are identified and screened. External and internal initiating events are discussed in Subsection 3.2.

In Step 5, event trees are developed for credible initiating events that could lead to a significant radiological release. Subsection 3.3 describes the event trees for accidents involving failures of repository items.

In Step 6, radiation doses are calculated to determine the consequences of each scenario in the event trees. Subsection 4.1 describes the radiation dose calculations.

In Step 7, scenarios are classified as credible or not credible, as described in Subsection 4.2.

In Steps 8 and 9, credible scenarios that exceed the dose criterion of 0.5 rem or other specified criteria are identified. These scenarios are designated as Q-scenarios. These steps are described in Subsection 5.1.

In Step 10, other accident scenarios that either are not credible or do not exceed the dose criterion are designated as NQ-scenarios and are eliminated from further analyses used to identify items important to safety. This step is discussed in Subsection 5.2.

In Step 11, Q-scenarios are evaluated to identify the specific items that would prevent or mitigate the events and thus be classified as important to safety. This evaluation is presented in Subsection 5.3.

In Step 12, the list of items important to safety is developed. This is discussed in Subsection 5.3. A list of items that are not important to safety is also included.

In Step 13 (the last step), the above steps are iterated for future stages of the design. This step and other areas requiring further analyses are described in Subsection 7.2.

This report (and the list of references in Section 8.0) comprises the documentation that demonstrates that each of the above steps has been completed.

2.4 Assumptions

Major assumptions used in this assessment are as follows:

- The SCP-CDR conceptual design (SNL, 1987) is used as the reference repository configuration.

- To identify items important to safety, all repository items are initially assumed to be designed and constructed to requirements for items that are not important to safety (i.e., not designed or constructed specifically to withstand maximum credible events). Appropriate changes in design requirements are then recommended for any items that are identified by analysis as important to safety.

- Operations at the geologic repository are carried out at the maximum capacity and rate of receipt of radioactive waste, per 10CFR60.21(c) (NRC, 1989a), which corresponds to 3,000 MTU/yr of spent fuel and 400 MTU/yr of HLW.

- In the SCP-CDR, the repository areas containing radioactive materials are surrounded by a fence, which serves to provide access controls for purposes of radiological protection. This fence is about 100 m from surface facilities containing spent fuel and HLW, and is assumed to be the boundary between the restricted and unrestricted areas.

- Although the containers that hold spent fuel and HLW canisters are not part of the repository (DOE, 1989a), they are included in this analysis because their failure to perform a containment function can affect accidental releases of radioactivity during the preclosure period.

- Shipping casks are not part of the repository and will be licensed separately in accordance with 10CFR71 (NRC, 1989c). Therefore, they are not assessed in this study to determine if they are important to safety. This assessment assumes that shipping casks will function (i.e., provide confinement and protect the waste) during accidents when the casks are bolted and sealed, unless the casks could be subjected to accidental conditions that exceed their design basis.

- The shipping casks will be critically safe when fully flooded.

3.0 DEVELOPMENT OF POTENTIAL ACCIDENT SCENARIOS

3.1 Repository Compartments

To facilitate the assessment, the facility is divided into 20 compartments (Step 2 of Table 2-1), as listed in Table 3-1. Each compartment is uniquely characterized by its waste forms and inventories, design features, and associated operations and functions during the preclosure period. The locations of these compartments are shown in Figures 3-1 through 3-4.

A listing of repository items subject to the quality level assignment process (DOE, 1989a), along with their respective compartment locations, is given in Table 3-2. Most of the items listed in the table are part of the exploratory shaft facility (ESF). However, all ESF items will be removed prior to repository operations, except underground openings (shafts and excavations), shaft liners, and ground support (DOE, 1987a). The remaining repository items listed in the table are identified at a very general level (e.g., facilities and buildings); however, in this assessment, these items are further subdivided to a more detailed level to identify the structures, systems, and components important to safety.

3.2 Initiating Events

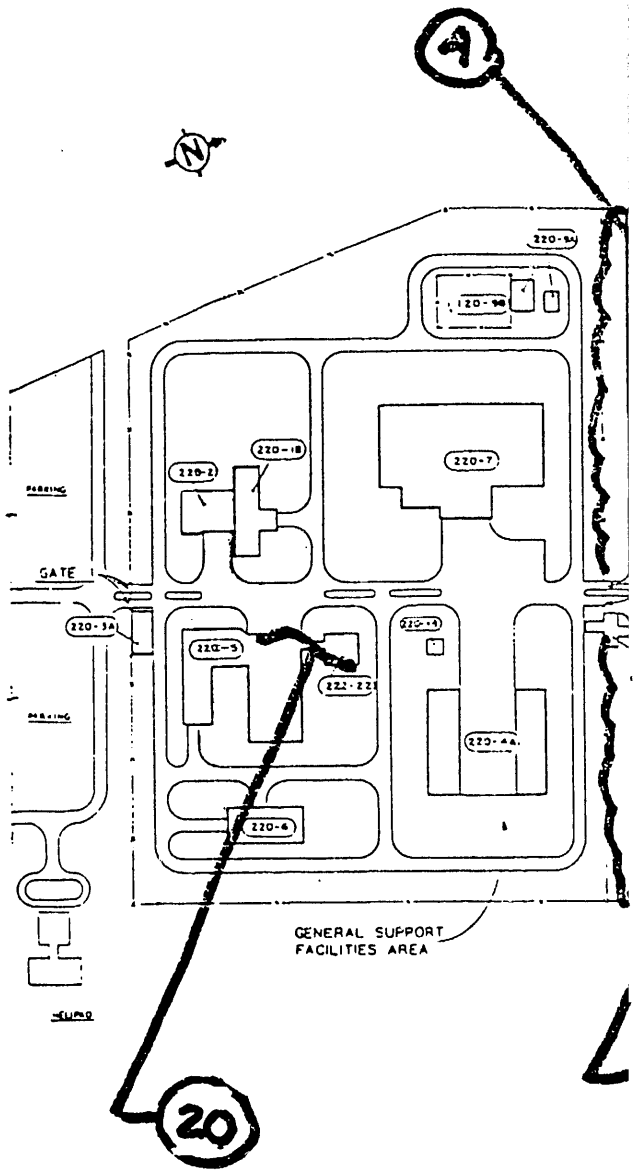
In this section, both external and internal initiating events are evaluated to identify the ones applicable to the Yucca Mountain repository. External events are those caused by natural phenomena or human activities external to the repository (as defined in Appendix A). Internal events are those caused by failures or operator activities at the repository.

Compartment	
1. Cask Transportation Areas	Spent fuel and Spent fuel an
2. Decontamination Building	Site-generated
3. Waste Treatment Building	Radioactive s Radioactive li
4. Performance Confirmation Building	Spent fuel (b
5. WHB-1 Cask Receiving and Shipping Air Lock	Spent fuel and Site-generated
6. WHB-1 Cask Receiving and Shipping Bay	Spent fuel and Site-generated
7. WHB-1 Unloading Hot Cell	Spent fuel and
8. WHB-1 Surface Storage Vault	Spent fuel and
9. WHB-1 Support Facilities	Site-generated
10. WHB-2 Cask Receiving and Shipping Air Lock	Same as Com
11. WHB-2 Cask Receiving and Shipping Bay	Same as Com
12. WHB-2 Unloading Hot Cell	Same as Com
13. WHB-2 Consolidation Hot Cell	Spent fuel
14. WHB-2 Packaging Hot Cell	Spent fuel in s
15. WHB-2 Surface Storage Vault	Spent fuel in
16. WHB-2 Support Facilities	Same as Com
17. Underground Waste Emplacement Access Drifts	Spent fuel an
18. Underground Waste Emplacement Drifts	Spent fuel an and in t
19. Underground Development Facilities	None
20. Other Repository Support Facilities	None

TABLE 3-1

REPOSITORY COMPARTMENTS

Waste Types	Major Items	Major Operations and Functions
HLW in shipping casks HLW containers in underground transporter	Underground transporter Waste container Site-generated waste vehicles Other items not affecting radiological releases	Cas. parking Transfer of waste among the cask parking lot, WHBs, waste treatment building, and waste ramp portal
waste with less than 5 Ci of Co-60	Building structure Ventilation system Other items not affecting radiological releases	Decontamination of contaminated items
id waste comprised of Co-60 uid waste comprised of Co-60	Building structure Fire protection system Waste handling equipment Filtered ventilation exhaust system Other items not affecting radiological releases	Receipt, interim storage, processing, and packaging of site-generated waste
e assemblies and containers)	Building structure Waste handling equipment Waste container Filtered ventilation exhaust system Other items not affecting radiological releases	Receipt of containers, examination, and repackaging
HLW in shipping casks waste in transfer casks	Building structure Waste handling equipment (cask for site-generated waste) Other items not affecting radiological releases	Ingress and egress of shipping casks for WHB
HLW in shipping casks (bolted and unbolted) waste in transfer casks	Building structure Waste handling equipment (bridge crane, cask transfer cart, etc.) Other items not affecting radiological releases	Removal of casks from vehicles Cask preparation for unloading
HLW	Same as Compartment 4	Removal of waste from casks Interim storage and packaging of waste
HLW in containers	Building structure Waste handling equipment Waste container Filtered ventilation exhaust system Other items not affecting radiological releases	Interim storage of containers Transfer of containers
waste with less than 5 Ci of Co-60	Building structure Ventilation system Other items not affecting radiological releases	Support for hot cell operations and maintenance
partment 5	Same as Compartment 5	Same as Compartment 5
partment 6	Same as Compartment 6	Same as Compartment 6
partment 7	Same as Compartment 7	Same as Compartment 7
	Building structure Waste handling equipment Filtered ventilation exhaust system Other items not affecting radiological releases	Consolidation of spent fuel assemblies
in sealed and unsealed containers	Same as Compartment 4	Packaging of consolidated spent fuel into containers
in containers	Same as Compartment 8	Same as Compartment 8
Compartment 9	Same as Compartment 9	Same as Compartment 9
and HLW containers in underground transporter	Drift structure Underground transporter Waste container Filtered ventilation exhaust system Other items not affecting radiological releases	Movement of transporter between surface and underground emplacement drifts
and HLW containers in underground transporter in the emplacement borehole	Drift structure Underground transporter Waste handling equipment Waste container Filtered ventilation exhaust system Other items not affecting radiological releases	Movement of transporter to emplacement borehole Emplacement and retrieval of waste containers
	Drift structure Tuff handling systems Other items not affecting radiological releases	Development of underground Support for underground operations
	Various structures and facilities	Various supporting functions and operations



9
N
220-7
220-18
220-2
GATE
220-3A
220-5
220-4A
220-6
GENERAL SUPPORT
FACILITIES AREA
20

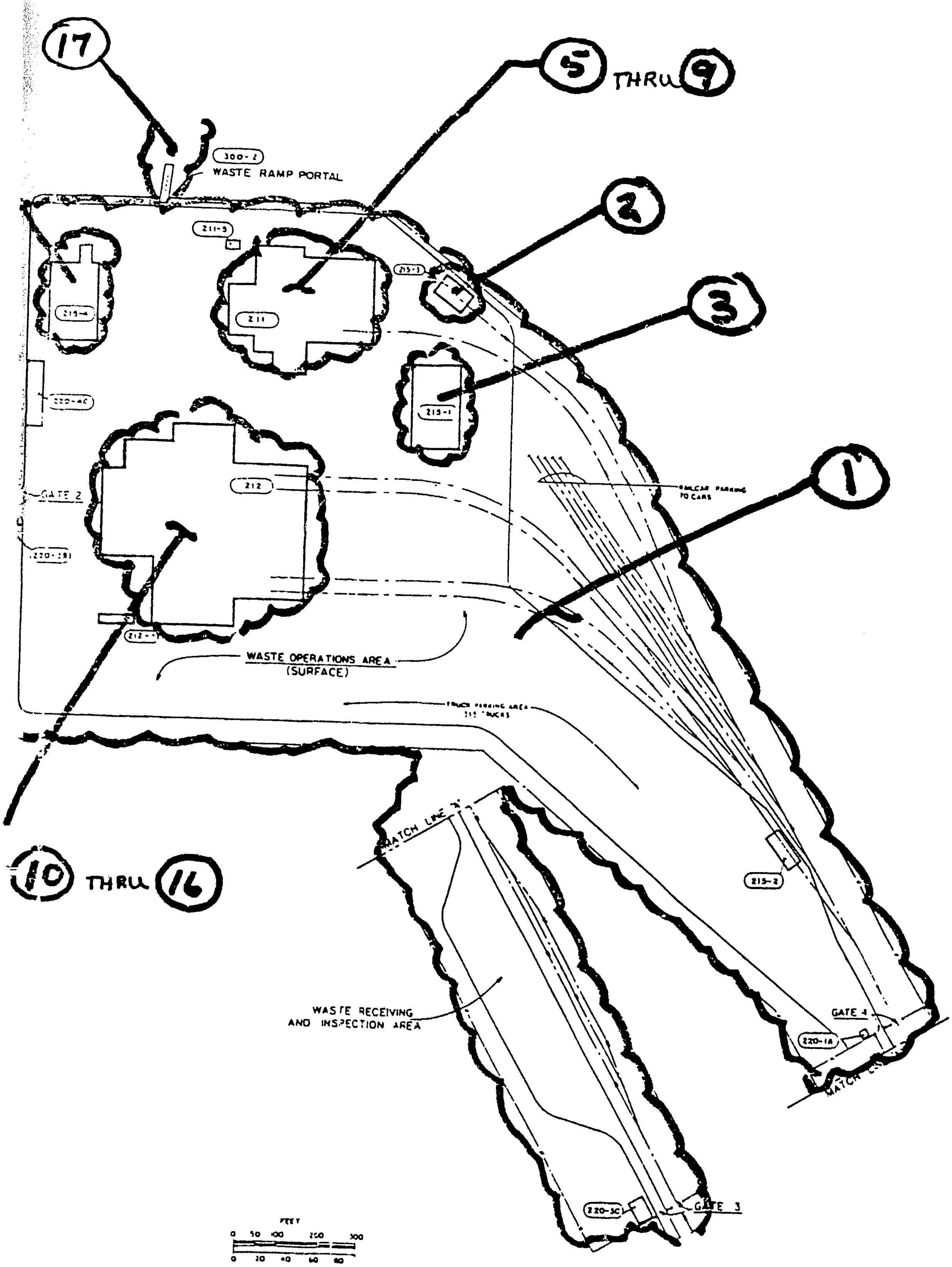


Figure 3-1. Compartments of the Central Surface Facilities

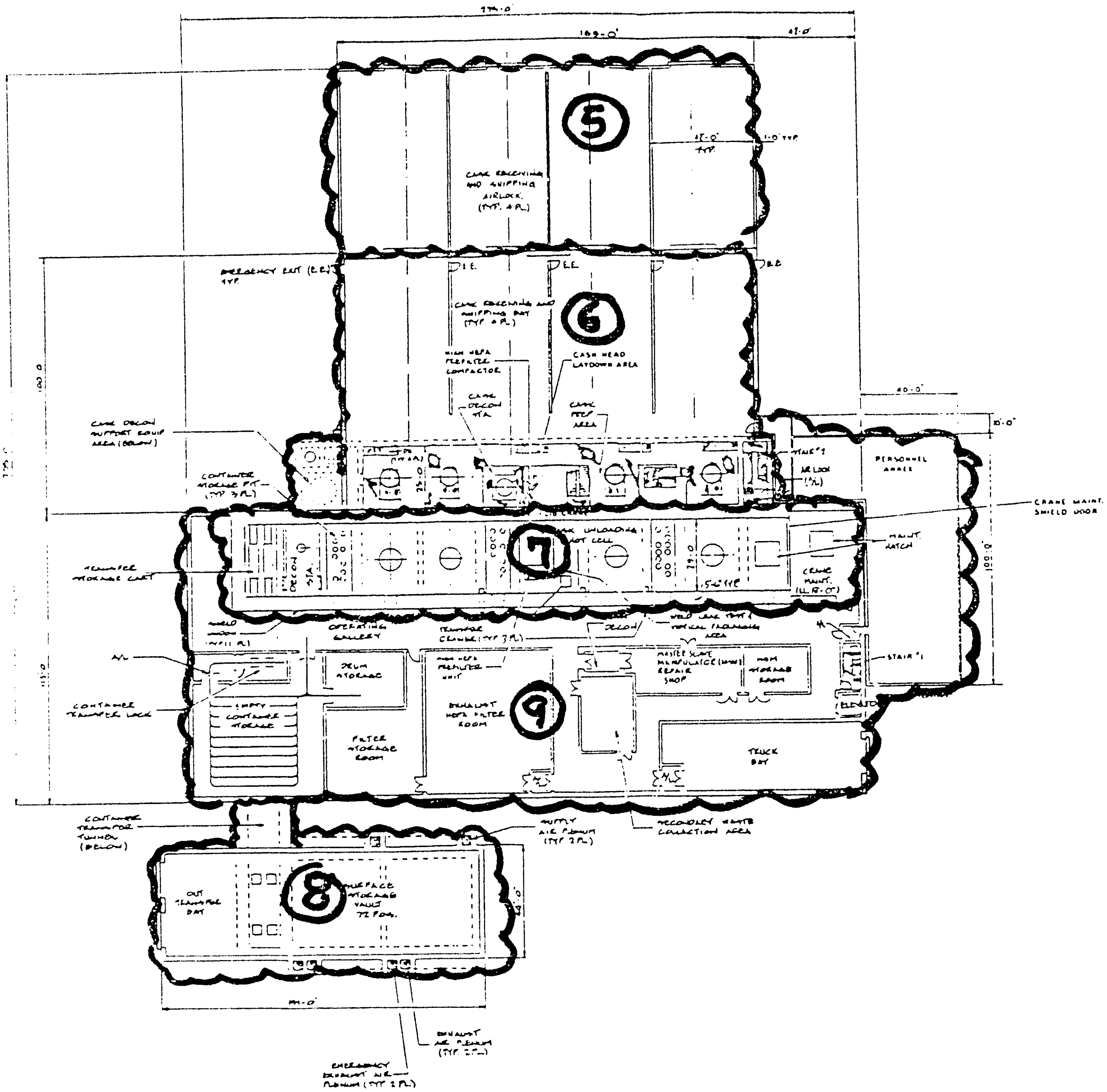
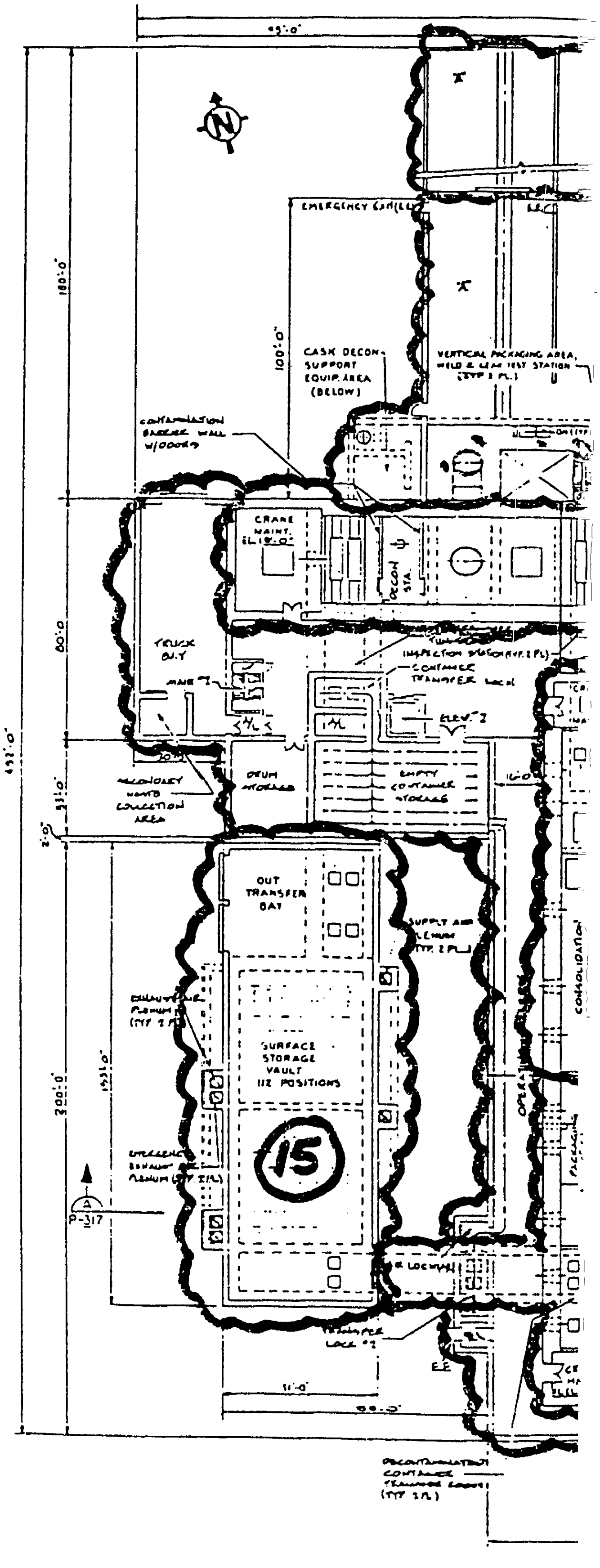


Figure 3-2. Compartments of WHB-1



45'-0"
N
EMERGENCY EXHAUST
180'-0"
100'-0"
CASK DECON- SUPPORT EQUIP. AREA (BELOW)
VERTICAL PACKAGING AREA, WELD & LEAK TEST STATION (TYP 2 PL.)
CONTAMINATION BARRIER WALL W/DOORS
CRANE MAINT. EL. 18'-0"
DECON STA.
80'-0"
TRUCK BAY
INSPECTION STATION (TYP. 2 PL.)
CONTAINER TRANSFER LOCK
ELEV. #2
SECONDARY WASTE COLLECTION AREA
DRUM STORAGE
EMPTY CONTAINER STORAGE
16'-0"
2'-0"
OUT TRANSFER BAY
SUPPLY AIR PLENUM (TYP. 2 PL.)
EXHAUST AIR PLENUM (TYP 2 PL.)
SURFACE STORAGE VAULT 112 POSITIONS
CONSOLIDATION
300'-0"
153'-0"
15
EMERGENCY EXHAUST AIR PLENUM (TYP 2 PL.)
A
P-317
TRANSFER LOCK #2
51'-0"
DECONTAMINATION CONTAINER TRANSFER ROOM (TYP 2 PL.)

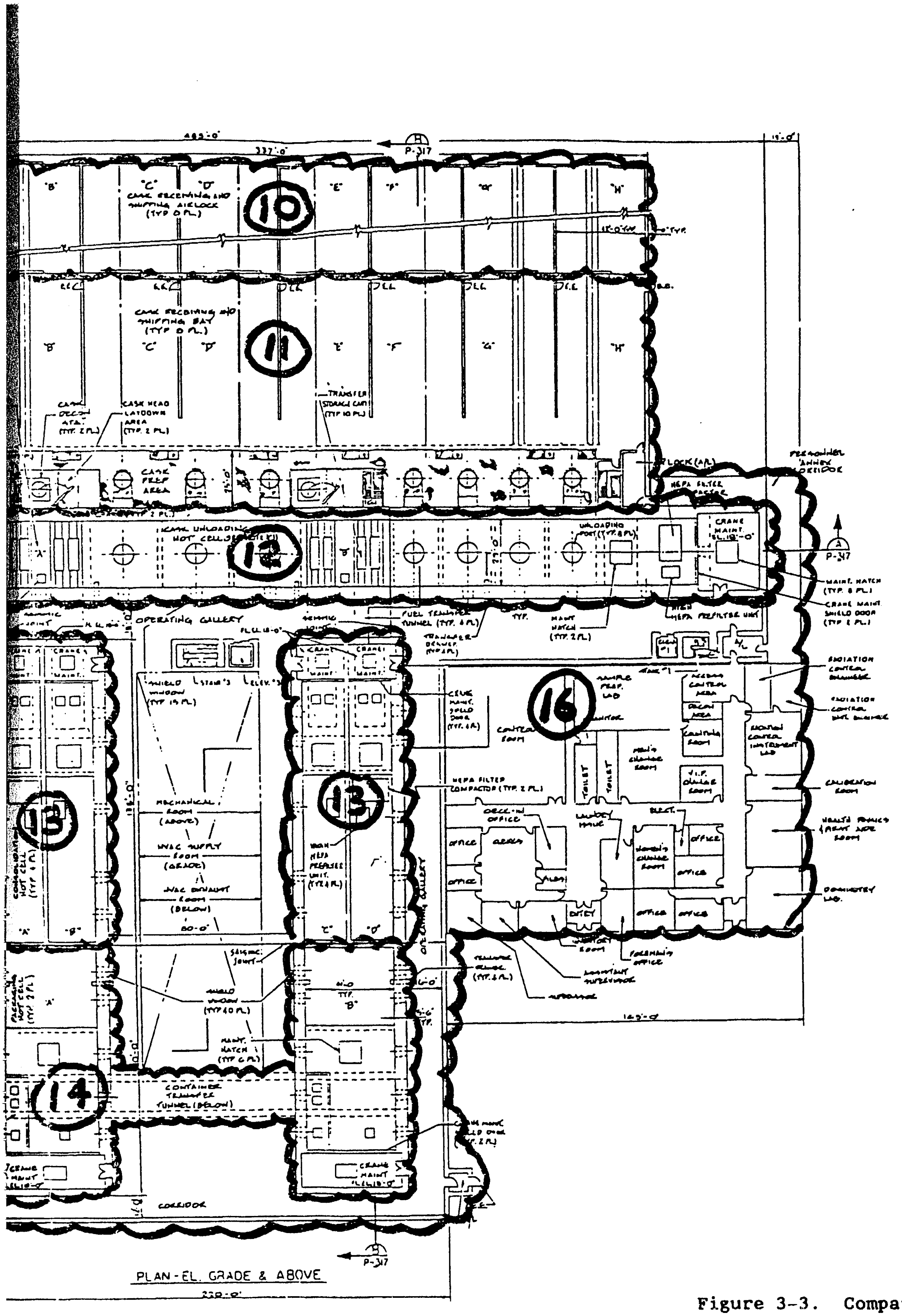
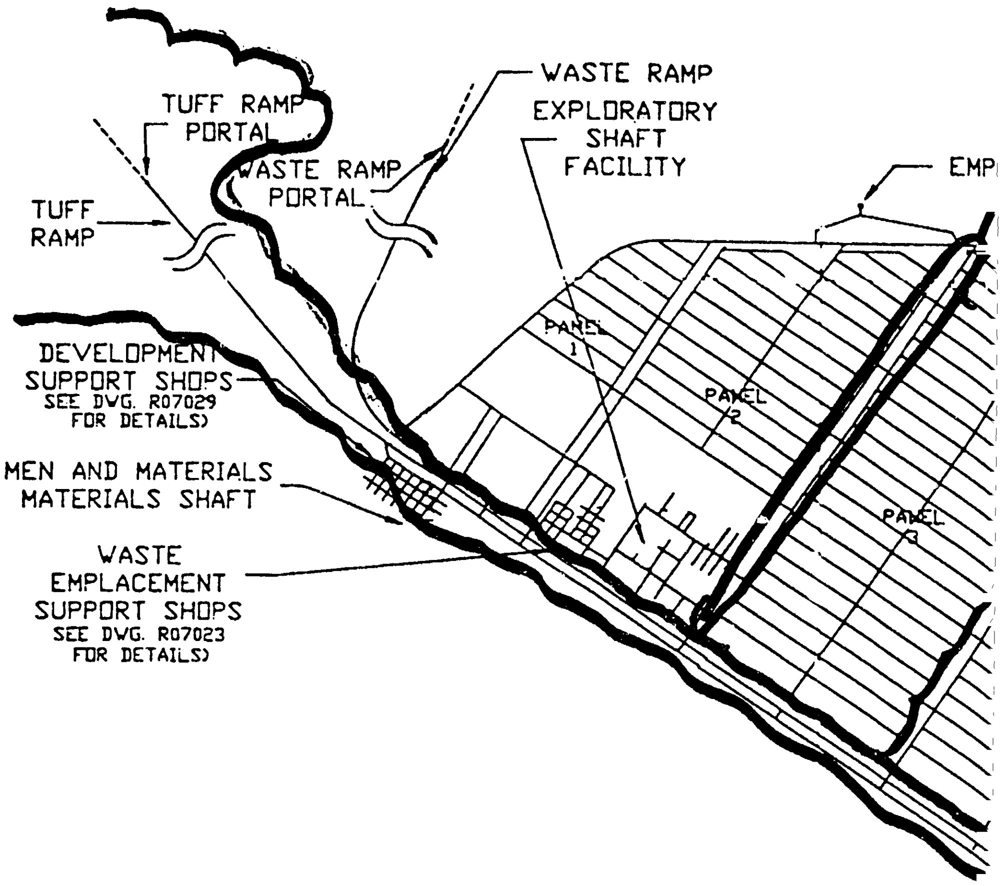


Figure 3-3. Compartments of WHB-2



WASTE RAMP
EXPLORATORY SHAFT FACILITY
TUFF RAMP PORTAL
WASTE RAMP PORTAL
TUFF RAMP
DEVELOPMENT SUPPORT SHOPS
SEE DWG. R07029 FOR DETAILS)
MEN AND MATERIALS MATERIALS SHAFT
WASTE EMPLACEMENT SUPPORT SHOPS
SEE DWG. R07023 FOR DETAILS)
PANEL 1
PANEL 2
PANEL 3

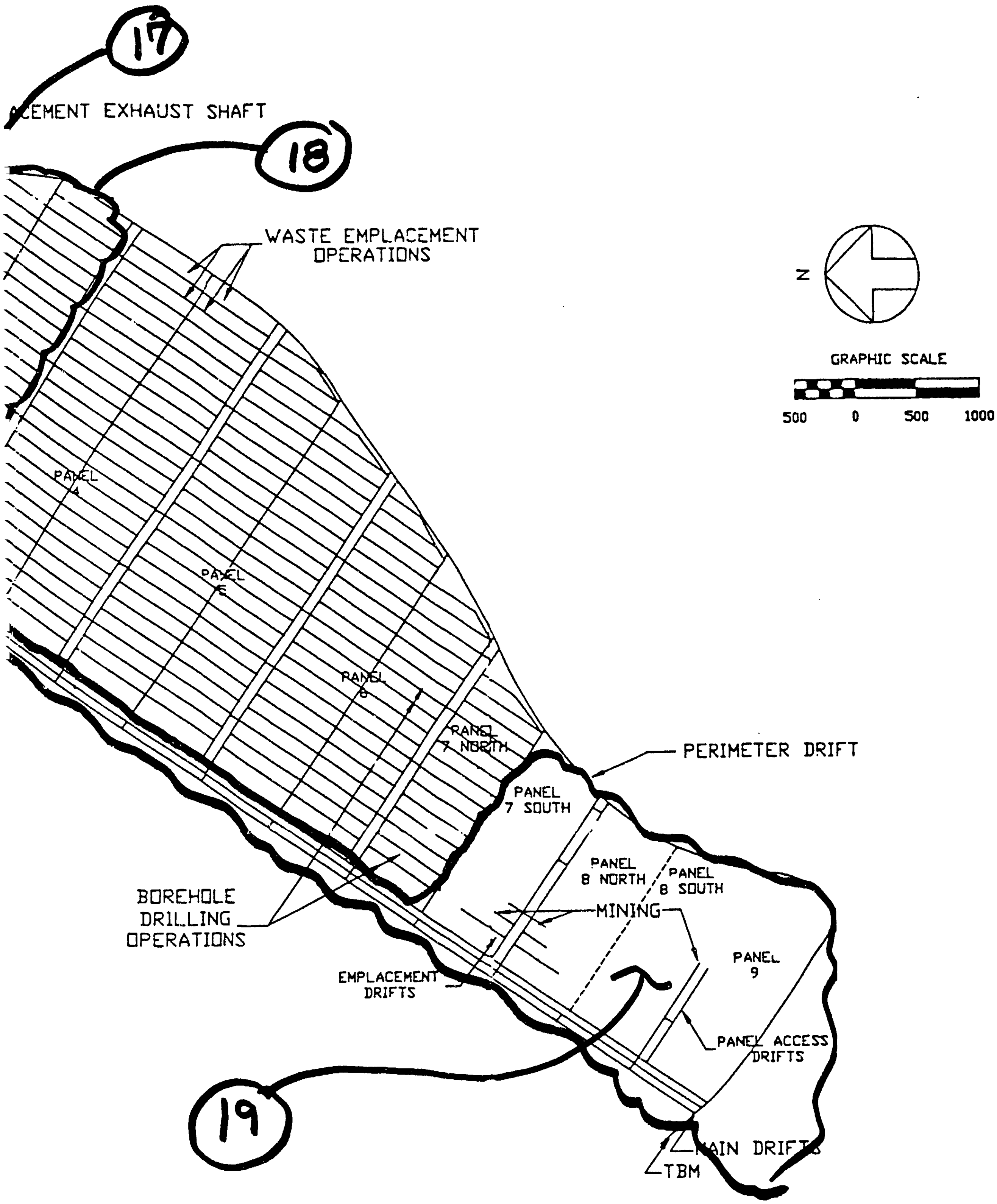


Figure 3-4. Compartments of the Underground Facilities

Repository Item (see Note a)	Item Number	Compartment Numb (see Note b)
Waste Package	1.2.2	-
Waste Form	-	1, 4-8, 10-15, 17,
Container	-	4, 7, 8, 12, 14-15, 1
Emplacement Borehole	-	18
Liner & Mechanical Appurtenances	-	18
Repository	1.2.4	-
Seals	1.2.4.2.3	17
Facilities	1.2.4.3	-
Site Preparation	1.2.4.3.1	20
Communications System	-	20
Drainage Control System	-	20
Fencing	-	20
Landscaping	-	20
Railroad	-	20
Roads	-	20
Utilities	-	20
Surface Facilities	1.2.4.3.2	-
Waste Handling Facilities	-	1-16
Balance of Plant	-	20
Exhaust Shaft Filter Building	-	17
Shafts and Ramps	1.2.4.3.3	-
Emplacement Exhaust Shaft	-	17
Exploratory Shafts	-	17
Men-and-Materials Shaft	-	19
Tuff Ramp	-	19
Waste Ramp	-	17
Underground Excavations	1.2.4.3.4	17-19
Underground Service Systems	1.2.4.3.5	17-19
Exploratory Shaft Facility	1.2.6.0	Note c
ESF Site	1.2.6.1	Note c
Main Pad	1.2.6.1.1	Note c
Auxiliary Pad	1.2.6.1.2	Note c
Access Roads	1.2.6.1.3	Note c
Site Drainage	1.2.6.1.4	Note c
Surface Utilities	1.2.6.2	Note c
Power Systems	1.2.6.2.1	Note c
Water Systems	1.2.6.2.2	Note c
Sewage Systems	1.2.6.2.3	Note c
Communication System	1.2.6.2.4	Note c
Mine Wastewater System	1.2.6.2.5	Note c

Notes:

a. Identification and numbering of repository items subject to the quality level as

b. Repository compartments and numbers are described in Table 3-1.

c. All ESF items will be removed prior to repository operations except undergrou

TABLE 3-2

E COMPARTMENTS IN WHICH THEY ARE LOCATED

r	Repository Item (see Note a)	Item Number	Compartment Number (see Note b)
	Surface Utilities, Continued		
18	Compressed Air System	1.2.6.2.6	Note c
7-18	Surface Facilities	1.2.6.3	Note c
	Ventilation System	1.2.6.3.1	Note c
	Test Support Facilities	1.2.6.3.2	Note c
	Sites for Temporary Facilities	1.2.6.3.3	Note c
	Parking Areas	1.2.6.3.4	Note c
	Material Storage Facilities	1.2.6.3.5	Note c
	Shop	1.2.6.3.6	Note c
	Warehouse	1.2.6.3.7	Note c
	Temporary Structures	1.2.6.3.8	Note c
	Communications/Data Building	1.2.6.3.9	Note c
	First Shaft	1.2.6.4	-
	Collar	1.2.6.4.1	17
	Lining	1.2.6.4.2	17
	Stations	1.2.6.4.3	Note c
	Furnishings	1.2.6.4.4	Note c
	Hoist System	1.2.6.4.5	Note c
	Sump	1.2.6.4.6	Note c
	Second Shaft	1.2.6.5	-
	Collar	1.2.6.5.1	17
	Lining	1.2.6.5.2	17
	Stations	1.2.6.5.3	Note c
	Furnishings	1.2.6.5.4	Note c
	Hoist System	1.2.6.5.5	Note c
	Sump	1.2.6.5.6	Note c
	Underground Excavations	1.2.6.6	-
	Operations Support Areas	1.2.6.6.1	19
	Test Areas	1.2.6.6.2	17-19
	Underground Support Systems	1.2.6.7	Note c
	Power Distribution System	1.2.6.7.1	Note c
	Communications System	1.2.6.7.2	Note c
	Lighting System	1.2.6.7.3	Note c
	Ventilation Distribution System	1.2.6.7.4	Note c
	Water Distribution System	1.2.6.7.5	Note c
	Mine Wastewater Collection System	1.2.6.7.6	Note c
	Compressed Air Distribution Systems	1.2.6.7.7	Note c
	Fire Protection System	1.2.6.7.8	Note c
	Muck Handling System	1.2.6.7.9	Note c
	Sanitary Facilities	1.2.6.7.10	Note c
	Monitoring and Warning System	1.2.6.7.11	Note c
	Underground Tests	1.2.6.8	Note c
	Integrated Data System (IDS)	1.2.6.8.1	Note c

signment process are taken from DOE (1989a).

nd openings (shafts and excavations), shaft liners, and ground support (DOE, 1987*a*).

A list of credible initiating events is shown in Table 3-3. The list was compiled on the basis of surveys of literature on safety analyses (NRC, 1983b; Elder, 1986; Brynda, 1981) and reviews of previous assessments for the repository (SNL, 1987), and was developed on a compartment-by-compartment basis.

Certain types of initiating events are not credible and, therefore, were screened from the list. For example, tsunami and high tides are not included because the site is far from the ocean. Meteorite impacts are not credible, and volcanic activity is not expected during operations at the Yucca Mountain site (DOE, 1986). Chemical effects from nearby industrial accidents do not require evaluation (for nuclear power plants licensed by the NRC) if there are no major storage areas or shipments of hazardous chemicals within 5 mi of the facility (NRC, 1974a); the Yucca Mountain site satisfies this criterion, so chemical effects are not credible. Water intrusion due to flooding into repository facilities is precluded by locating the waste-handling areas above the probable maximum flood level.

3.3 Event Trees

In this subsection, event trees are developed for each compartment to depict the relevant combinations of failures of items that could result from various initiating events. The Candidate List of Items and Activities Subject to the Quality Level Assignment Process (DOE, 1989a) includes general facilities without identifying individual structures, systems, and components in each facility. The event trees are developed to include a more detailed breakdown of individual items that comprise the facilities listed in DOE (1989a); however, only items whose failure could affect radiological releases are included in the event trees.

Although there are a large number of initiating events applicable to each compartment, the relevant combinations of failures of

TABLE 3-3

CREDIBLE INITIATING EVENTS

External Events
Earthquakes (including effects of underground weapons testing)
Flooding
Tornadoes and extreme winds
Aircraft crashes
Loss of offsite electrical power
Offsite range fire
Internal Events
Equipment failures
Operator errors
Fires
Inadvertent detonation of explosives used for mining

Note: Internal events, such as equipment failures, operator errors, and fires, include events such as the following:

Vehicle collisions
Crane load drops
Loss of electrical power
Improper procedures or procedural non-compliance
Other accidents (e.g., spills)

items in the compartment are basically the same for all types of initiating events. Therefore, only one event tree is needed for each compartment, and the same event tree can be used for each type of initiating event.

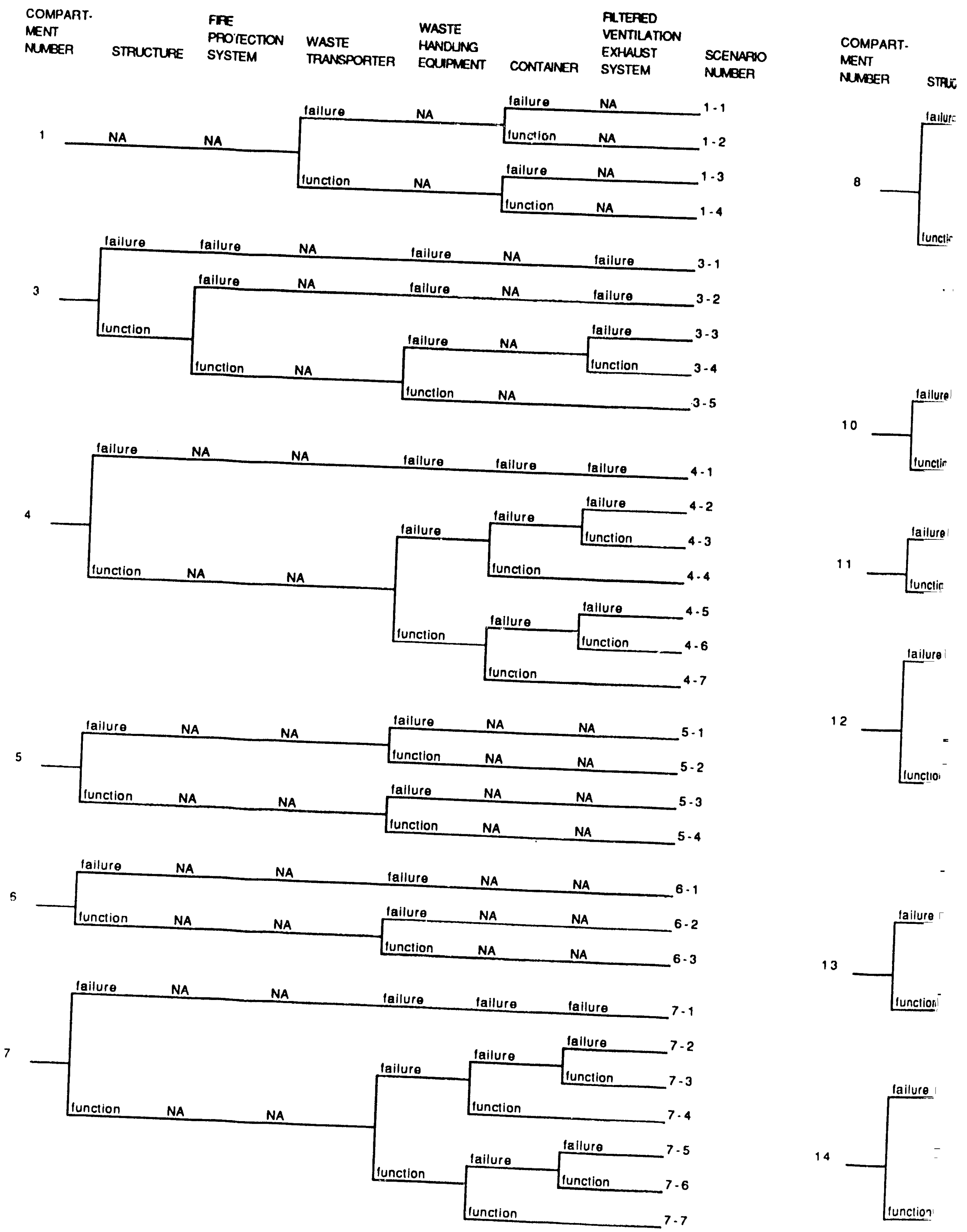
Event trees for each compartment are shown in Figure 3-5 and are described below. Event trees are not developed for compartments that contain insignificant radioactivity, as described below.

3.3.1 Compartment 1 - Cask Transportation Areas

In the cask transportation areas, the items whose failure could affect radiological releases are the underground waste transporter and the container. The event tree depicts all combinations of failures of these items.

Shipping casks, which are also in this compartment, are not included in the event tree because they are not a repository item. Th event tree assumes that shipping casks will function (i.e., provide confinement and protect the waste) during accidents when the casks are bolted and sealed (see Subsection 2.4). Failures of site-generated waste casks and vehicles are evaluated as part of the waste treatment building, WHB cask receiving and shipping air lock, and WHB support facilities compartments.

The functions of the waste transporter include radiation shielding, protection of the waste from damage due to impacts from external objects, and confinement of radioactive material. Failure of this item could involve loss of any or all of these functions. The container confines the radioactivity associated with spent fuel or HLW, and failure of the container could involve loss of confinement and potential release of radioactivity from the waste.



Note: NA means "not applicable" because the item is not in the associated compartment.

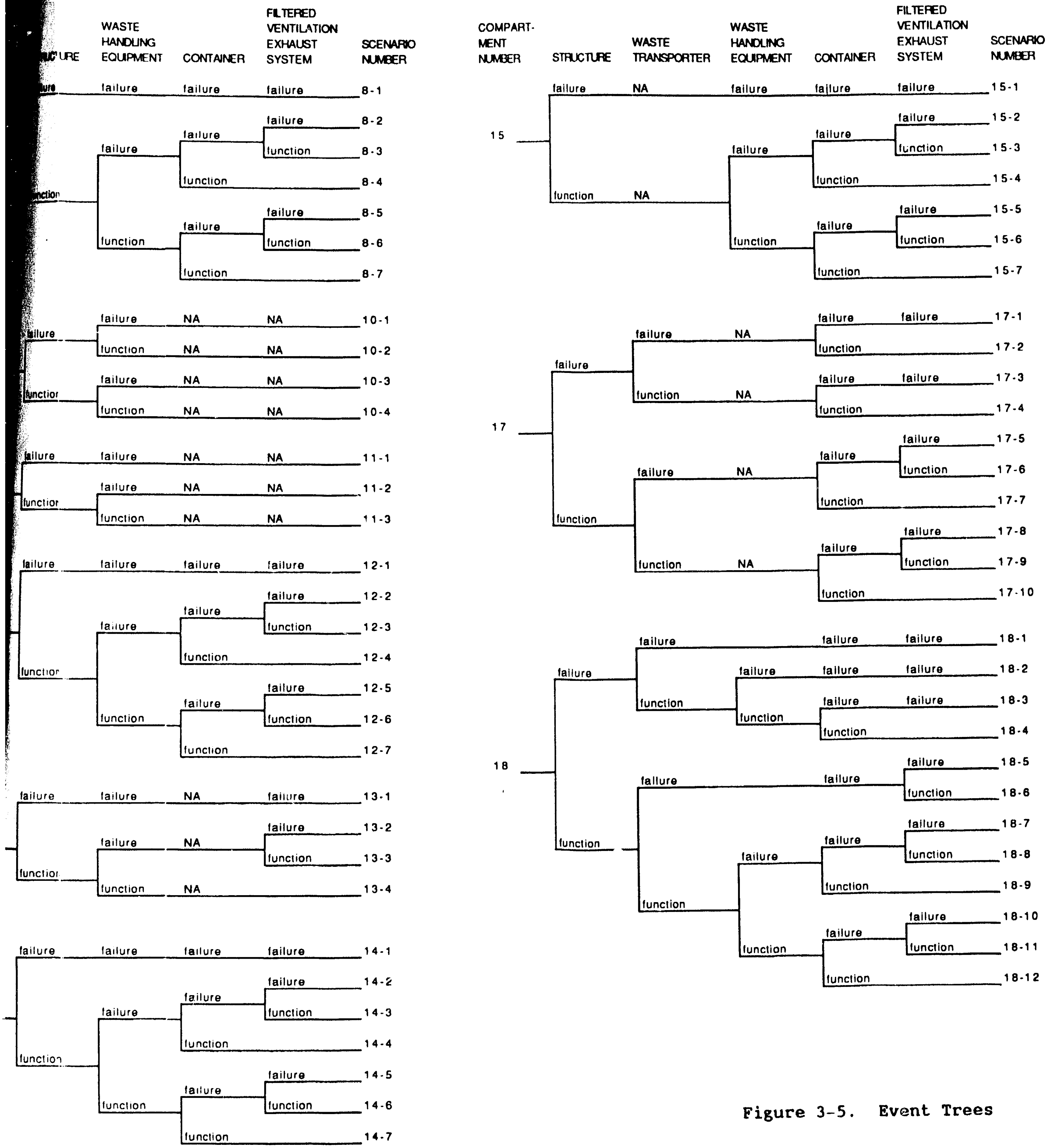


Figure 3-5. Event Trees

3.3.2 Compartment 2 - Decontamination Building

The decontamination building does not handle or store highly radioactive materials, and the maximum inventory of radioactivity is less than 5 Ci. The radioactivity in the decontamination building is predominantly Co-60 from crud (activated corrosion products) from spent fuel. On the basis of calculations in Section 4.0 and the limited quantity of radioactivity in this compartment, no combination of accidental failures could result in doses exceeding 0.5 rem at the nearest boundary of the unrestricted area. Therefore, no event tree is developed for this compartment.

3.3.3 Compartment 3 - Waste Treatment Building

The items in the waste treatment building whose failure could affect radiological releases are the building structure, fire protection system, waste-handling equipment, and filtered ventilation exhaust system. The fire protection system includes the sprinkler system, sensors and controls that actuate the system, water supply system, and all other items needed for the fire protection function. The waste-handling equipment includes storage tanks, process equipment, transfer casks, cranes, and other associated items. The filtered ventilation exhaust systems includes the exhaust ducting, filters, fans, electrical power supply, and any other items necessary to ensure that exhaust air is filtered prior to discharge. The event tree for this compartment depicts the various combinations of failures of these items.

The function of the building structure is to physically support the systems and equipment in the facility and to prevent structural collapse. Failure of the structure could therefore result in collapse along with the loss of functions of other items. If the fire protection system fails, fire could spread throughout the building, igniting combustible materials (including the site-generated waste) and causing other items to fail. Failure of waste-handling equipment could involve impacts resulting from drops or collisions that result in releases of radioactive materials. If the filtered ventilation exhaust system fails, unfiltered airborne radioactivity could escape to the atmosphere.

3.3.4 Compartment 4 - Performance Confirmation Building

In the performance confirmation building, the items whose failure could affect radiological releases are the hot cell structure, waste-handling equipment, container, and filtered ventilation exhaust system. The waste-handling equipment includes the overhead bridge crane in the hot cell, transfer carts, container cutting machine, electromechanical manipulators, and other equipment that handles containers, spent fuel, and HLW canisters. The hot cell structure includes the shield plugs, shield windows, transfer drawers, and other items that are part of the hot cell confinement barrier.

The hot cell structure supports the shielding and other items, and confines any accidental releases of radioactivity in the hot cell. Failure of the structure could result in collapse along with the loss of functions of other items. The event tree for this compartment shows relevant combinations of failures of the structure and other items. Because there are no significant combustible materials in the hot cells containing spent fuel and HLW, this study assumes that failures of fire protection systems (if any) would not affect releases of radioactive materials from these areas.

3.3.5 Compartment 5 - WHB-1 Cask Receiving and Shipping Air Lock

This compartment includes an air lock structure with an unfiltered ventilation exhaust system. The event tree for this compartment depicts all combinations of failures of the structure and any waste-handling equipment therein. The waste-handling equipment includes transfer casks for site-generated waste. The event tree assumes that the shipping casks will function during accidents (see Subsection 3.3.1).

3.3.6 Compartment 6 - WHB-1 Cask Receiving and Shipping Bay

In the WHB-1 cask receiving and shipping bay, the items whose failure could affect radiological releases are the structure and the waste-handling equipment. The waste-handling equipment includes the bridge crane, the cask cavity sampling and venting system, and the cask transfer car. Failure of the structure could result in collapse on a cask with an unbolted lid along with failures of other waste-handlng equipment. Failure of the waste-handling equipment could involve a crane dropping the cask, a crane falling on a cask (with unbolted lid), a cask falling from the cask transfer car, or release of airborne radioactivity from the cask sampling and venting system. Because this compartment does not have a filtered ventilation exhaust system (SNL, 1987), any radioactivity released into this compartment could escape to the atmosphere unfiltered.

3.3.7 Compartment 7 - WHB-1 Unloading Hot Cell

This compartment includes structures, systems, and components that are similar to those of the performance confirmation building, and the event trees are the same for these the compartments. The waste-handling equipment in this compartment includes the overhead bridge cranes in the hot cell, waste lifting fixtures, waste packaging equipment, and waste transfer/storage carts.

3.3.8 Compartment 8 - WHB-1 Surface Storage Vault

The WHB-1 surface storage vault stores waste containers in sleeves in a concrete-shielded structure. The items whose failure could affect radiological releases are the structure, the waste-handling equipment, the container, and the filtered ventilation exhaust system for the vault. The waste-handling equipment includes the container transfer machine (CTM) and interfacing shield valves. The ventilation exhaust system includes the exhaust ducting, filters, fans, electrical power supply, radiological monitoring system that controls the filtration mode, and any other items necessary to ensure that airborne radioactivity is filtered prior to discharge. Failures of the underground waste transporter are evaluated as part of the cask transportation area and underground compartments.

If the structure fails, it could collapse onto stored containers and cause the failures of other items in the building. Failure of the waste-handling equipment could involve a CTM dropping a container or other damage to the container and waste. Failure of the filtered ventilation exhaust system could result in an unfiltered release of airborne radioactivity to the atmosphere if the container and waste were damaged. Relevant combinations of these failures are depicted in the event tree shown in Figure 3-5.

3.3.9 Compartment 9 - WHB-1 Support Facilities

Only relatively small amounts of radioactivity associated with site-generated waste are handled or stored in this compartment. The maximum inventory of radioactivity is less than 5 Ci, which is predominantly Co-60 from crud. On the basis of calculations in Section 4.0 and the limited quantity of radioactivity in this compartment, no combination of accidental failures could result in doses exceeding 0.5 rem at the nearest boundary of the unrestricted area. Therefore, no event tree is developed for this compartment.

3.3.10 Compartment 10 - WHB-2 Cask Receiving and Shipping Air Lock

This compartment includes structures, systems, and components that are similar to those in the WHB-1 cask receiving and shipping air lock, and the event trees are the same for these two compartments.

3.3.11 Compartment 11 - WHB-2 Cask Receiving and Shipping Bay

This compartment includes items that are similar to those in the WHB-1 cask receiving and shipping bay, and the event trees are the same for these two compartments.

3.3.12 Compartment 12 - WHB-2 Unloading Hot Cell

This compartment includes structures, systems, and components that are similar to those in the WHB-1 unloading hot cell, and the event trees are the same for these two compartments.

3.3.13 Compartment 13 - WHB-2 Consolidation Hot Cell

In the WHB-2 consolidation hot cell, failure of the structure, waste-handling equipment, or filtered ventilation exhaust system can affect radiological releases. The event tree for this compartment depicts relevant combinations of failures of the structures, systems, and components. The waste-handling equipment includes the overhead bridge crane, manipulators, fuel consolidation equipment, and other equipment that handles the spent fuel.

3.3.14 Compartment 14 - WHB-2 Packaging Hot Cell

The packaging hot cell in WHB-2 includes a shielded concrete structure, waste-handling equipment, containers of spent fuel and HLW, and the filtered ventilation exhaust system for the hot cell. Failure of any of these items could affect radiological releases. Relevant combinations of failures of these items are shown in the event tree for this compartment. The waste-handling equipment includes the overhead bridge crane, manipulators, container welding station, waste transfer carts, and other equipment that handles waste containers.

3.3.15 Compartment 15 - WHB-2 Surface Storage Vault

This compartment includes structures, systems, and components that are similar to those in the WHB-1 surface storage vault, and the event trees are the same for these two compartments.

3.3.16 Compartment 16 - WHB-2 Support Facilities

Only relatively small amounts of radioactivity associated with site-generated waste (i.e., less than 5 Ci of Co-60) are handled or stored in this compartment, and no combination of accidental failures could result in excessive doses. Therefore, no event tree is developed for this compartment.

3.3.17 Compartment 17 - Underground Waste Emplacement Access Drifts

The structures in this compartment include the rock surrounding the drifts, ramps, and shafts, and any associated support structures (e.g., liners, rock bolts). For this event tree, the doors between the underground development facilities (compartment 19) and the waste emplacement area are also considered to be part of the structure. The waste transporter, container, and filtered ventilation exhaust system (in the emplacement exhaust filtration facility on the surface) are also included in this compartment. Relevant combinations of failures of these items are depicted in the event tree for this compartment. The event tree reflects the consideration that failure of the drift structure could prevent the proper function of the filtered ventilation exhaust system due to blockage of drifts and associated exhaust air pathways (i.e., for scenarios in which the drift structure fails, the filtered ventilation exhaust system is also assumed to fail).

3.3.18 Compartment 18 - Underground Waste Emplacement Drifts

This compartment includes the same items as the waste emplacement access drifts plus waste-handling equipment, which includes the shield valve, borehole with liner, and shield plug. Relevant combinations of failures of these items are depicted in the event tree for this compartment.

3.3.19 Compartment 19 - Underground Development Facilities

No radioactive waste is handled or stored in this compartment, and there are no items whose failures could affect radiological releases. Therefore, no event tree is developed for this compartment.

3.3.20 Compartment 20 - Other Repository Support Facilities

No radioactive waste is handled or stored in this compartment, and there are no items whose failure could affect radiological releases. Therefore, no event tree is developed for this compartment.

4.0 EVENT TREE ANALYSIS

4.1 Radiation Doses

The dose consequences for each branch of the event trees should be calculated (see Step 6 of Table 2-1). This subsection describes the method of calculation and presents the analysis of the radiation doses resulting from each postulated accident scenario. The assessment of accident doses is performed in accordance with DOE Procedure AP-6.10Q (Appendix A). Note that this work was completed in April, 1990 and AP-6.10Q was later superseded.

4.1.1 Methodology

In certain compartments, accidents could damage the waste (e.g., by mechanical impact), resulting in the release of airborne radioactive materials. Once released from the waste, the radioactivity would be directed through various confinement barriers (e.g., filters) prior to release to the atmosphere. This radioactivity would be transported and dispersed through the atmosphere and could result in radiation doses to nearby individuals.

A graphical model of the process described above is presented in Figure 4-1. The model provides the basis for evaluating the accident doses.

The first step of the dose analysis is to determine the characteristics and types of radioactive materials that may be damaged during accidents. The radionuclide inventory of the waste is subsequently evaluated.

Next, the extent of damage to the waste is evaluated to estimate the fraction of the radioactivity that could be accidentally released into the compartment. Any reduction or removal mechanisms prior to release to the atmosphere are identified and quantified. Then, the transport of radioactivity through the atmosphere is

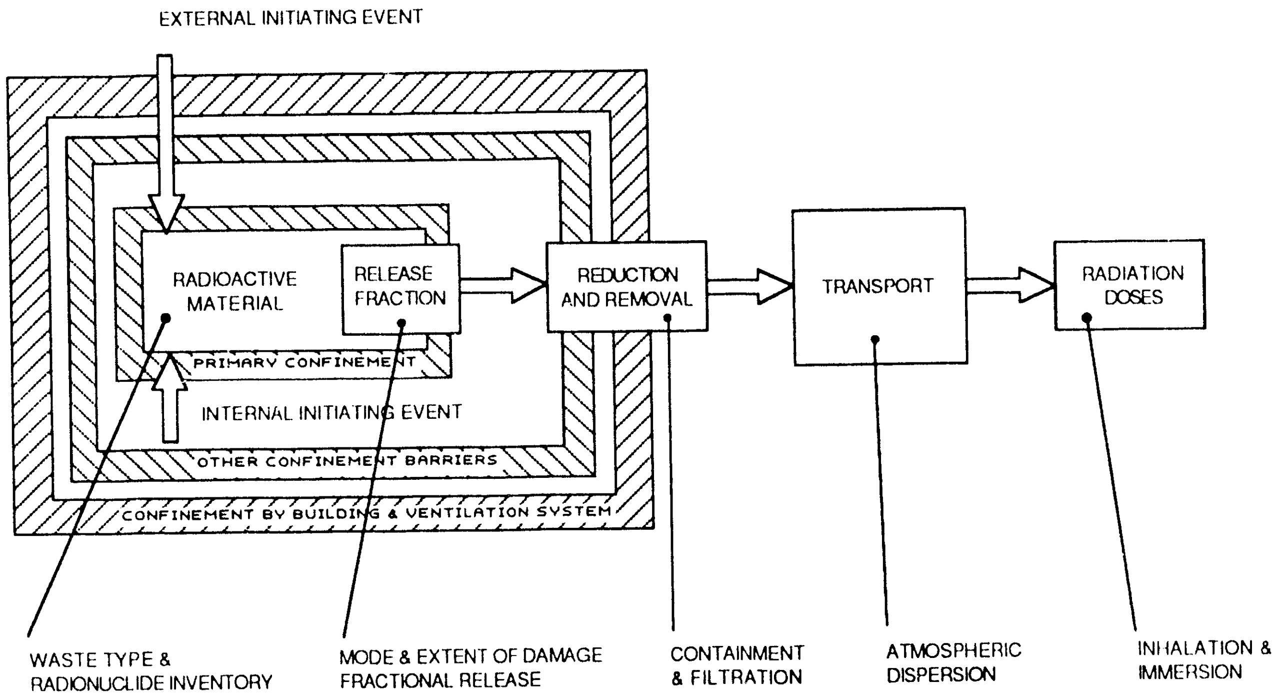


Figure 4-1. Graphical Model of Accidental Radiation Exposure Pathway

evaluated and the radiological consequences to individuals exposed to the radioactivity are calculated.

Individuals could receive accidental radiation doses through various exposure pathways. The exposure pathways are inhalation, immersion, ingestion, and direct shine from an external radiation source. Of these four pathways, only the inhalation pathway is significant and hence is the only one calculated in the accident dose analysis.

Appendix A indicates that the accident analysis should consider radiological consequences to the whole body and maximally exposed organs. The dose calculations in this study consider the major organs addressed in NRC (1977) and NRC (1981).

Based on the information above, the accident radiation dose is given by the following equation:

$$D_o = n \left\{ \sum_i f_i A_{i,o} DCF_{i,o} R_i \right\} (\chi/Q)(BR)$$

D_o = inhalation dose to organ o for an individual (rem)

n = number of damaged waste units (e.g., fuel assemblies)

f_i = fraction of radioistope i released from the damaged waste as respirable airborne radioactivity (dimensionless)

A_i = inventory of isotope i (Ci/waste unit)

$DCF_{i,o}$ = inhalation dose conversion factor for organ o and isotope i (rem/Ci)

R_i = fraction of radioisotope i released from damaged waste that could escape to the atmosphere (dimensionless)

χ/Q = atmospheric dispersion factor (sec/m^3)

BR = breathing rate (m^3/sec)

Application of the above calculation model to the accident analysis is described below.

4.1.1.1 Waste Types and Radionuclide Inventories

Three types of radioactive wastes are handled at the repository: spent fuel, HLW, and site-generated waste. Each of these waste types is described below.

Spent Fuel

Spent fuel is received at the repository in the form of intact fuel assemblies. The fuel assemblies are either placed directly in containers or consolidated and subsequently placed in containers. The repository handles both pressurized water reactor (PWR) fuel and boiling water reactor (BWR) fuel.

The radioactive materials available for release from spent fuel are the fuel pellet matrix, fission gases in the fuel rod gap and plenum, and the activated corrosion products (crud) on the exterior surface of the fuel rods. The fuel pellet matrix is assumed to consist of a homogeneous mixture of the solid radioactive fission products generated in the UO_2 fuel pellet from reactor operation. Fission gases are generated in the fuel pellet matrix during reactor operation, where a fraction of the gas migrates along the UO_2 grain boundaries to the fuel rod gap and plenum. The only significant fission gases in 5-yr-old spent fuel are H-3, C-14, and Kr-85. An additional source of radioactive material associated with spent fuel is the crud on the exterior surface of the fuel assembly. The radioactivity of crud consists primarily of Co-60 (NRC, 1984). Compared to the particulate radioisotopes in the spent fuel material, the fission gases and crud are insignificant contributors to the accident doses and can be neglected for the worst-case spent fuel accidents considered in this study.

The bounding source term (maximum radionuclide inventory) for spent fuel is selected for this study to determine a maximum dose. As stated in Appendix A, the radionuclide inventory of spent fuel is obtained from Roddy (1986). On the basis of data presented in Roddy (1986), the fuel type that results in the maximum radionuclide inventory corresponds to PWR fuel having a burnup of 60 GWd/MTU and 5 yr out of the reactor. Given a maximum initial uranium loading for a PWR fuel assembly of 0.469 MTU (DOE, 1987b) and the data in Roddy (1986) for the reference fuel type, the radionuclide inventory of spent fuel is calculated, and the results are presented in Table 4-1.

HLW

HLW is received at the repository in canisters. The HLW canisters are overpacked in containers at the repository for emplacement underground. Assessment of the radiological consequences of accidents involving HLW indicate that the doses are much less than those involving spent fuel (primarily due to the lower inventory of transuranics in HLW compared to spent fuel). Because spent fuel is handled in the same compartments as HLW, only the radiological consequences of spent fuel accidents are calculated in this study.

Site-Generated Waste

Various types of solid and liquid wastes are generated on site during repository operations. This waste is characterized in SNL (1987). The radioactivity of the site-generated wastes primarily consists of Co-60 (SNL, 1987).

TABLE 4-1

RADIONUCLIDE INVENTORY OF PWR SPENT FUEL ASSEMBLY AND ASSOCIATED DOSE CONVERSION FACTORS (DCFs)

Radionuclide (Note a)	Inventory (Ci)	Inhalation DCF (Note b) (Rem/Ci)		
		Whole Body	Bone Surfaces	Lung
H-3	4.40E+02	1.25E+02	9.85E+01	1.25E+02
C-14	1.14E+00	1.41E+01	5.08E+01	6.18E+00
Fe-55	8.44E+02	6.29E+02	7.40E+02	2.51E+04
Co-60	2.65E+03	8.20E+04	5.08E+04	1.30E+06
Ni-63	4.78E+02	1.81E+03	--	2.23E+04
Kr-85	4.83E+03	--	--	2.41E+00
Sr-90	4.83E+04	2.40E+05	2.20E+06	8.50E+06
Y-90	4.83E+04	8.90E+02	9.06E+02	3.93E+04
Ru-106	1.15E+04	6.18E+04	1.00E+04	3.80E+06
Rh-106	1.15E+04	3.44E-01	1.14E-02	2.32E+01
Sb-125	3.10E+03	1.58E+03	--	2.18E+05
Te-125m	7.55E+02	5.84E+01	--	3.92E+04
Cs-134	3.20E+04	4.55E+04	5.89E+04	3.38E+04
Cs-137	7.60E+04	3.26E+04	5.31E+04	1.62E+04
Ba-137m	7.18E+04	3.21E-01	1.74E-01	7.09E+00
Ce-144	5.72E+03	1.70E+05	6.10E+05	2.90E+06
Pr-144	5.72E+03	9.75E+00	6.91E+00	6.67E+02
Pm-147	1.53E+04	2.46E+04	9.07E+04	2.90E+05
Sm-151	2.41E+02	3.55E+03	--	4.45E+04
Eu-154	7.93E+03	6.48E+04	--	5.84E+05
Eu-155	3.82E+03	9.21E+03	--	9.46E+04
Pu-238	3.95E+03	1.11E+08	8.40E+09	6.08E+08
Pu-239	1.72E+02	1.24E+08	1.04E+10	5.80E+08
Pu-240	3.21E+02	1.24E+08	1.04E+10	5.79E+08
Pu-241	7.27E+04	2.50E+06	1.94E+08	1.10E+06
Am-241	7.83E+02	1.28E+08	9.73E+09	6.15E+08
Am-242m	8.11E+00	1.26E+08	8.18E+09	2.21E+08
Am-243	3.38E+01	1.28E+08	1.03E+10	5.95E+08
Cm-243	3.89E+01	8.70E+07	5.78E+09	6.27E+08
Cm-244	6.26E+03	6.80E+07	4.56E+09	6.07E+08

Notes: a. The list includes radionuclides that each contribute at least 0.1% of the total radionuclide inventory or individual dose.

b. For accidental releases of radionuclides from the fuel pellet, the maximally exposed organ is the bone surfaces; for releases of Co-60 from site-generated waste, the maximally exposed organ is the lung.

Most of the site-generated waste exists in the waste treatment building. According to the material block flow diagrams presented in SNL (1987), there may be a maximum of approximately 57 Ci of solid waste and 112 Ci of liquid waste in the waste treatment building. Waste-handling equipment in the waste treatment building contains in the range of 16 to 80 Ci of waste (solid or liquid). Other areas outside the waste treatment building can accumulate up to 5 Ci of site-generated waste before the waste is transferred to the waste treatment building.

4.1.1.2 Release Mechanism and Radionuclide Release Fractions

The fuel rod cladding could breach if spent fuel assemblies are subjected to a severe mechanical impact. This would allow a path for the release of airborne radioactive materials, including respirable-size fuel particles (diameters less than 10 microns) Other waste that experiences a severe mechanical impact or that burns could also release airborne radioactive materials. The fraction of radioactive material that could be released from waste subjected to these mechanisms is described below.

This study considers two sources of airborne fuel pellet material that could be released from spent fuel. The first is the fraction of the fuel that exists as respirable particles in the fuel rod when spent fuel is received at the repository. The second is that generated from accidental impact.

NRC (1988a) indicates that respirable particles of spent fuel pellet material are generated during reactor operation and are present in the gap in spent fuel rods. The fraction of the total spent fuel pellet material that could exist in this form is reported to be up to $1 * 10^{-4}$ (NRC, 1988a).

Spent fuel accidents involving severe mechanical impacts may cause fuel pellet fracture. Studies of the fracture mechanism by Jardine (1982) and Mecham (1983) indicate that the fraction of simulated HLW glass specimen fractured into respirable particles is linearly proportional to the impact energy density:

$$PF = (2 * 10^{-4})\ (E/V)$$

where PF = fraction of waste specimen fractured into respirable particles (dimensionless)

E = impact energy absorbed by specimen (J)

V = volume of specimen (cm^3)

This linear relation is assumed to be valid for spent fuel rods containing many irradiated fuel pellets.

The energy associated with mechanical impact on spent fuel assemblies during postulated accidents is estimated by assuming a fuel drop height. Based on dimensional data in DOE (1987b) for the referenced spent fuel assembly, the respirable-size fraction can be expressed as follows:

$$PF = (2 * 10^{-5})h$$

where h = fuel assembly drop height (m)

Therefore, the fraction of airborne fuel pellet material released from a spent fuel assembly as a result of mechanical impact, f_{fp}, is calculated using the following equation:

$$f_{fp} = (1 * 10^{-4}) + (2 * 10^{-5})h$$

The fission gases of H-3, C-14, and Kr-85 are generated in the fuel pellet matrix during reactor operation and migrate to the fuel rod gap. Only a fraction of the total inventory of these gases generated during reactor operation migrate to the rod gap and are available for release. The fraction of H-3 and Kr-85 assumed to be present in the rod gap and plenum are 0.1 and 0.3, respectively (NRC, 1972). It is assumed in this study that the fraction of C-14 present in the rod gap and plenum is also 0.1. If it is assumed that 100% of the gas in the rod gap and plenum is released as a result of a severe impact or other accident, the resulting gaseous contribution to radiation doses at the unrestricted area boundary is negligible compared to the doses due to spent fuel particles.

The release of radioactive materials from solid and liquid site-generated waste may result from both combustion and large mechanical impact. ANSI (1981) recommends a value of 0.01 for the fraction of radioactive material released from either volatile or nonvolatile solids from combustion. This study conservatively assumes that this fraction is applicable to releases of radioactivity from liquid site-generated waste. Additionally, the value of 0.01 is conservatively assumed for the fraction of waste (solid and liquid) released or dispersed as airborne activity as a result of severe mechanical impacts.

4.1.1.4 Reduction and Removal Mechanisms

Subsequent to release from the waste form, airborne radioactivity may be confined by barriers or may be directed through filters prior to release to the atmosphere. In the calculation model, this reduction and removal mechanism is taken into account by the parameter R. When functioning, the filtered ventilation exhaust system reduces the amount of airborne radioactive materials released to the atmosphere. Also, the waste containers and other containment features provide 100% confinement when functioning ($R = 0$) or no confinement upon failure ($R = 1$).

The filtered ventilation exhaust systems associated with repository containment structures consist of two HEPA filters in series for removal of airborne particulate materials. ANSI (1981) recommends that the efficiency of the first stage of the system is 99.9% and the second stage is 99.0%. Therefore, the filtered ventilation exhaust systems at the repository are effective at reducing the amount of solid airborne particles released from waste by a factor of $1 * 10^{-5}$ prior to release to the atmosphere ($R = 1 * 10^{-5}$). Gaseous radionuclides (e.g., H-3, C-14, and Kr-85) would pass through the particulate filters ($R = 1$).

This study conservatively neglects deposition of airborne material within any of the containment barriers.

4.1.1.5 Transport Mechanisms

The primary mode of transport of radionuclides from accident releases is the atmospheric pathway. Radionuclides that are released into the atmosphere will be diluted by atmospheric dispersion as they are transported to the point of exposure. An atmospheric dispersion factor, x/Q, is used to estimate concentrations of airborne radioactivity downwind from the release point. The atmospheric dispersion factor is calculated in accordance with the requirements in Appendix A.

NRC (1974b) indicates that the basic equation expressing the atmospheric diffusion, x/Q, for a ground-level release of radioactivity for accidents less than 8 hr in duration is:

$$\chi/Q = 1/\pi W u \sigma_y \sigma_z$$

where

W = building wake factor
u = wind speed (m/sec)
s_y = horizontal standard deviation of the plume (m)
s_z = vertical standard deviation of the plume (m)

The equation for a ground-level release is used because NRC (1974b) only allows credit for elevated releases if the point of release is more than two and one-half times the height of any adjacent structure (i.e., the release is not entrained in the building wake), which is not the case for the SCP-CDR configuration. The building wake factor W accounts for the additional dispersion due to the turbulent wake of nearby buildings.

Accidents addressed in this study are assumed to last less than 8 hr. NRC (1974b) states that for this time duration, a wind speed of 1 m/sec and Pasquill Type F atmospheric condition should be used. Values of σ_y and σ_z for Pasquill Type F conditions are presented in NRC (1982).

The study assumes that the maximum exposed individual is at the boundary of the unrestricted area of the repository directly downwind from the radioactive release. This study assumes the boundary to be 100 m from the point of release for dose calculation purposes. The building wake factor W = 1 unless the radioactivity is released from repository structures, in which case W = 3 (NRC, 1972).

Based on the above, the atmospheric dispersion factor x/Q is calculated to be $3.18 * 10^{-2}$ sec/m^3, without building wake effects, and $1.06 * 10^{-2}$ sec/m^3, with building wake effects. These values overestimate the atmospheric dispersion factors measured experimentally under similar conditions at other sites (NRC 1982). When meteorological data specified by NRC (1982) become available for the Yucca Mountain site, more realistic atmospheric dispersion factors can be used (see Section 7.2).

4.1.1.6 Dose Conversion Factors

Appendix A states that the radiological consequences be evaluated for a 50-yr dose commitment to the whole body and maximally exposed organ. As previously stated, the primary exposure pathway is through the inhalation of airborne radioactivity. Thus, inhalation organ dose conversion factors (DCFs) are used in the dose calculations. The DCFs are obtained primarily from NRC (1981); DCFs not available in this document are obtained from NRC (1977).

Considering the radioactive material releases associated with the wastes handled at the repository, the maximum exposed organs are the bone surfaces for the release of spent fuel particles and the lung for the release of radioactivity from site-generated wastes (Co-60). Accidental doses to the whole body are much less than the corresponding doses to the maximally exposed organ.

4.1.2 Accident Dose Analysis

Radiation doses for each accident scenario of the event trees in Figure 3-5 are evaluated using the calculation model presented in Subsection 4.1.1. The doses for each of these scenarios are evaluated by identifying and quantifying the waste involved in the scenario, the fraction of material released from the waste, the reduction and removal fraction, and the atmospheric dispersion factor. These parameters and the resulting accident doses are quantified and listed in Table 4-2 for each accident scenario presented in Figure 3-5.

In Table 4-2, the number of spent fuel assemblies associated with the scenarios involving failure of waste-handling equipment in each compartment is evaluated based on the capacity of the equipment. The fraction of fuel released is estimated based on the height from which the fuel assemblies could fall as a result of failure of the equipment. The number of spent fuel assemblies and the extent of

Scenario Number (Note a)	n (Note b)	f (Note c)	R (Note d)	X/Q (sec/m^3) (Note e)	Dose (rem) (Note f)
1 - 1	6	1.7E-04	1	3.18E-02	>> 0.5
1 - 2	6	1.7E-04	0	3.18E-02	0
1 - 3	6	1.7E-04	0	3.18E-02	0
1 - 4	-	0	0	3.18E-02	0
3 - 1	169 Ci	1.0E-02	1	1.06E-02	8.1
3 - 2	169 Ci	1.0E-02	1	1.06E-02	8.1
3 - 3	16 to 80 Ci	1.0E-02	1	1.06E-02	0.8 to 3.8
3 - 4	16 to 80 Ci	1.0E-02	1E-05	1.06E-02	<< 0.5
3 - 5	-	0	0	1.06E-02	0
4 - 1	6	3.4E-04	1	1.06E-02	>> 0.5
4 - 2	6	2E-4 to 3.4E-4	1	1.06E-02	>> 0.5
4 - 3	6	2E-4 to 3.4E-4	1E-05	1.06E-02	3.9 to 6.7
4 - 4	6	2E-4 to 3.4E-4	0	1.06E-02	0
4 - 5	6	3.4E-04	1	1.06E-02	>> 0.5
4 - 6	6	3.4E-04	1E-05	1.06E-02	6.7
4 - 7	-	0	0	1.06E-02	0
5 - 1	16 Ci	0.01	1	1.06E-02	0.8
5 - 2	16 Ci	0.01	0	1.06E-02	0
5 - 3	16 Ci	0.01	1	1.06E-02	0.8
5 - 4	-	0	0	1.06E-02	0
6 - 1	14	3.4E-04	1	1.06E-02	>> 0.5
6 - 2	14	1E-4 to 3.4E-4	1	1.06E-02	>> 0.5
6 - 3	-	0	0	1.06E-02	0
7 - 1	12	3.4E-04	1	1.06E-02	>> 0.5
7 - 2	3 to 12	2.2E-4 to 3.4E-4	1	1.06E-02	>> 0.5
7 - 3	3 to 12	2.2E-4 to 3.4E-4	1E-05	1.06E-02	2.2 to 13.4
7 - 4	3 to 12	2.2E-4 to 3.4E-4	0	1.06E-02	0
7 - 5	3	3.4E-04	1	1.06E-02	>> 0.5
7 - 6	3	3.4E-04	1E-05	1.06E-02	3.4
7 - 7	-	0	0	1.06E-02	0
8 - 1	3	2.8E-04	1	1.06E-02	>> 0.5
8 - 2	3	2.8E-04	1	1.06E-02	>> 0.5
8 - 3	3	2.8E-04	1E-05	1.06E-02	2.8
8 - 4	3	2.8E-04	0	1.06E-02	0
8 - 5	3	2.8E-04	1	1.06E-02	>> 0.5
8 - 6	3	2.8E-04	1E-05	1.06E-02	2.8
8 - 7	-	0	0	1.06E-02	0
10 - 1	16 Ci	0.01	1	1.06E-02	0.8
10 - 2	16 Ci	0.01	0	1.06E-02	0
10 - 3	16 Ci	0.01	1	1.06E-02	0.8
10 - 4	-	0	0	1.06E-02	0
11 - 1	14	3.4E-04	1	1.06E-02	>> 0.5
11 - 2	14	1E-4 to 3.4E-4	1	1.06E-02	>> 0.5
11 - 3	-	0	0	1.06E-02	0

Notes

a. Numbers are assigned to each scenario in the event trees shown in Figure 3-5. The first di

b. n denotes the number of spent fuel assemblies that could be damaged in the accident scen suffering damage in the accident scenario.

c. f denotes the fraction of radioactivity released from the waste form.

d. R denotes the fraction of released radioactivity that could escape to the atmosphere, and i

e. X/Q denotes the atmospheric dispersion factor.

f. The reported doses are those that could be received by the maximally exposed organ (i.e., ei

TABLE 4-2

:ALCULATION PARAMETERS AND RESULTS

Scenario Number (Note a)	n (Note b)	f (Note c)	R (Note d)	X/Q (sec/m^3) (Note e)	Dose (rem) (Note f)
12 - 1	12	3.4E-04	1	1.06E-02	>> 0.5
12 - 2	3 to 12	2.2E-4 to 3.4E-4	1	1.06E-02	>> 0.5
12 - 3	3 to 12	2.2E-4 to 3.4E-4	1E-05	1.06E-02	2.2 to 13.4
12 - 4	3 to 12	2.2E-4 to 3.4E-4	0	1.06E-02	0
12 - 5	3	3.4E-04	1	1.06E-02	>> 0.5
12 - 6	3	3.4E-04	1E-05	1.06E-02	3.4
12 - 7	-	0	0	1.06E-02	0
13 - 1	10	3.4E-04	1	1.06E-02	>> 0.5
13 - 2	1 to 6	3.4E-04	1	1.06E-02	>> 0.5
13 - 3	1 to 6	3.4E-04	1E-05	1.06E-02	1.1 to 6.7
13 - 4	-	0	0	1.06E-02	0
14 - 1	24	3.4E-04	1	1.06E-02	>> 0.5
14 - 2	6 to 24	1.6E-4 to 3.4E-4	1	1.06E-02	>> 0.5
14 - 3	6 to 24	1.6E-4 to 3.4E-4	1E-05	1.06E-02	3.2 to 26.8
14 - 4	6 to 24	1.6E-4 to 3.4E-4	0	1.06E-02	0
14 - 5	6	1.6E-4 to 3.4E-4	1	1.06E-02	>> 0.5
14 - 6	6	1.6E-4 to 3.4E-4	1E-05	1.06E-02	3.2 to 6.7
14 - 7	-	0	0	1.06E-02	0
15 - 1	6	2.8E-04	1	1.06E-02	>> 0.5
15 - 2	6	2.8E-04	1	1.06E-02	>> 0.5
15 - 3	6	2.8E-04	1E-05	1.06E-02	5.5
15 - 4	6	2.8E-04	0	1.06E-02	0
15 - 5	6	2.8E-04	1	1.06E-02	>> 0.5
15 - 6	6	2.8E-04	1E-05	1.06E-02	5.5
15 - 7	-	0	0	1.06E-02	0
17 - 1	6	2.8E-04	1	3.18E-02	>> 0.5
17 - 2	6	2.8E-04	0	3.18E-02	0
17 - 3	6	2.8E-04	1	3.18E-02	>> 0.5
17 - 4	6	2.8E-04	0	3.18E-02	0
17 - 5	6	2.8E-04	1	3.18E-02	>> 0.5
17 - 6	6	2.8E-04	1E-05	3.18E-02	16.6
17 - 7	6	2.8E-04	0	3.18E-02	0
17 - 8	6	2.8E-04	1	3.18E-02	>> 0.5
17 - 9	6	2.8E-04	1E-05	3.18E-02	16.6
17 - 10	-	0	0	3.18E-02	0
18 - 1	6	2.2E-04	1	3.18E-02	>> 0.5
18 - 2	6	2.2E-04	1	3.18E-02	>> 0.5
18 - 3	6	2.2E-04	0	3.18E-02	0
18 - 4	6	2.2E-04	0	3.18E-02	0
18 - 5	6	2.2E-04	1	3.18E-02	>> 0.5
18 - 6	6	2.2E-04	1E-05	3.18E-02	13.0
18 - 7	6	2.2E-04	1	3.18E-02	>> 0.5
18 - 8	6	2.2E-04	1E-05	3.18E-02	13.0
18 - 9	6	2.2E-04	0	3.18E-02	0
18 - 10	6	2.2E-04	0	3.18E-02	0
18 - 11	6	2.2E-04	0	3.18E-02	0
18 - 12	-	0	0	3.18E-02	0

ıit(s) correspond to the compartment number and the second digit(s) correspond to a sequential number for the scenario
.rio, except that values given in units of Curies (Ci) indicate the amount of Co-60 contained in the site-generated waste

ıcludes a containment factor (0 or 1) and a filtration factor (1E-05), as appropriate.

her the bone surfaces or the lung, depending on the radioactive material released).

damage to the fuel for failures such as structural collapse are estimated to be representative of typical inventories in each area.

The quantity of radioactivity for accident scenarios associated with compartments handling site-generated waste is estimated in the same manner as spent fuel. The capacity of the waste-handling equipment is used as the basis for determining the quantity of radioactivity for scenarios involving failure of this equipment. The total inventory of waste is used for scenarios involving failures of the structure or fire protection system.

The fraction of radioactivity released to the atmosphere, as listed in Table 4-2, accounts for any containment functions or filtration functions indicated for each scenario in the event tree.

The calculated accident doses listed in Table 4-2 are used for identifying Q-scenarios and items important to safety, as discussed in the following sections of this report.

4.2 Scenario Probabilities

The credible initiating events described in Subsection 3.2 could impose severe loads, temperatures, or other adverse conditions on the repository structures, systems, and components. Because items that are not important to safety would be designed and constructed to normal industrial standards (Brynda, 1981; DOE, 1989b) and not necessarily designed and constructed to withstand severe credible events, these items could fail as a result of the severe events. To determine which repository items are important to safety, this assessment initially assumes that repository items would be designed and constructed to normal industrial standards (if certain items are then identified as important to safety, more stringent design criteria would be applied to those items).

Given the above assumption, all scenarios described in Subsection 3.3 (and shown in Figure 3-5) are credible. Section 6.0 of this report describes the recommended design requirements and operational limitations that would reduce the likelihood of failures of items identified as important to safety, so that many of the scenarios with potentially excessive consequences would not be credible.

5.0 IDENTIFICATION OF ITEMS IMPORTANT TO SAFETY

5.1 Identification of Q-Scenarios

Credible accident scenarios exceeding the dose criterion of 0.5 rem should be classified as Q-scenarios (see Step 8 of Table 2-1). As discussed in Subsection 4.2, all scenarios identified in Figure 3-5 and described in Subsection 3.3 are credible. Therefore, the scenarios shown in Figure 3-5 with potentially excessive doses are classified as Q-scenarios. These Q-scenarios are listed in Table 5-1.

Step 9 of Table 2-1 indicates that other scenarios not satisfying the above criteria (i.e., not classified as Q-scenarios in Step 8) shall be reclassified as Q-scenarios if other conditions exist (e.g., if the calculated dose and probability of the scenario are close to the values used to define Q-scenarios). None of these conditions have been identified for any other scenarios, and no other scenarios need to be reclassified as Q-scenarios.

5.2 Elimination of NQ-Scenarios

Scenarios that either are not credible or do not have consequences exceeding the dose criterion of 0.5 rem are not Q-scenarios (i.e., NQ-scenarios) (see Step 10 of Table 2-1). In accordance with DOE Procedure AP-6.10Q, (this work was completed in April, 1990 and AP-6.10Q was later superseded) all NQ-scenarios are eliminated from further consideration in identifying items important to safety. These NQ-scenarios are those scenarios from Figure 3-5 that are not listed in Table 5-1.

5.3 Items Important to Safety

Q-scenarios are assessed further to identify which of the items in the repository are to be classified as important to safety (see Step 11 of Table 2-1). The Q-scenarios are characterized by failures of certain structures, systems, and components, as shown in Table 5-1. As described in Subsection 2.2, items are important

Scenario Number (see Note b)	Associated Failures
1 - 1	Waste transporter, container
3 - 1	Structure, fire protection system, waste-handling equipment, filtered ventilat
3 - 2	Fire protection system, waste-handling equipment, filtered ventilation exhau
3 - 3	Waste-handling equipment, filtered ventilation exhaust system
4 - 1	Structure, waste-handling equipment, container, filtered ventilation exhaust
4 - 2	Waste-handling equipment, container, filtered ventilation exhaust system
4 - 3	Waste-handling equipment, container
4 - 5	Container, filtered ventilation exhaust system
4 - 6	Container
5 - 1	Structure, waste-handling equipment
5 - 3	Waste-handling equipment
6 - 1	Structure, waste-handling equipment
6 - 2	Waste-handling equipment
7 - 1	Structure, waste-handling equipment, container, filtered ventilation exhaust
7 - 2	Waste-handling equipment, container, filtered ventilation exhaust system
7 - 3	Waste-handling equipment, container
7 - 5	Container, filtered ventilation exhaust system
7 - 6	Container
8 - 1	Structure, waste-handling equipment, container, filtered ventilation exhaust
8 - 2	Waste-handling equipment, container, filtered ventilation exhaust system
8 - 3	Waste-handling equipment, container
8 - 4	Waste-handling equipment
8 - 5	Waste-handling equipment, container
8 - 6	Container
10 - 1	Structure, waste-handling equipment
10 - 3	Waste-handling equipment
11 - 1	Structure, waste-handling equipment
11 - 2	Waste-handling equipment
12 - 1	Structure, waste-handling equipment, container, filtered ventilation exhaust
12 - 2	Waste-handling equipment, container, filtered ventilation exhaust system
12 - 3	Waste-handling equipment, container

Notes: a. Each scenario could result from any of the credible initiating events listed on Table 3-3 (unless th

b. Numbers are assigned to each scenario in the event trees shown in Figure 3-5. The first digit(s)

TABLE 5-1

Q-SCENARIOS

	Scenario Number (see Note b)	Associated Failures
	12-5	Container, filtered ventilation exhaust system
ıtilation exhaust system	12-6	Container
ıaust system	13-1	Structure, waste-handling equipment, filtered ventilation exhaust system
	13-2	Waste-handling equipment, filtered ventilation exhaust system
ust system	13-3	Waste-handling equipment
	14-1	Structure, waste-handling equipment, container, filtered ventilation exhaust system
	14-2	Waste-handling equipment, container, filtered ventilation exhaust system
	14-3	Waste-handling equipment, container
	14-5	Container, filtered ventilation exhaust system
	14-6	Container
	15-1	Structure, waste-handling equipment, container, filtered ventilation exhaust system
	15-2	Waste-handling equipment, container, filtered ventilation exhaust system
	15-3	Waste-handling equipment, container
ust system	15-5	Container, filtered ventilation exhaust system
	15-6	Container
	17-1	Structure, waste transporter, container, filtered ventilation exhaust system
	17-3	Structure, container, filtered ventilation exhaust system
	17-5	Waste transporter, container, filtered ventilation exhaust system
ust system	17-6	Waste transporter, container, filtered ventilation exhaust system
	17-8	Container, filtered ventilation exhaust system
	17-9	Container
	18-1	Structure, waste transporter, container, filtered ventilation exhaust system
	18-2	Structure, waste-handling equipment, container, filtered ventilation exhaust system
	18-3	Structure, container, filtered ventilation exhaust system
	18-5	Waste transporter, container, filtered ventilation exhaust system
	18-6	Waste transporter, container
	18-7	Waste-handling equipment, container, filtered ventilation exhaust system
	18-8	Waste-handling equipment, container
ust system	18-10	Container, filtered ventilation exhaust system
	18-11	Container

s the associated items were identified as important to safety and designed accordingly).

t(s) correspond to the compartment number and the second digit(s) correspond to a sequential number for the scenarios.

to safety if, in the event they fail, an accident could result which causes a dose commitment greater than 0.5 rem (NRC, 1983a). The items important to safety are those that are relied on in accident analyses for preventing or mitigating accidents that could result in an excessive dose (i.e., greater than 0.5 rem).

In the surface facilities (Compartments 1 through 16), accident scenarios that could result in excessive doses can be prevented or mitigated by the functions associated with the hot cell structures, cask receiving and shipping bay structure, fire protection system (for the waste treatment building), waste-handling equipment, containers, and filtered ventilation exhaust systems. The backup electrical power supply system is also needed for proper functioning of the filtered ventilation exhaust systems in the event that offsite power is lost. Therefore, these items should be classified as important to safety and are listed in Table 5-2.

In the underground facilities (Compartments 17 and 18), scenarios that could result in excessive doses can be prevented or mitigated by the functions of the waste transporter, waste-handling equipment, and container. These items should be classified as important to safety and are listed in Table 5-2. Because the transporter and waste-handling equipment can be designed to withstand any failures of the drift structure (e.g., drift collapse), the accident analyses do not need to rely on or take credit for the drift structure's function (i.e., failure or collapse of the drift will not result in excessive doses). The waste transporter and waste-handling equipment protect the container from severe impacts. Similarly, excessive doses can be prevented or mitigated without the need for the filtered ventilation exhaust system (on the surface), because any releases from damaged containers can be contained within the transporter and waste-handling equipment (e.g., shield valve). Therefore, the underground excavations, drifts, waste emplacement ventilation exhaust system, and other items that do not prevent or mitigate excessive doses are classified as not important to safety. These are listed in Table 5-3. More specific design recommendations for

TABLE 5-2

LIST OF REPOSITORY ITEMS CLASSIFIED AS IMPORTANT TO SAFETY

Repository Item (see Note a)	Item Number	Compartment Number (see Note b)
Waste Package	1.2.2	-
Container	-	4, 7-8, 12, 14-15, 17-18
Liner & Mechanical Appurtenances	-	18
Repository	1.2.4	-
Facilities	1.2.4.3	-
Surface Facilities	1.2.4.3.2	-
Waste Handling Facilities	-	1-16
Hot Cell Structures Containing Spent Fuel and HLW		
Cask Receiving and Shipping Bay Structure		
Waste Treatment Building Structure		
Waste Treatment Building Fire Protection System		
Filtered Ventilation Exhaust Systems		
Waste Handling Equipment (see Note d)		
Backup Electrical Power Supply System		
Underground Service Systems	1.2.4.3.5	17-19
Waste Transporter		
Underground Waste Emplacement Equipment		
Ventilation Barrier Doors		

Notes: a. Identification and numbering of repository items subject to the quality level assignment process are taken from DOE (1989a).

b. Repository compartments and numbers are described in Table 3-1.

c. All ESF items will be removed prior to repository operations, except underground openings (shafts and excavations), shaft liners, and ground support (DOE, 1987a).

d. Waste handling equipment on this list include the following items that handle spent fuel and HLW: cranes, electromechanical manipulators, transfer carts, lifting fixtures, waste packaging equipment, cask cavity sampling & venting system, consolidation equipment, waste storage racks, container transfer machine and interfacing shield valves, and other items that handle spent fuel and HLW. Also included are the waste treatment building tanks, process equipment, transfer casks, and cranes that handle highly radioactive site-generated waste.

Repository Item (see Note a)	Item Number	Compartment (see Note
Waste Package	1.2.2	-
Waste Form	-	1, 4-8, 10-15,
Emplacement Borehole	-	18
Repository	1.2.4	-
Seals	1.2.4.2.3	17
Facilities	1.2.4.3	-
Site Preparation	1.2.4.3.1	20
Communications System	-	20
Drainage Control System	-	20
Fencing	-	20
Landscaping	-	20
Railroad	-	20
Roads	-	20
Utilities	-	20
Surface Facilities	1.2.4.3.2	-
Balance of Plant	-	20
Exhaust Shaft Filter Building	-	17
Shafts and Ramps	1.2.4.3.3	-
Emplacement Exhaust Shaft	-	17
Exploratory Shafts	-	17
Men-and-Materials Shaft	-	19
Tuff Ramp	-	19
Waste Rar	-	17
Underground Excavations	1.2.4.3.4	17-19
Underground Service Systems	1.2.4.3.5	17-19
Exploratory Shaft Facility	1.2.6.0	Note c
ESF Site	1.2.6.1	Note c
Main Pad	1.2.6.1.1	Note c
Auxiliary Pad	1.2.6.1.2	Note c
Access Roads	1.2.6.1.3	Note c
Site Drainage	1.2.6.1.4	Note c
Surface Utilities	1.2.6.2	Note c
Power Systems	1.2.6.2.1	Note c
Water Systems	1.2.6.2.2	Note c
Sewage Systems	1.2.6.2.3	Note c
Communication System	1.2.6.2.4	Note c
Mine Wastewater System	1.2.6.2.5	Note c
Compressed Air System	1.2.6.2.6	Note c

Notes:
a. Identification and numbering of repository items subject to the quality le

b. Repository compartments and numbers are described in Table 3-1.

c. All ESF items will be removed prior to repository operations except und

TABLE 5-3

ITORY ITEMS NOT IMPORTANT TO SAFETY

umber b)	Repository Item (see Note a)	Item Number	Compartment Number (see Note b)
17, 18	Surface Facilities	1.2.6.3	Note c
	Ventilation System	1.2.6.3.1	Note c
	Test Support Facilities	1.2.6.3.2	Note c
	Sites for Temporary Facilities	1.2.6.3.3	Note c
17	Parking Areas	1.2.6.3.4	Note c
	Material Storage Facilities	1.2.6.3.5	Note c
20	Shop	1.2.6.3.6	Note c
20	Warehouse	1.2.6.3.7	Note c
20	Temporary Structures	1.2.6.3.8	Note c
20	Communications/Data Building	1.2.6.3.9	Note c
20	First Shaft	1.2.6.4	-
20	Collar	1.2.6.4.1	17
20	Lining	1.2.6.4.2	17
20	Stations	1.2.6.4.3	Note c
-	Furnishings	1.2.6.4.4	Note c
20	Hoist System	1.2.6.4.5	Note c
17	Sump	1.2.6.4.6	Note c
-	Second Shaft	1.2.6.5	-
17	Collar	1.2.6.5.1	17
17	Lining	1.2.6.5.2	17
19	Stations	1.2.6.5.3	Note c
19	Furnishings	1.2.6.5.4	Note c
17	Hoist System	1.2.6.5.5	Note c
7-19	Sump	1.2.6.5.6	Note c
7-19	Underground Excavations	1.2.6.6	-
	Operations Support Areas	1.2.6.6.1	19
ote c	Test Areas	1.2.6.6.2	17-19
ote c	Underground Support Systems	1.2.6.7	Note c
ote c	Power Distribution System	1.2.6.7.1	Note c
ote c	Communications System	1.2.6.7.2	Note c
ote c	Lighting System	1.2.6.7.3	Note c
ote c	Ventilation Distribution System	1.2.6.7.4	Note c
ote c	Water Distribution System	1.2.6.7.5	Note c
ote c	Mine Wastewater Collection System	1.2.6.7.6	Note c
ote c	Compressed Air Distribution Systems	1.2.6.7.7	Note c
ote c	Fire Protection System	1.2.6.7.8	Note c
ote c	Muck Handling System	1.2.6.7.9	Note c
Note c	Sanitary Facilities	1.2.6.7.10	Note c
Note c	Monitoring and Warning System	1.2.6.7.11	Note c
	Underground Tests	1.2.6.8	Note c
	Integrated Data System (IDS)	1.2.6.8.1	Note c

ality level assignment process are taken from DOE (1989a).

1.

pt underground openings (shafts and excavations), shaft liners, and ground support (DOE, 1987a).

the waste transporter, waste-handling equipment, and container are included in Subsection 6.1.2.

As discussed in Section 3.2, the inadvertent detonation of explosives used for mining may be considered a credible event. If the underground ventilation barrier doors fail to control access and fail to preclude the inadvertent introduction of explosives into the waste emplacement area, the waste could be damaged as a result of an accidental explosion. Therefore, the underground ventilation doors are recommended as important to safety and are listed in Table 5-2.

Some of the Q-scenarios could result in accidental criticality if the spent fuel containers failed or were not designed to be critically safe in the event of water intrusion. Therefore, spent fuel containers are important to safety for criticality safety reasons. Section 6.0 describes the recommended general design requirements for criticality safety.

This study assumes that repository waste facilities are located above the probable maximum flood level so that no engineered features would be needed for flood protection; otherwise, some flood protection features may be important to safety.

6.0 SAFETY RECOMMENDATIONS

6.1 Design

This subsection describes the general recommendations for design of items important to safety and then discusses some specific recommendations that result from the preceding analyses.

6.1.1 General

Items important to safety should be designed to more stringent requirements than items that are not important to safety. This approach is reflected in many documents that specify general design criteria for the repository and other nuclear facilities, as discussed below.

DOE Order 6430.1A, General Design Criteria (DOE, 1989b), states that "the design of systems, components, and structures that are not safety class items shall, as a minimum, be subject to conventional industrial design standards, codes, and quality standards. Safety class items shall be subject to appropriately higher-quality design, fabrication, and industrial test standards and codes...to increase the reliability of the item and allow credit to be taken for its capabilities in safety analysis." Also, "safety class items shall be designed to withstand the effects of, and be compatible with, the environmental conditions associated with operation, maintenance, shutdown, testing, and accidents."

Similarly, 10CFR60.131 (NRC, 1989a) requires that "the structures, systems, and components important to safety shall be designed so that natural phenomena and environmental conditions anticipated at the geologic repository operations area will not interfere with necessary safety functions." The regulation also requires that such items be designed to withstand other accidental conditions, such as dynamic effects of equipment failures, fires, explosions, and other emergencies.

As described above, accident conditions (in addition to normal conditions) should be used as a basis for the design of items important to safety. Because items important to safety are relied on in accident analysis, they should be designed to withstand the associated credible initiating events, as listed in Table 3-3. These credible events are called design basis accidents for items important to safety.

Numerous documents describe design basis accidents for nuclear facilities. Section 0111-99 of DOE Order 6430.1A (DOE, 1989b) specifies some general methods to select the design basis tornado and extreme wind, design loads from flooding, design basis earthquake, and other accident conditions that should be used for designing safety class items for nuclear facilities, including repositories. Other reports (Brynda, 1981; Elder, 1986) also provide guidance in selecting design basis accident conditions for use as design requirements for items important to safety.

NRC (1988b) states that "many guidelines and standards have been developed in the reactor program and other nuclear programs which may be applicable for the geologic repository program. For example, these are regulatory guides covering design basis earthquakes, floods, and tornado wind velocities which may be used in the design of the HLW facility and developing the associated Q-list. While some of these guidelines and standards may not be directly applicable to a geologic repository, DOE should consider their use, to the extent practicable, to eliminate the need to develop new approaches."

However, the inherent radiological hazards of a repository are much smaller than those at nuclear power plants. The inventory of radioactive materials in the repository surface facilities is much less than that in an operating reactor. Also, the temperatures, pressures, and other conditions that contribute to the volatility and dispersal of radioactivity are significantly less at a repository. The stringency of design standards should be commensurate with the importance of the safety functions, which reflect the inherent radiological hazards of the facility. The NRC recognizes this approach in the design of nuclear power plants by applying design criteria to radioactive waste facilities that are less stringent than the criteria applied to reactor safety systems. Because the radiological hazards are much smaller at repositories, use of nuclear power plant design basis accidents and associated safety criteria and standards may be overly stringent for repositories in many cases. The design standards for repositories may be less stringent than those for nuclear power plants without compromising radiological safety. This should be considered in the selection of criteria, standards, and design basis accidents for the repository.

As an example, when determining the design basis earthquake conditions for the repository, a peak ground acceleration that is less than a "safe shutdown earthquake" for reactors could be selected. Alternatively, less stringent structural load combinations or higher allowable stresses could be considered for repository design in comparison with reactor designs.

6.1.2 Specific Recommendations

As discussed in Subsection 4.2, certain design requirements would reduce the likelihood of failures of items identified as important to safety, so that many of the scenarios with potentially excessive consequences would not be credible. Recommended design requirements and design changes that result from the accident analyses in previous subsections are presented below.

Accidental doses are calculated at the nearest boundary of the unrestricted area, in accordance with 10CFR60 (NRC, 1989a). Because accidental radiation doses decrease with distance from the point of release, relocating the nearest boundary of the unrestricted area farther from waste-handling facilities would decrease the accidental doses at that location. Moving the boundary an additional several hundred meters away could possibly eliminate several items from the Q-list. This should be considered in future repository designs.

Criticality accidents should be prevented at the repository by (1) the absence of water or other moderating materials in areas containing bare spent fuel and (2) the inherently safe configuration of spent fuel in areas where water could exist (e.g., in shipping casks or in containers emplaced for long durations in boreholes). Therefore, the repository hot cells should be designed to preclude water intrusion and the spent fuel containers may need to be designed to be critically safe when fully flooded.

Subsection 5.3 explained that underground drifts are not important to safety because the transporter and waste emplacement equipment would be designed to protect the waste and confine any potential releases. Designing these items accordingly would prevent accidental damage to the containers and waste in the event of drift collapse or other credible impacts and loads (even during container emplacement or retrieval). Also, the underground emplacement area ventilation exhaust system is not important to safety because the transporter and waste emplacement equipment would be designed to contain any releases of radioactivity from containers damaged during emplacement or retrieval (e.g., accidental drops). Therefore, the transporter and waste emplacement equipment should be designed to provide these functions of protection and confinement of waste during all credible events. This includes drift collapse, single failure of brakes (e.g., provide redundant brakes), and malfunction of stabilizing jacks (e.g., provide redundant stabilizers).

The consequences of other potential accidents can be reduced by incorporating certain design features. In the SCP-CDR, shipping casks are lifted from their carriers by overhead cranes in the cask receiving and shipping bay and moved to below-grade cask transfer cars. The cask receiving and shipping bay can be designed to enhance safe handling of casks (e.g., eliminating the possibility of lifting one cask over another, etc.). Also, separating the cask preparation area from the cask receiving and shipping bay, and providing a filtered ventilation exhaust system for the cask preparation area would ensure that potential releases of radioactivity in this area are filtered prior to discharge to the atmosphere. In addition, potential damage to open (unsealed) containers of spent fuel can be reduced by designing the associated handling facilities so that the lifting heights are minimized (thus, minimizing potential drop heights).

6.2 Operations

Certain operational limitations will be needed for safe handling of radioactive materials at the repository. These limitations are reflected by the assumptions or conditions associated with the accident analyses, as described below.

Spent fuel and HLW acceptance criteria will be needed to ensure that the types of radioactive materials received at the repository can be safely handled and that the accident analyses bound all possible conditions. For example, the results of the accident analyses in this report are valid on the condition that spent fuel has a maximum burnup of 60 GWd/MTU and a minimum cooling time of 5 yr out of the reactor. These types of limitations on radioactive materials received at the repository should be considered in the operational procedures for the repository.

Consequences of accidental releases from various hot cells and repository compartments will depend on the inventory of radioactive material in each hot cell or in each area that could be affected by an accident. The inventory of radioactive materials in each hot cell or compartment should therefore be limited to the minimum necessary for proper handling and processing of the waste for disposal. Excessive accumulation of radioactive materials in one area may result in situations that are not covered by the safety analyses, and therefore the inventories of radioactive materials should be limited by design and operational procedures.

Certain accidents can be prevented by proper operational controls, such as precluding explosives or other hazardous materials from the vicinity of radioactive materials, and preventing transporter collisions during container emplacement or retrieval by limiting access to underground emplacement drifts to one transporter at a time. Periodic surveillance and inspections can also be performed to prevent the accumulation of combustible materials in repository facilities.

6.3 Quality Assurance

Subpart G of 10CFR60 (NRC, 1989a) requires that "DOE shall implement a quality assurance program based on the criteria of Appendix B of 10 CFR Part 50... The quality assurance program applies to all systems, structures, and components important to safety." Appendix B of 10CFR50 (NRC, 1989d) states that "quality assurance comprises all those planned and systematic actions necessary to provide adequate confidence that a structure, system, or component will perform satisfactorily in service." Quality assurance (QA) provides additional confidence that the design requirements (see Subsection 6.1) are properly implemented during fabrication and construction. NRC (1988b) provides the NRC staff positions on QA for geologic repositories, including guidance on applying QA measures to items important to safety.

Appropriate QA programs have been implemented successfully in the design, construction, and licensing of nuclear power plants. However, the stringency of requirements should be commensurate with the importance of the safety functions or the inherent radiological hazards of a facility. This pertains to QA requirements in addition to design standards, as discussed in Subsection 6.1. Appendix B of 10CFR50 (NRC, 1989d) states that "the quality assurance program shall provide control over activities affecting the quality of the identified structures, systems, and components, to an extent consistent with their importance to safety." Because the radiological hazards are much smaller at repositories, the QA requirements could be less stringent than those for nuclear power plants. This should be considered in the development of the QA program for the design and construction of repository items important to safety.

7.0 CONCLUSIONS

7.1 Results

This assessment recommends which structures, systems, and components of the Yucca Mountain repository are important to safety. A list of repository items important to safety is given in Table 5-2. These items include the structures that house spent fuel and HLW, the associated filtered ventilation exhaust systems, certain waste-handling equipment, the waste containers, and other items listed on the table. A list of repository items that are not important to safety is given in Table 5-3.

This report provides the documentation demonstrating that each step of the DOE procedure AP-6.10Q (in Appendix A) has been completed. This work was completed in April, 1990 and AP-6.10Q was later superseded. Table 2-1 identifies the required steps of the assessment and the corresponding subsection of this report in which each step is discussed.

Safety recommendations that result from the analyses are presented in Section 6.0, including recommended design requirements (for items important to safety), operational limitations, and QA stringency.

7.2 Areas Requiring Further Evaluation

The assessment of identifying items important to safety should be reviewed, revised, and updated in each design stage (see Step 13 of Table 2-1). The list of items important to safety could be different if the repository design changes. In addition, as additional information is developed in further stages of the design, the list of items subject to the quality level assignment process (DOE, 1989a) should also be updated to show a more detailed breakdown of repository structures, systems, and components. The assessment of items important to safety should be updated to reflect the new design information and to determine a more detailed breakdown of the structures, systems, and components important to safety.

When meteorological data specified by NRC (1982) become available for the Yucca Mountain site, more realistic atmospheric dispersion models can be used. The corresponding atmospheric dispersion factors may be significantly less than those used in this study. Consideration of this data in future accident analyses could result in changes to the Q-list.

More detailed requirements need to be developed for design of items important to safety. These requirements should reflect the considerations discussed in Section 6.0.

In support of future accident analyses for the repository, further evaluation and testing of the generation of respirable particles resulting from impacts on spent fuel are recommended.

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Appendix A

PROCEDURE AP-6.10Q, IDENTIFICATION OF ITEMS IMPORTANT TO SAFETY

Title

AP-6.10Q IDENTIFICATION OF ITEMS IMPORTANT TO SAFETY

1.0 PURPOSE AND SCOPE

1.1 The purpose of this procedure is to identify the exploratory shaft facility (ESF) and repository structures, systems, and components important to safety (ITS) which are subject to 10 CFR 60, Subpart G Quality Assurance requirements. This procedure specifies the responsibilities and the methods to be used.

1.2 To determine items important to safety, assessments are applied to the appropriate and available repository desigr configuration including the incorporation of all ESF items. The assessments evaluate potential preclosure accident conditions during the repository waste-receiving, handling, processing, emplacement, caretaking, performance conformation, and decommissioning operations. References are given that contain examples of the application of such assessments to a repository conceptual design (SAND84-2641-F).

1.3 This procedure is iterated or repeated for each completed design phase of a repository or an ESF in order to review, identify, revise, and establish the final list of items important to safety.

2.0 APPLICABILITY

This procedure applies to the Yucca Mountain Project Office, Project participants and their contractors and subcontractors engaged in either the ESF design and construction, repository design and construction or the preclosure performance assessments of the potential repository accident conditions used to establish the repository items important to safety.

3.0 DEFINITIONS

3.1 ACTIVITIES

3.1.1 Activities means deeds, actions, work, or performance of a specific function or task. In the HLW geologic repository program, the 10 CFR Part 60 Subpart G QA program applies to activities affecting the quality of all systems, structures, and components important to safety, and to the design and characterization of barriers important to waste isolation. These activities include: site characterization, facility and equipment construction, facility operation, performance confirmation, permanent closure, and decontamination and dismantling of surface facilities as they relate to items important to safety and barriers important to waste isolation (10 CFR 60.151). In addition, the pertinent requirements of 10 CFR Part 50 Appendix B apply to all activities affecting the quality of structures,

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systems, and components important to safety and engineered barriers important to waste isolation. These activities include: designing (including such activities as safety analyses, laboratory testing of waste package materials to characterize their performance, and performance assessments), purchasing, fabricating, handling, shipping, storing, cleaning, erecting, installing, inspecting, testing, operating, maintaining, repairing, and modifying. These types of activities do not need to be identified as part of the Q-List or Quality Activities List. However, activities related to natural barriers important to waste isolation should be identified and listed on a Quality Activities List. These activities include: performance assessments, site characterization testing, and activities that may impact the waste isolation capability of the natural barrier. For example, site characterization activities such as exploratory shaft construction, borehole drilling, and other activities that could physically or chemically alter properties of the natural barriers in an adverse way. (NUREG-1318)

3.2 CONSEQUENCE ANALYSIS

Consequence analysis is a method by which the consequences of an event are calculated and expressed in some quantitative way, e.g., money loss, deaths, or quantities of radionuclides released to the accessible environment.

3.3 CREDIBLE EVENT OR CREDIBLE ACCIDENT

"Credible event or credible accident" means an event or accident scenario which needs to be considered in the design of the geologic repository (NUREG-1318).

3.4 DESIGN BASIS ACCIDENT

A design basis accident (DBA) is a set of well-defined postulated accidents chosen to establish or measure the adequacy of the safety design of the facility.

3.5 DETERMINISTIC SAFETY ANALYSIS

"Deterministic safety analysis" is a form of safety analysis intended primarily to generate safety design parameters for a facility rather than to measure its safety. Deterministic safety analyses are characterized by (1) evaluation of accident processes and consequences but not of accident likelihood, (2) the use of selected, representative accidents (generally design basis accidents) rather than a comprehensive, complete set of accidents to which the facility might be subject, and (3) the use of pessimistic assumptions and conservatism intended to ensure the presence of margins in the design (but at some cost to the realism of the analysis).

Enhanced margins in the design provide safety margins to account for uncertainties in the assumptions and inputs to the analyses.

3.6 EVENT TREE ANALYSIS

An event tree analysis defines a comprehensive set of accident sequences that encompasses the effects of all realistic and physically possible potential accidents. By definition, an initiating event is the beginning point in the sequence. Hence, a comprehensive list of accident-initiating events must be compiled to ensure that the event trees properly depict all important sequences.

3.7 EXTERNAL EVENTS

External events are those caused by natural phenomena or human activities external to the repository.

3.8 FAULT TREE ANALYSIS

A fault tree analysis examines the various ways in which a system designed to perform a safety function can fail. Each system identified in the event tree as involved in an accident is examined to determine how failures of components within that system could cause the failure of the entire system (NUREG-1318).

3.9 IMPORTANT TO SAFETY

Important to safety, with reference to structures, systems, and components, means those engineered structures, systems, and components essential to the prevention or mitigation of an accident that could result in a radiation dose to the whole body, or any organ, of 0.5 rem or greater at or beyond the nearest boundary of the unrestricted area at any time until the completion of permanent closure (10 CFR 60.2)

3.10 INITIATING EVENT

An initiating event is the starting point of an accident sequence that is generally depicted in an event tree analysis. Initiating events are also used as the starting point in design basis accidents.

3.11 INTERNAL EVENTS

Internal events are those caused by failures or operator activities at the repository.

3.12 INTERACTION MATRIX

Interaction matrix is a systematic way to develop potential initiating events for each of the system compartments in the repository.

3.13 ITEMS IMPORTANT TO SAFETY

Items important to safety are those engineered structures, systems, and components essential to the prevention or mitigation of an accident that could result in a radiation dose to the whole body, or any organ, of 0.5 rem or greater at or beyond the nearest boundary of the unrestricted area at any time until the completion of permanent closure. (NUREG-1318)

3.14 MITIGATIVE SYSTEM

A mitigative system is any system whose design and function actively or passively reduces the severity or consequences of an event once the event has occurred.

3.15 NON-MECHANISTIC FAILURES

Non-mechanistic failures are postulated failures which are not based on previously observed modes or mechanisms but which are assumed to provide conservatism in safety assessments.

3.16 PREVENTIVE SYSTEM

Preventive means to keep from happening or to avert some occurrence from taking place. Hence, a preventive system is one which anticipates some undesirable occurrence or process and counters it in advance of its actual occurrence.

3.17 PROBABILISTIC RISK ASSESSMENT

Probabilistic risk assessment (PRA) (also called "probabilistic safety analysis") is a structured and methodological analytical approach to safety analysis intended primarily to give a realistic picture of the safety profile or risk of the facility.

3.18 Q-LIST

In the geologic repository program, a list of structures, systems, and components important to safety, and engineered barriers important to waste isolation that must be covered under the QA requirements of 10 CFR 60, Subpart G. (NUREG-1318)

3.19 SAFETY ANALYSIS

A safety analysis is a process to systematically identify the hazards of a DOE operation, to describe and analyze the adequacy of the measures taken to eliminate, control, or mitigate identified hazards, and to analyze and evaluate potential accidents and their associated risks.

3.20 SCENARIO

A scenario is an account or sequence of a projected course of action or event.

3.21 UNDERGROUND FACILITY

Underground facility is the underground structure, including openings and backfill materials, but excluding shafts, boreholes, and their seals. (10 CFR 60.2)

3.22 UNRESTRICTED AREA

An unrestricted area is any area to which access is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, and any area used for residential quarters.

4.0 RESPONSIBILITIES

4.1 YUCCA MOUNTAIN PROJECT MANAGER (PM)

The PM assigns a Technical Project Officer (TPO) or a Project Designee to ensure that the provisions of this procedure are implemented. The PM authorizes modification or creation of the list of Items Important to Safety. From time to time, the PM may direct that technical assessment reviews are conducted on the results of this procedure.

4.2 The PM shall assign the responsibility to the cognizant TPO or a Project Designee to implement this procedure and assign personnel to identify items important to safety in the ESF and the repository designs.

4.3 The Yucca Mountain Project Quality Manager and Systems Branch Chief (or their designees) are responsible for review and approval of the lists of items important to safety, items not important to safety, and any reports completed and approved by the TPO as a result of implementing this procedure. The purpose of the review is to provide assurance that the candidate list is consistent with Project Office and participant procedures. The approval does not indicate authentication of the technical data or interpretations contained in the document, nor does the approval relieve the assigned participant of the responsibility for the defense of technical data or interpretations contained therein.

4.3.1 The PM shall issue the results to the Change Control Board (CCB) for construction and baseline control of the Project Q-List.

4.4 TECHNICAL PROJECT OFFICER (TPO)

The TPO shall assign an appropriately qualified participant staff member (PSM) to perform the assessment and to develop the list of items important to safety. The TPO shall ensure that qualified individuals perform any technical reviews of the completed assessments of the items important to safety. After the PSM completes the assessments, the TPO shall, after review, approve and transmit the lists of items important to safety, items not important to safety, and other assessment documentation to the PM.

4.5 YUCCA MOUNTAIN PROJECT PARTICIPANT STAFF MEMBER(S) (PSM)

The PSM shall assemble a group of people from multiple engineering, technical, and scientific disciplines, including personnel who were not a part of the original design team to implement the AP 6.10Q assessments. The group shall be referred to as the Assessment Team.

4.6 The Assessment Team shall carry out the procedure by evaluating the responses, including the offsite doses consequences, of the facility design for credible accident conditions that might affect the facilities performance. The calculated performance predictions shall be compared with the regulatory dose criteria to determine which items from AP-6.9Q should be classified as items important to safety.

4.7 The Assessment Team shall produce a list of the items classified as important to safety (i.e., a major input for the Q-list). The team shall also produce a report that documents the assessments conducted to implement the procedure. A list shall also be prepared of the items classified as not important to safety.

4.8 After completion of the assessment, the PSM shall review and revise any previous list of items important to safety developed in accordance with AP-6.10Q and/or AP-5.4Q. If a previous assessment has assigned a different Quality Level or classification of an item, the PSM shall notify the cognizant TPO or Project Designee that a change request needs to be initiated for the Q-List maintained by the CCB.

4.9 The lists of items important to safety, items not important to safety, and the supporting report documentation shall be submitted by the PSM to the cognizant TPO or the Project Designee for approval and transmittal to the PM.

4.10 CHANGE CONTROL BOARD (CCB)

The CCB shall receive the approved list of items important to safety from the PM and combine this list with any list of items important to waste isolation from AP-6.8Q to compile the Yucca Mountain Q-list. The CCB will, after their approval, baseline the Q-list and maintain the official Project Q-list.

4.11 Exhibit 1 is a flow chart summarizing these responsibilities discussed in 4.1 to 4.10.

5.0 PROCEDURE

5.1 This procedure generates a list of items important to safety. Exhibit 2 summarizes the major steps involved in the procedure.

5.2 As indicated in Step 1 of Exhibit 2, a documented repository and ESF design configuration shall be selected by the assessment team for the application of this procedure. The assessment team shall document the design documents used in their assessments.

5.3 In Step 2, the documented design configuration shall be separated into small zones or areas called facility and system compartments. The compartments shall be named uniquely and shall be selected to facilitate a systematic assessment process.

5.4 In Step 3, all of the items from AP-6.9Q shall be assigned a compartment location and the results documented.

5.5 In step 4, site specific initiating events shall be identified and screened for applicability to all compartments. Initiating events shall be separated into internal and external initiating events. Lists of credible and significant internal and external initiating events requiring further assessment shall be developed on a compartment-by-compartment basis.

5.6 To establish the internal initiating events in 5.5, at least two methods shall be used to generate the list. The methods and the screening criteria shall be documented. The screening process should not reject a credible event that could lead to a significant radiological release yet should reduce the number of events requiring detailed assessments in Step 5.

5.7 Although not mandatory, survey forms and interaction matrices are two methods that have been used in previous repository assessments to identify internal events. The survey forms document accident scenarios for each compartment that are judged by experienced designers to be credible. Interaction matrices identify items in each compartment and use the items as row and column designators. Each row in the matrix is then analyzed column by column to identify possible interactions between items and then potential initiating events and credible accident scenarios are developed and documented.

5.8 To establish the external initiating events in 5.5, a checklist of a wide spectrum of external events shall be used in conjunction with site-specific screening criteria. The checklist, the screening criteria, and the list of credible initiating events requiring further assessment shall be documented.

5.9 In step 5, event trees shall be developed for each internal and each external event in the screened list to depict, logically and systematically, the various accident scenarios. The intermediate events in the event trees shall represent responses of various items in the facility design that occur after the initiating event and hence continue the accident progression into an accident scenario (NUREG/CR-2300).

5.10 In Step 5, fault trees shall not be developed until the advanced conceptual repository design is completed due to the lack of sufficient design details for their development until the advanced conceptual design is completed. Fault trees shall be used to systematically examine the various ways that a system, an item or a major component can fail and result in an initiating event or an intermediate event in an accident scenario.

5.11 In Step 6, offsite dose consequences shall be calculated for each branch in the event tree. The dose consequences shall be calculated for a 50-yr dose commitment to a maximally exposed member of the offsite public at the nearest boundary of the unrestricted area.

5.12 Assessments shall be conducted to calculate source terms and the associated offsite doses. To establish radioactive source terms, the quantities of radioactive materials present, the chemical and physical forms of radioactive materials, the radionuclide content, and the accident conditions shall be considered. Estimates of release fractions of radionuclides for each specific accident scenario shall be made and documented based on their physical and chemical properties and the accident conditions at the time of the release.

5.13 The dose assessments shall be calculated as the total of the external exposure from the passing cloud and the internal exposure from inhalation of radionuclides in the cloud. Dose calculations shall be performed using:

1. X/Q values obtained from Regulatory Guide 1.25 and Regulatory Guide 1.3.

2. Immersion 50-yr dose conversion factors obtained from Regulatory Guide 1.109 and NUREG/CR-1918.

3. Internal 50-yr dose equivalent conversion factors obtained from Regulatory Guide 1.109; NUREG/CR-0150, Volume 3, and NUREG/CR-0172.

4. The radionuclide inventory (Ci/MTU) of the spent fuel shall be obtained from ORNL/TM-9591. If site meteorology is available, the X/Q from Regulatory Guide 1.145 may be used to establish the dose if radioactive plume meander and directionality are to be taken into account.

5.14 In Step 7, the probability or frequency of occurrence of the accident scenarios in the event trees shall be classified. It is sufficient to denote these events as either credible or not credible. It is not required to determine a numerical probability for external, internal, and intermediate events in the event trees. Similarly, numerical values for fault trees are not required.

5.15 Assessments of the probability of occurrences of initiating and intermediate events shall be based on the following considerations:

1. Use of existing or published data.
2. Accepted predictive techniques.
3. Analyses of the performance of the system, and
4. Engineering judgment and experience.

5.16 The probability assessments may utilize previously published data of equipment failures and documented judgments of engineers and technical specialists experienced in nuclear facility designs and their potential failure modes.

5.17 Although not mandatory, standardized forms have been used in previous repository assessments to document judgments.

5.18 The event trees constitute a data base for establishing the list of items important to safety. The regulation 10 CFR 60 provides a single criterion, a dose specification, for identifying items important to safety

The following two considerations shall be used in Step 8 to identify items important to safety:

1. The dose criterion-an accident scenario must cause an offsite dose of 0.5 rem or greater to merit consideration in identifying items important to safety.

2. The probability criterion-an initiating event (internal or external) or an accident scenario must either be termed "credible" or be estimated to have a probability of occurrence greater than 1×10^{-6}/year to be considered in identifying items important to safety.

5.19 In addition to the two above considerations in 5.18, other considerations shall be used in Step 9 to identify items important to safety based on other project criteria such as:

1. Probability of occurrence.
2. Historical licensing experience.
3. Consensus judgment.

5.20 Using the criteria in 5.18 or 5.19, the event trees shall be assessed in Steps 8, 9, 10, 11, and 12 to identify which items established in the design or AP-6.9Q are important to safety. If the dose screening criterion of 0.5 rem is exceeded in a credible accident scenario, that scenario shall be classified in Step 8 as a Q-scenario. The Q-scenario shall be further assessed in Step 10 to identify specific items important to safety. Scenarios not exceeding these criteria of 5.17 and 5.18 are classified as not Q-scenarios or NQ-scenarios.

5.21 All NQ-scenarios from Step 8 shall be assessed again in Step 9 using the criteria of 5.19 in order to introduce a degree of conservatism into the assessments of items important to safety. Because of this conservatism, which could be unnecessarily excessive, some NQ-scenarios from Step 8 reclassified as Q-scenarios in Step 9 may be reclassified as NQ during a subsequent assessment using this procedure. In such cases, all items involved in the reclassified Q-scenario will be removed from the list of items classified as important to safety.

5.22 For Step 9, scenarios not satisfying the criteria of Step 8 shall be reclassified as Q-scenarios (1) if the scenario is sufficiently similar to others historically classified as Q-scenarios, or (2) when practical considerations based on judgment indicate it could be a Q-scenario, or (3) a calculated probability is sufficiently close to either of the two probability criteria of 5.18 that a variation in assumptions or data could cause either criterion to be exceeded, or (4) when both dose consequences and a calculated probability are sufficiently close to the criteria values in 5.18 that a variation in assumptions or data could cause them to exceed these values. Scenarios not reclassified in Step 9 as Q-scenarios shall remain as NQ-scenarios.

5.23 In Step 10, all NQ-scenarios shall be eliminated from further consideration in identifying items important to safety.

5.24 In Step 11, the Q-scenarios from Steps 8 and 9 shall be assessed further to identify which of the possible items in the facility design or established in AP-6.9Q are to be classified as important to safety. The assessment shall determine which role specific items play in the accident scenarios. These assessments and the rationale for assigning specific items as important to safety shall be documented.

5.25 The assessment in Step 11 shall include a classification of items from AP-6.9Q. The results shall include a summary tabulation of the items compartment location, their classification, and a basis for their classification as either important to safety or not important to safety. Exhibit 3 is a sample format for reporting the summary tabulation of items not important to safety.

5.26 Considerations for classifying specific items as important to safety may include:

1. Their failure directly causes the release of radioactive materials that exceed the 0.5 rem dose criterion.

2. Their failure causes the loss of essential consequence mitigating items that are relied on to lower the probability of exceeding any offsite accident dose limit criterion (e.g., 5 rem) to less than 10^{-6}/year, taking into account the initial failure probability.

5.27 In Step 12, a summary listing of all items classified as important to safety shall be compiled and documented. The sample format for reporting this compilation is shown as Exhibit 4 and shall be referred to as the list of items important to safety.

5.28 The assessment in this procedure is iterative. In the facility design context, iterative means that each stage of design generated in the design description documents shall be assessed using the process in Exhibit 2 and the list of items important to safety (Exhibit 4) revised if necessary.

5.29 In Step 13, any list of items important to safety from an earlier design stage shall be reviewed, revised, and updated to reflect the current design stage and assessment using this procedure. In this iterative design process, some items initially classified as important to safety and hence placed in the Project Q-list will likely be removed and some new items added. This iterative process is illustrated by the feedback loop in Exhibit 2.

5.30 The results from these assessments to identify items as important to safety shall be used to guide the design process by feeding back new requirements to the facility designers or to the design bases (WM-87). Such recommendations from these assessments for new requirements, which should result in an overall improvement in the safety of the repository design, shall be documented and be included in the assessment documentation as recommendations for further evaluation by those responsible for the facility design.

5.31 All source information on which the analyses of items important to safety is based will be listed in the documentation of the results of this procedure and will be baselined as discussed in Section 5.35. This listing must be sufficient to uniquely identify the specific sources of information used.

5.32 To implement this procedure, the PM shall assign the cognizant TPO or the Project Designee to implement this procedure. The TPO shall assign a PSM. The PSM shall appoint an assessment team and conduct the assessments required by this procedure.

5.33 When the PSM completes the assessment, the PSM shall transmit to the TPO for approval the results which include: (1) the list of items important to safety, (2) the list of items not important to safety and (3) any report documentation. The report documentation shall include objective evidence, or reference thereto, demonstrating that each step in the process shown in Exhibit 2 has been completed.

5.34 The TPO shall review, approve, and transmit the results of implementing this procedure to the PM.

5.35 The PM (or assigned Designees) shall, after review, accept the results approved by the TPO. The purpose of the review is to provide assurance that the candidate list is consistent with Project procedures. The approval does not indicate authentication of the technical data or interpretations contained in the document, nor does the approval relieve the assigned participant of the responsibility for the defense of technical data or interpretations contained therein. The PM (or assigned Designees) shall transmit the list of items important to safety and the associated source information (para. 5.31) to the Project Change Control Board to be baselined in accordance with AP-3.3Q. The CCB will transmit the baselined list to Document Control for distribution and control in accordance with AP-1.5Q (Document Control).

Step 1: Select documented design configuration.

Step 2: Define facility and system compartments.

Step 3: Assign compartment locations to items from AP-6.9Q.

Step 4: Identify and screen initiating events to establish credible and significant internal and external events.

Step 5: Develop event trees for accident scenarios. If necessary, develop fault trees.

Step 6: Estimate dose consequences for event trees.

Step 7: Classify accident scenarios as (1) credible, (2) not credible or (3) make (optional) qualitative estimates of frequency of occurrences.

Step 8: Identify credible scenarios in event trees that exceed dose criterion and denote as Q-scenarios requiring further assessment.

Step 9: Identify any other scenarios in event trees that exceed other project criteria and denote as Q-scenarios requiring further assessment.

Step 10: Eliminate all NQ-scenarios in event trees from further assessment.

Step 11: Evaluate all Q-scenarios to identify specific items important to safety.

Step 12: Construct list of items identified as important to safety.

Step 13: Repeat, or iterate, steps 1 to 12 for the various stages of design and review, revise, and update items previously identified as important to safety.

Exhibit 2. General Steps: Flow Chart for Identifying Items Important to Safety.

CLASSIFICATION OF ITEMS: SUMMARY OF ITEMS NOT IMPORTANT TO SAFETY

ITEM	COMPARTMENT LOCATION(S)	COMMENTS

Exhibit 3. Sample Format for List of Items Classified as Not Important to Safety.

LIST OF ITEMS CLASSIFIED AS IMPORTANT TO SAFETY

ITEM	COMPARTMENT LOCATION(S)	COMMENTS

Exhibit 4. Sample Format for List of Items Classified as Important to Safety.

Appendix B

INFORMATION FROM THE REFERENCE INFORMATION BASE USED IN THIS REPORT

This report contains no information from the Reference Information Base.

CANDIDATE INFORMATION FOR THE REFERENCE INFORMATION BASE

This report contains no candidate information for the Reference Information Base.

CANDIDATE INFORMATION FOR THE SITE & ENGINEERING PROPERTIES DATA BASE

This report contains no candidate information for the Site and Engineering Properties Data Base.

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