1.0 INTRODUCTION

This section consists of two parts. The first part describes the process by which items and activities are determined to be important to safety or waste isolation; the second part describes the Graded Quality Assurance Program to be applied to these as well as other items and activities associated with the Basalt Waste Isolation Project (BWIP).

2.0 Q-LISTED ITEMS AND ACTIVITIES

This section describes a general philosophy to be used by the Rockwell Hanford Operations (Rockwell) BWIP organization in determining the structures, systems, components, and activities that are important to safety or waste isolation for the mined geologic disposal system (MGDS). The resulting list of structures, systems, components, and activities that are important to safety or waste isolation is called the "Q-List" (see glossary). The items and activities on the list will be subject to the highest quality level of a formal quality assurance program as required for site characterization and licensing of the geologic repository and will be subject to U.S. Nuclear Regulatory Commission (NRC) licensing review and oversight. Although other items and activities will not be subject to regulatory control, the NRC may examine any item or activity not on the Q-List to ensure that no items or activities important to safety or waste isolation have been omitted from the Q-List.

The Q-List will change over time, with a final list emerging at the completion of NRC's review of the U.S. Department of Energy (DOE) license application. During the evolutionary period, two milestones stand out: (1) the Q-List to support the Site Characterization Plan (SCP) data-gathering and design efforts and (2) the Q-List to support the License Application Design (LAD) phase. The methodology to generate the SCP phase Q-List is based primarily on engineering judgment and is described in section 2.1. As site characterization and design activities progress to the point of allowing quantification of key input parameters, the LAD phase methodology described in section 2.2 will be implemented.

2.1 DETERMINATION OF Q-LIST ITEMS AND ACTIVITIES FOR SITE CHARACTERIZATION PHASE

2.1.1 Items and Activities Important to Safety at Site Characterization Plan Design Phase

Structures, systems, components, and activities that are important to safety are defined by the NRC in 10 CFR 60, "Disposal of High-Level Radioactive Wastes in Geologic Repositories," section 60.2, 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."

An equivalent statement used by the NRC in its draft "Generic Technical Position on Licensing Assessment Methodology for High-Level Waste Geologic Repositories," is

> "...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."

Items and activities important to safety must be on the Q-List to ensure that the design addresses their safety requirements and that appropriate quality assurance controls are applied. Central to the above NRC definitions is the dose consequence of the failure of the items. The assessment of the dose consequences of the failure of structures, systems, or components, however, requires a detailed assessment of their functions under design basis conditions that are not available until (1) the design effort attains a certain maturity, (2) design basis conditions are identified, and (3) the analytical assumptions to be employed during safety analysis are established. Prior to that time, the methodology that is employed will be based on the application of the criteria specified below through the use of engineering judgment. The methodology for the LAD phase (i.e., the mature design) is described in section 2.2.

The NRC definition of "important to safety" contains a criterion for assigning an item to the Q-List: whether the item can prevent or mitigate an accident that could result in a dose in the uncontrolled area of 0.5 rem or greater. Another criterion may be inferred from the definition, although it is not explicitly stated: a determination that the accident scenario is credible. The term "credible accident" as used here implies that the accident has an overall probability of occurrence that is smaller than the probability for anticipated operational events but yet not so small as to be considered insignificant or incredible. The

quantitative limit below which an event ceases to be considered credible is not identified in 10 CFR 60. Specific direction is provided, however, under methodology "A" in "Guidance for Developing the SCP-CDR and SCP Q-List," 1986, Office of Civilian Radioactive Waste Management.

For purposes of identifying structures, systems, components, and activities as important to safety, failures that would have an annual probability of 1×10^{-5} or less of exceeding the 0.5 rem threshold will be disregarded. The probability of incurring a health effect from a one-rem whole body exposure is on the order of 1×10^{-4}. The combined annual probability of incurring a health effect among the offsite population is therefore less than 1×10^{-9} from a failure with a probability of 1×10^{-5} or less per year of resulting in an offsite dose of 0.5 rem. This risk is significantly smaller than "risks that would be regarded as negligible by the exposed individuals," which are on the order of 1×10^{-6} health effects per year.

The dose consequence estimate should be based on a radiation transport model that uses conservatively estimated parameters where design and site details are lacking. One such model is set forth in NRC Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequence of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors."

To summarize, the important to safety Q-List at the SCP design phase shall be composed of the structures, systems, components, and activities whose probability of failure is greater than 10^{-5} per year, with dose consequences exceeding 500 mrem. Additional factors or a lower dose threshold shall be considered to account for technical and other uncertainties.

2.1.2 Items and Activities Important to Waste Isolation at Site Characterization Plan Design Phase

From 10 CFR 60.2, it may be inferred that structures, systems, and components important to waste isolation would be those natural and engineered barriers that are relied on to inhibit "...the transport of radioactive material so that amounts and concentrations of this material entering the accessible environment will be kept within prescribed limits." These items must function in a certain way in order to meet the long-term isolation objective after repository closure.

In 10 CFR 60, paragraph 60.113, the NRC has defined performance objectives for the repository after closure. The four performance objectives are related to the following performance measures:

- Waste package containment time
- Rate of release of radionuclides from the engineered barrier system
- Preemplacement groundwater travel time
- Cumulative release to the accessible environment.

Consequently, structures, systems, and components important to waste isolation may include shaft and borehole seals, engineered barriers (e.g., waste package), and features of the natural site system.

Performance goals will be set for selected structures, systems, components, and activities of the MGDS, which, when attained, will provide appropriate assurance that the above performance objectives are met. Structures, systems, components, and activities necessary to comply or demonstrate compliance with these performance objectives will be placed on the Q-List.

The designation of structures, systems, components, and activities to be placed on the Q-List at the SCP design phase, and all site characterization activities that are essential to adequately evaluate these items, shall be based on technical judgment of the items that will be required to comply or demonstrate compliance with the repository performance objectives as the repository performance analyses are completed.

As the site characterization activities take place and as the understanding of the site changes, the performance goals may change. As a consequence, some changes to the Q-List are expected as the site characterization program progresses. All site characterization tests and activities must therefore be carefully planned and must not only take into account the primary tests for the items initially placed on the Q-List, but must also include some contingency for items that may be later added to the Q-List. A conservative approach at the SCP design phase is to be adopted to ensure that data necessary to demonstrate compliance with 10 CFR 60 are obtained and preserved in accordance with quality assurance requirements.

2.2 DETERMINATION OF Q-LIST ITEMS AND ACTIVITIES FOR LICENSE APPLICATION DESIGN PHASE

2.2.1 Items And Activities Important to Safety at License Application Design Phase

A risk assessment methodology shall be used for preclosure safety analysis and for the identification of structures, systems, components, and activities important to safety. An example of such a methodology is described in NUREG/CR-4303, "High-Level Waste Preclosure Systems Safety Analysis, Phase 1 Final Report."

The Q-List development shall be accomplished by examining an event sequence frequency and dose consequence. The structures, systems, components, and activities that are involved in the event sequence shall be examined to determine their contribution to risk. If the consequence of an item failure (and the associated accident scenario) is greater than or equal to 0.5 rem at the nearest boundary of the unrestricted area, and if the item failure probability is greater than 10^{-5} per year, then the item shall be placed on the Q-List.

2.2.2 Items and Activities Important to Waste Isolation at License Application Design Phase

The determination of the structures, systems, components, and activities important to waste isolation at the LAD phase shall be accomplished in a similar fashion to that required at the SCP phase. However, at this more mature stage, the evaluation of importance to isolation shall be based on direct assessments of whether the performance objective will be met rather than indirect assessments based on the preliminary performance goals set at the SCP phase. Throughout site characterization and performance assessment, activities shall be geared to demonstrate compliance with the performance objective of 10 CFR 60, including the release limits set in the U.S. Environmental Protection Agency document 40 CFR 191.

Retrieval of the waste from the repository for public radiological safety, resource recovery, or because of Congressional mandate is a design contingency and shall be treated in the same manner as waste emplacement. The same procedures and criteria used to classify the items and activities needed to emplace the waste shall be used to determine if equipment and activities needed to retrieve the waste should be included on the Q-List.

3.0 GRADED QUALITY ASSURANCE PROGRAM

The purpose of a graded quality assurance program is to select the quality assurance measures to be applied to items and activities in the repository program consistent with their importance to safety, waste isolation, and the achievement of program success. This will be accomplished by deliberate quality planning and selective application of quality assurance criteria to the item or activity to be performed, with varying degrees of quality assurance controls (e.g., implementation of criteria requirements) applied depending on item function, complexity, consequence of failure, reliability, replicability of results, and economic considerations.

This approach involves identifying those items and activities whose failure could cause undue risks to the public and facility personnel and/or extended shutdown of the facility with critical economic losses, and ensuring that they are covered with a commensurate quality assurance program. On the other hand, an item whose failure or malfunction could result only in operational inconvenience or negligible economic loss may warrant only a quality inspection by the purchaser on delivery of the item. Between these two extremes, there are varying degrees of quality assurance to achieve the desired confidence in the quality of the completed item or activity.

The graded quality assurance approach set forth here provides flexibility in the selection of the level of the quality program to be applied to an item or activity that is commensurate with the relative importance of the role or function assigned to the item or activity.

The requirements of this subsection are applicable to all items and activities required during geologic repository site characterization, facility and equipment construction, facility operation, performance confirmation, permanent closure, decommissioning, and dismantling of surface facilities.

The three quality assurance levels are defined as follows.

Quality Level 1 — The highest quality level available for assignment on the BWIP. This level is assigned to Q-List items or activities and requires a comprehensive quality assurance program for compliance with applicable requirements.

Quality Level 2 — The intermediate quality level available for assignment on the BWIP. This level is assigned to items and activities with importance to DOE mission objectives and requires a quality assurance program for compliance with applicable requirements that are less extensive than for Quality Level 1.

Quality Level 3 — The lowest quality level available for assignment on the BWIP. This level is assigned to all items and activities included in the quality assurance program but not assigned Quality Levels 1 or 2. It requires good management, engineering, or laboratory work practices for compliance with quality assurance requirements.

A determination will be made if the item or activity is important to safety or waste isolation, in accordance with the methodology described in section 2.0, and is to be placed on the Q-List for the project. If included on the Q-List, the item or activity will be assigned a Quality Level 1 program.

The appropriate quality level for any item or activity shall be determined by the application of the decision criteria in Attachment A. The basis for the selection of the quality level shall be documented.

Once a quality level is selected, the quality program criteria/requirements of Attachment B shall be applied. However, further grading beyond the selection of the quality level may be undertaken in accordance with approved procedures to select the specific criteria/requirements to be applied. This shall be accomplished by an evaluation by technical and quality system personnel of the scope and type of work involved and other factors, as appropriate, that may influence the selection of those criteria/requirements that are necessary and sufficient. The Q-List items and activities shall meet the applicable requirements of 10 CFR 50, Appendix B, and conditions of the NRC QA Review Plan. The scope of work involved in completing an item or activity may be divided into subelements and the criteria/requirements contained in Attachment B evaluated for application to these subelements.

For example, one Quality Level 1 safety-related (Q-List) item may involve an engineered piece of equipment that is very complex to design and manufacture, which calls for special design controls, verification, and development tests in addition

to special controls during manufacture. Thus, it may be subject to all the requirements, supplements, appendices, and other requirements set forth in Attachment B for Quality Level 1. On the other hand, another Quality Level 1 item may actually be a commercial off-the-shelf item that has a proven design, is easy to build, has a good quality history, and is well within the state-of-the-art. The appropriate quality program requirements for this second example may be less than for the first example involving the newly engineered piece of equipment, and several of the criteria/requirements listed in Attachment B may not be required to ensure that appropriate quality is achieved.

To the maximum extent possible, quality program requirements should be tailor-made for individual items, activities, and contracts; however, written justification is required for any deviation from the requirements specified in Attachment B. The term "deviation" as used here means the deletion, addition, or modification of any requirement listed in Attachment B.

The depth of coverage and comprehensiveness of any given quality program criteria/requirement within a quality level may be additionally increased, decreased, or modified as deemed necessary for each item or activity. The technical and quality assurance system personnel shall evaluate each item and activity to determine the depth and comprehensiveness of coverage within each applicable criterion/requirement that is appropriate for that item or activity. Factors to be considered in making this determination include complexity of design or fabrication, uniqueness of the item or activity, the need for controls over special processes or tests, ability to demonstrate functional compliance by inspection or test, and the quality history of the item or activity. Appendix 4A-1 of NQA-1 may also be used as guidance in applying graded quality assurance.

For example, NQA-1 Basic Requirement 10, Supplement 10S-1, and Appendix B criterion 10 may all apply to sealing a repository shaft and welding a shaft liner. Both processes require inspections to verify conformance with design requirements. As it may be difficult to verify that the shaft sealing has been properly performed after placement, continuous surveillance may be appropriate. Conversely, welding is normally verified after completion, and only normal examinations and inspections of completed weldments may be necessary.

Written justification shall be provided for cases where deviations are made from NQA-1 basic requirements, supplementary requirements, appendices, and/or quality assurance criteria of 10 CFR 50, Appendix B, or other requirements, specified in Attachment B as being necessary and sufficient for a certain quality level. Deviations are defined as additions of specified requirements, deletion of specified requirements, or modifications to specified requirements. The written justification for additions is necessary to support and explain the basis for the additional quality assurance requirements and thus justify the corresponding additional cost and effort.

For special items and activities (e.g., potential Q-List items and activities for which DOE has specified quality assurance requirements), the written justification may consist of a reference to the DOE direction.

The quality levels, presented in descending order, have decreasing scope of quality program criteria/requirements. This is evident in the matrix comparison of quality levels shown in Attachment B.

A description of each quality level and guidance for application of each level follows.

Quality Level 1

- Description

  This is the highest quality level available and requires the responsible organization to implement a quality assurance program meeting, as a minimum, all applicable requirements of the Basalt Quality Assurance Requirements Document (BQARD). Quality Level 1 programs require quality planning; preparation of a quality assurance manual/plan and supporting administrative and technical procedures; adherence to procedures and drawings; personnel qualification and training programs; documentation of activities performed and results obtained; and comprehensive review, inspection, management assessment, verification, surveillance, and auditing activities.

  Quality Level 1 programs shall meet the criteria/requirements listed in Attachment B for Quality Level 1 as a minimum, unless appropriate written justification for any deviation is provided.

- Application

  Quality Level 1 shall be applied to all items that have been identified as important to safety or waste isolation (Q-List items). Activities to be considered for inclusion under Quality Level 1 include site selecting, designing, fabricating, purchasing, handling, shipping, storing, cleaning, erecting, installing, emplacing, inspecting, testing, operating, maintaining, monitoring, repairing, modifying, decommissioning, and site characterization (e.g., drilling, excavation and testing, control of software development, validation and verification of software, software configuration management, control of data collection, and calibration of field and laboratory instruments).

Quality Level 2

- Description

  This is the second highest level available for assignment to items and activities on geologic repository projects. Responsible organizations are required to implement quality assurance programs. A quality assurance manual/plan and supporting procedures are required. The same basic NQA-1 quality assurance requirements that apply to Quality Level 1 also apply to

Quality Level 2. However, 10 CFR 50, Appendix B and the NRC QA Review Plan do not apply to Quality Level 2. Fewer NQA-1 supplemental requirements apply to Level 2 with corresponding reductions in quality assurance controls. Quality Level 2 programs shall meet the criteria/requirements listed in Attachment B for Quality Level 2, as a minimum, unless appropriate written justification for any deviation is provided. Other specific requirements that are unique to the item or activity may be specified by the organization responsible for quality level selection.

- Application

Quality Level 2 shall be applied to those items or activities that are not Q-list items but that are of major importance to the attainment of DOE programmatic objectives. Quality Level 2 is also to be applied to items and activities that have potential impact on public and occupational radiological health and safety under 10 CFR 20, and to items involving a significant number of field and laboratory investigations, and complex manufacturing, assembly, and construction processes.

Quality Level 3

- Description

This is the lowest quality level available for assignment and does not require the responsible organization to implement a formal quality assurance program. However, Quality Level 3 items and activities may be required to meet appropriate quality and administrative requirements as determined on a case-by-case basis. The quality requirements to be met for each item or activity, including any required documentation, shall be identified and justified. Quality Level 3 items and activities generally require the use of good management, engineering, or laboratory work practices to prepare them for their intended use.

- Application

Quality Level 3 shall be applied to those items and activities that do not meet the criteria for Quality Assurance Level 1 or 2. This quality level shall be applied to items that can be inspected for acceptance on completion or delivery, or to activities that can be accepted by evaluation of a final report. The quality requirements of subpart 46.202-1 of the Federal Acquisition Regulations, which requires that the contractor perform an inspection, are applicable to Level 3 activities. When deemed appropriate, the requirement to obtain a "Certificate of Conformance" from the supplier may be invoked by the organization responsible for quality level selection.

Typical items and activities that shall be covered by this quality level include the following:

a. Items that are noncomplex and are normally considered commercially available standard hardware

b. Activities that are routine or purely developmental in nature and will not produce data or results that will be used for design, environmental, or licensing applications.

The Rockwell BWIP organization shall develop a procedure for the application of graded quality assurance. The procedure shall be in consonance with the quality assurance program requirements specified herein and shall be submitted to the U.S. Department Of Energy-Richland Operations Office for approval.

ATTACHMENT A

Decision Criteria for Determining Quality Levels of Items and Activities. (sheet 1 of 4)

Categories of Statements of Work	Quality level		
	1	2	3
1. Items (Hardware)			
A. Public Health and Safety Considerations			
• Is the item on the Q-List?	X		
• Is the item intended to control radiation exposure or release levels and/or effluent radioactivity within the limits prescribed by 10 CFR 20?		X	
B. DOE Programmatic Objectives Considerations			
• Failure or malfunction of the item could cause the following potential impact on DOE mission objectives:			
- Impact on cost or schedule impact $\geq$10,000 K		X	
- Impact on cost or schedule impact $\leq$10,000 K.			X
C. Worker Health and Safety Considerations			
• Failure or malfunction of the item could have potential impact on the radiological or nonradiological health and safety of the workers.		X	
D. Lead Time and Cost Considerations			
• Does procurement of the item involve long lead time and/or is the item extremely costly?		X	
E. ASME-BPVC Applicability Considerations			
• Section III applies	X		
• Section VIII applies.		X	
2. Activities			
A. Computer Software Modeling/Development			
1. Are the computer models used to support an item on the Q-list?	X		
2. Do the computer models and codes supply data to support a licensing decision such as performance assessment?	X		
3. Are the computer models complex, and are peer or technical reviews required?		X	
4. Does the work support critical DOE mission documents?		X	
5. If the collected data or records were lost/discarded or of indeterminate quality, the following would occur:			
- The quality of a Q-Listed item or activity would be indeterminate	X		

ATTACHMENT A

Decision Criteria for Determining Quality Levels of Items and Activities. (sheet 2 of 4)

Categories of Statements of Work	Quality level		
	1	2	3
- Repetition resulting in cost impact or schedule impact >10,000 K		X	
- Repetition would result in a cost impact or schedule impact ≤10,000 K.			X
6. Is the computer program only utilized for such tasks as data sorting and collation?			X
B. Field Testing, Data Acquisition, Data Analysis, and Reports			
1. Are the data utilized to support an engineering design criterion for a Q-list item?	X		
2. Will the data support a major licensing document?	X		
3. Will the data become part of the technical data base needed to support licensing?	X		
4. Does the work provide input to critical DOE mission documents?		X	
5. If the collected data or records were lost/discarded or of indeterminate quality, the following would occur:			
- The quality of a Q-Listed item or activity would be indeterminate	X		
- Repetition resulting in cost impact or schedule impact >10,000		X	
- Repetition resulting in cost impact or schedule impact ≤10,000 K.			X
C. Storage of Records/Samples			
1. Do records/samples support licensing activities?	X		
2. Do records/samples support items on the Q-list?	X		
3. Do records/samples support critical DOE mission documents?		X	
4. If the collected data or records/samples were lost/discarded or of indeterminate quality, the following would occur:			
- The quality of a Q-Listed item or activity would be indeterminate	X		
- Repetition resulting in cost impact or schedule impact >10,000 K		X	
- Repetition with cost impact or schedule impact ≤10,000 K.			X
D. Historical or Background Studies and Reports			
1. Will the information produced be utilized in a licensing document?	X		

ATTACHMENT A

Decision Criteria for Determining Quality Levels of Items and Activities. (sheet 3 of 4)

Categories of Statements of Work	Quality level		
	1	2	3
2. Do the studies support a computer model or design criterion for a Q-list item?	X		
3. Does the work support critical DOE mission documents?		X	
4. If the collected data or records were lost/discarded or of indeterminate quality, the following would occur:			
- The quality of a Q-Listed item or activity would be indeterminate	X		
- Repetition resulting in cost impact or schedule impact >10,000 K		X	
- Repetition with cost impact or schedule impact ≤100,000 K.			X
E. Environmental/Socioeconomic Studies and Reports			
1. Do the reports or studies provide critical information to support requirements of the Nuclear Waste Policy Act of 1982?	X		
2. Will the reports or studies be used for major portions of a licensing document?	X		
3. Does the work support DOE mission documents?		X	
4. If the collected data or records were lost/discarded or of indeterminate quality, the following would occur:			
- The quality of a Q-Listed item or activity would be indeterminate	X		
- Repetition resulting in cost impact or schedule impact >10,000 K		X	
- Repetition with cost impact or schedule impact ≤10,000 K.			X
F. Laboratory Experimental (Scoping) or Testing/Analysis and Reports			
1. Will the data results be utilized to support licensing activities?	X		
2. Does the experimental testing provide analytical data to support functional design bases?	X		
3. If the collected data or records were lost/discarded or of indeterminate quality, the following would occur:			
- The quality of a Q-Listed item or activity would be indeterminate	X		
- Repetition resulting in cost impact or schedule impact >10,000 K		X	

ATTACHMENT A

Decision Criteria for Determining Quality Levels of Items and Activities. (sheet 4 of 4)

Categories of Statements of Work	Quality level		
	1	2	3
- Repetition with cost impact or schedule impact ≤10,000 K.			X
G. Construction/Manufacturing Activities			
1. Is the construction/manufacturing activity supporting a Q-list structure, system, or component?	X		
2. Is the activity intended to control radiation exposure or release and/or effluent radioactivity within the limits prescribed in 10 CFR 20?		X	
3. Is the construction/manufacturing activity supporting a highly critical item with a high cost of repair or replacement?		X	
4. Is the system important for reliability?		X	

ATTACHMENT B

Graded Quality Program Requirements Matrix. (sheet 1 of 4)

Quality program requirements	Quality level			Quality program requirements	Quality level		
	1	2	3		1	2	3
NQA-1 BASIC REQUIREMENTS			*	NQA-1 SUPPLEMENTS			
1. Organization	X	X	–	S-1 Terms and Definitions	X	X	–
2. Quality Assurance Program	X	X	–	1S-1 Organization	X	–	–
3. Design Control	X	X	–	2S-1 Qualification of Inspection and Test Personnel	X	–	–
4. Procurement Document Control	X	X	–	2S-2 Qualification of Nondestructive Examination Personnel	X	–	–
5. Instructions, Procedures, and Drawings	X	X	–	2S-3 Qualification of Quality Assurance Program Audit Personnel	X	X	–
6. Document Control	X	X	–	3S-1 Design Control	X	X	–
7. Control of Purchased Items and Services	X	X	–	4S-1 Procurement Document Control	X	–	–
8. Identification and Control of Items	X	X	–	6S-1 Document Control	X	–	–
9. Control of Processes	X	X	–	7S-1 Control of Purchased Items and Services	X	X	–
10. Inspection	X	X	–	8S-1 Identification and Control of Items	X	–	–
11. Test Control	X	X	–	9S-1 Control of Processes	X	–	–
12. Control of Measuring and Test Equipment	X	X	–	10S-1 Inspection	X	X	–
13. Handling, Storage, and Shipping	X	X	–	11S-1 Test Control	X	–	–
14. Inspection, Test, and Operating Status	X	X	–	12S-1 Control of Measuring and Test Equipment	X	–	–
15. Control of Nonconforming Items	X	X	–	13S-1 Handling, Storage, and Shipping	X	–	–
16. Corrective Action	X	X	–	15S-1 Control of Nonconforming Items	X	–	–
17. Quality Assurance Records	X	X	–	17S-1 Quality Assurance Records	X	X	–
18. Audits	X	X	–	18S-1 Audits	X	X	–
10 CFR 50 APPENDIX B 18 QA CRITERIA	X						
NRC QA REVIEW PLAN	X						

*Quality Program Requirements for Level 3 shall be developed on a case-by-case basis.

ATTACHMENT B

Graded Quality Program Requirements Matrix. (sheet 2 of 4)

Quality program requirements	Quality level		
	1	2	3
NQA-1 APPENDICES			
1A-1 Organization	–	–	–
2A-1 Qualification of Inspection and Test Personnel	X	–	–
2A-2 Quality Assurance Programs	–	–	–
2A-3 Education and Experience of Lead Auditors	–	–	–
3A-1 Design Control	–	–	–
4A-1 Procurement Document Control	–	–	–
7A-1 Control of Purchased Items and Services	–	–	–
17A-1 Quality Assurance Records	–	–	–
18A-1 Audits	–	–	–
OGR QA PLAN SUPPLEMENTS			
S-1 Qualification of Personnel Performing and Verifying Activities Affecting Quality	X	X	–
S-2 Overview of Quality Assurance Activities	X	X	–
S-3 Q-List Methodology	X	X	–
S-4 Quality Assurance Records	X	X	–
S-5 Quality Assurance for (R&D) Experiments	X	X	–
S-6 (Reserved)			
S-7 Peer Review	X	X	–
S-8 Graded Quality Assurance	X	X	–
S-9 Reliability of Data	X	X	–
S-10 (Reserved for Waste Form)			

ATTACHMENT B

Other Requirements Descriptions. (sheet 3 of 4)

Reference	Requirement description
NRC Review Plan Para. 2.5	1. Activity Planning--This applies to the application of graded approach of the QA program to the complexity commensurate with the importance of the work activities. Plans shall be developed and documented to describe how the activities shall be performed, the results expected, the major milestones, and other procedures to be used:
NRC Review Plan Para. 2.7	2. Management Assessment--This requirement applies to performance of annual management assessment to determine the scope, status, adequacy, and effectiveness of the QA program.
NRC Review Plan Para. 2.8 DOE/OGR Quality Program Requirements	3. Personnel Qualification and Certification--This requires that personnel who perform quality-related activities be properly trained, indoctrinated, and qualified. Personnel shall receive training in technical and quality assurance procedures. Management is required to monitor the performance of individuals involved in activities affecting quality and determine the need for retraining and/or replacement.
NRC Review Plan Para. 3.8	4. Technical and Peer Reviews--These are the requirements that define technical and peer reviews, and when they should be accomplished.
NRC Review Plan Para. 15.4 Para. 18.4	5. Trend Analysis--This requires that nonconformance reports be periodically analyzed to indicate quality trends and to help identify root causes of nonconformances. Results are to be reported to upper management for review and assessment.
DOE Order 5000.3	6. Unusual Occurrence Reporting--Contractors are required to report any significant event that results in any deviation from the planned or expected behavior of an activity or operation or course of events that has or could have significant programmatic (reliability, cost, or schedule), safety, health, or environmental impacts. Significant events are to be reported in accordance with DOE Order 5000.3.
NRC Review Plan Para. 2.2 DOE/OGR Quality Program Requirements	7. Software Control--These are detailed computer software quality assurance requirements that include validation, verification, code custodial and transfer requirements, and conformance with NUREG-0856.
NRC Review Plan Para. 8.3 Para. 13.1, 13.2	8. Sample Handling--This requires that samples of geological media (e.g., rock, core, soil) be shipped, handled, and stored in accordance with special procedures that describe the control of the activities related to handling of samples.
DOE/OGR Quality Program Requirements	9. Data Acquisition, Verification, and Analysis--This requires that data and information that is received from literature searches be validated from the technical adequacy perspective and for correct application.
DOE/OGR Quality Program Requirements	10. Reporting and Submittals--These are requirements that identify the types of reports and the frequency of submittals. Examples: Audit Reports and Schedule Revisions per OGR-B-3 -- each report, and each audit schedule revision Surveillance Reports -- each report Summary Reports -- quarterly (per OGR/B-3) Unusual Occurrence Reports -- Records Management Plan --

ATTACHMENT B

Other Requirements Descriptions. (sheet 4 of 4)

Reference	Requirement description
NRC Review Plan Para. 3.10	11. Configuration Control--A configuration system shall be established at the earliest practical time to ensure that design changes are analyzed and properly identified and documented.
DOE/OGR Quality Program Requirements	12. Records Management--Records shall be stored, identified, and reviewed as described in a Records Management Plan.

Quality program requirements	Quality level		
	1	2	3
Other requirements			
Activity planning	X		
Management assessment	X		
Personnel qualification and certification	X		
Technical and peer reviews	X		
Trend analysis	X		
Unusual occurrence reporting	X	X	
Software control	X		
Sample handling	X		
Configuration control	X		
Reporting and submittals	X	X	

Rockwell Hanford Operations
P.O. Box 800
Richland, WA 99352

Rockwell International

April 28, 1987

In reply, refer to letter R87-1832

Mr. J. H. Anttonen, Assistant Manager
Commercial Nuclear Waste
Department of Energy
Richland Operations Office
Richland, Washington 99352

Dear Mr. Anttonen:

TRANSMITTAL OF THE BASALT REPOSITORY PRELIMINARY Q-LIST FOR DEPARTMENT OF ENERGY-RICHLAND OPERATIONS OFFICE INTERNAL REVIEW
(Contract DE-AC06-77RL01030)

Reference: (a) Report, April 1986, Roy F. Weston, Inc., to the Office of Civilian Radioactive Waste Management, U.S. Department of Energy, Guidance for Developing the SCP-CDR and SCP Q-Lists

(b) Letter, March 26, 1987, Ralph Stein, Director, Engineering and Geotechnology, Office of Civilian Radioactive Waste Management, to J. H. Anttonen, J. Neff, D. Veith, Responses to July 1986 SCP Q-List Methodology Workshops

We are transmitting for your internal review six copies of the draft Basalt Repository Preliminary Q-List. This document was developed at the direction of Department of Energy-Richland Operations Office and following guidelines set forth in Reference (a). Additional guidance was provided by personnel from Weston and Department of Energy-Headquarters at a meeting here on July 17-18, 1986. The subject document will be issued as a Basalt Waste Isolation Project (BWIP) Supporting Document, SD-BWI-TA-025, after the remaining requirements of the project procedure controlling the issuance of supporting documents are satisfied. In particular, by procedure, a supporting document can only cite released documents; the subject document cites the Conceptual Design Report (Study 10) and is, itself, cited by Study 10. This implies that the two reports must be released at the same time.

Recently, reference (b) transmitted responses from Weston to questions raised by the BWIP in the July 1986 meeting cited above. The reference letter and the attached responses suggest changes in the guidelines for determination of the repository Q-List. The suggested changes in the guidelines do not significantly change the quality assurance requirements imposed on repository items and activities, but they will require significant revision of the Basalt Repository Preliminary Q-List document and the interfacing sections of the Site Characterization Report and the

Rockwell International

Conceptual Design Report. Upon receipt of direction from your office to implement the suggested changes, we will proceed with the required revision of the three documents.

If you have any questions, please call K. R. Fecht of my staff on 376-6558.

Very truly yours,

D. C. Gibbs, Director
Basalt Waste Isolation Project

DCG/JAT/JSD/dac

Att.

cc: J. J. Keating - DOE-RL
A. W. Kellogg - DOE-RL
R. J. Light - DOE-RL

CONTENTS

Figures: (cont.)

Tables:

1.0 EXECUTIVE SUMMARY

The concept of a Q-List, as it applies to nuclear waste repositories, is currently defined by the U.S. Department of Energy (DOE), Office of Civilian Radioactive Waste Management (OCRWM) to include the following:

- Those repository systems and components that are important to safety (as defined in 10 CFR 60 (NRC 1983) to mean important to public radiological safety)
- Those repository systems and components that are important to maintaining the possibility and practicality of retrieving part or all of the emplaced wastes
- Those repository systems or components and those natural barriers that are important to waste isolation
- Those site characterization program activities that have the potential to degrade long-term waste isolation or have the purpose of improving understanding of the repository and the ability to predict its long-term waste isolation performance.

A process using probabilistic risk assessment techniques was used to determine the Basalt Repository Important-to-Safety Q-List. This process was applied qualitatively to repository surface facilities and quantitatively to the subsurface facilities. In many cases, the conceptual status of repository design or understanding of physical processes relevant to repository accidents did not permit the assignment of well-documented values for accident initiator probabilities, equipment reliabilities, or accident source terms. In those cases, literature review and engineering judgment were used to establish estimates. Probabilities, source terms, and accident consequences will be defined more precisely as the Repository design progresses, and safety analyses are performed.

A summary of Basalt Repository Q-List items is provided in table 1.

1.1 ITEMS IMPORTANT TO SAFETY

The methodology used to determine those items important to safety involves the screening of potential accident initiators with respect to the credibility of their occurrence and the likelihood of significant impact on the public. For credible accident initiators, a detailed estimate is made of the subsurface and the surface radionuclide releases, assuming no beneficial mitigation of those releases by repository systems. The surface releases are then analyzed to determine radiological consequences using the standard radiological dose assessment codes used at the Hanford Site. If the radiological exposure to the "maximally exposed individual" standing at the nearest point of the "unrestricted area" is less than 100 mrem (based on the 10 CFR 60 (NRC 1983) limit of 500 mrem with an additional factor of conservatism requested by the OCRWM methodology), then the methodology

requires that that initiator be discarded. The nearest point of the unrestricted area for the repository is approximately 1.6 km (1 mi) from the repository surface facilities along Route 240. If the radiological consequences are greater than 100 mrem, event trees are constructed that represent the success or failure of repository systems with the potential for preventing or mitigating accident sequences (i.e., event tree branches) arising from the accident initiator.

For each event tree branch, the accident sequence must eventually be discarded either on the basis of successful prevention or mitigation of the sequence or the basis of noncredibility (i.e., the event tree branch probability falling below a "threshold" probability) for sequences involving failure of the preventive and/or mitigative systems. Finally, those repository systems or components whose successful functioning enabled the discarding of the accident sequences on the basis of prevention of the sequence or mitigation of its radiological consequences are put on the Important-to-Safety Q-List.

1.1.1 Surface Facilities

The Surface Facilities Important to Safety Q-List was determined using a methodology similar to the methodology used for the Subsurface Facilities Important-to-Safety Q-List. However, since there is more lead time before the start of construction of surface facilities and because of the conceptual status of the surface facility design, the analysis used to determine Q-List items was conducted qualitatively rather than quantitatively. The methodology used is described in more detail in sections 3.3 and 4.2.

Because the flow paths for radionuclides moving from the surface facilities to the site boundary are shorter and more direct than the pathways for the subsurface facilities, relatively more items appear on the Surface Facility Important-to-Safety Q-List than on the Subsurface Facility Important-to-Safety Q-List.

The determination of the Surface Facility Important-to-Safety Q-List is described in chapter 4 of this report. The Surface Facility Important-to-Safety Q-List appears in section 4.7 and is summarized in table 1.

1.1.2 Subsurface Facilities

The radioactive waste in the subsurface facility is always enclosed in a substantial waste package and separated from the site boundary by a lengthy and indirect transport path. Because of this, only a few of the potential subsurface accident initiators involve enough energy in a form that is sufficiently concentrated to overcome the natural and engineered barriers. Hence, the Subsurface Facility Important-to-Safety Q-List contains few entries.

The subsurface explosives magazines and the subsurface diesel fuel storage tank (together with that part of the fire suppression system protecting the diesel storage tank) are Q-Listed, as are the waste hoist facilities. These are engineered systems for which proper design and administrative control will act to prevent the occurrence of significant accident initiators.

The repository ventilation filtration system is necessary to mitigate the consequences of a hypothetical waste-handling shaft cask drop accident and hence is Q-Listed. Those portions of repository utility systems needed to detect airborne radioactivity and bring the confinement exhaust filtration on line when needed are also Q-Listed. Similarly, the systems for detecting and removing radionuclides from the dewatering system flow to the surface retention/percolation ponds need to be Q-Listed.

Details of the determination of the Subsurface Facilities Important-to-Safety Q-List are given in chapter 5, and the Q-List itself appears in section 5.5 and in table 1.

1.2 ITEMS IMPORTANT TO RETRIEVABILITY

The Important-to-Retrievability Q-List is still being developed. A joint Basalt Waste Isolation Project (BWIP) and Kaiser Engineers, Inc./Parsons Brinckerhoff Quade & Douglas, Inc. (KE/PB) team is studying the technical issues associated with retrievability. The Q-List will be updated to account for insights gained by the Retrievability Team.

The only phenomenon under design control that is recognized as having a potential for making local or global retrieval impractical is a systematic, early failure of the waste package leading to significant contamination of emplacement boreholes and drifts. Because of this, the only items that have been placed on the Important-to-Retrievability Q-List at this early stage of design are those aspects of the waste package design related to corrosion resistance and weld integrity. In addition, the project has committed to design and fabricate any equipment needed for abnormal retrieval (retrieval required because of waste package failure and release of radionuclides) as Q-Listed items.

1.3 ITEMS IMPORTANT TO WASTE ISOLATION

The OCRWM methodology describes two criteria for assignment to the Important-to-Waste Isolation Q-List. First, an item might be an engineered or natural barrier to radionuclide transport for which credit is being taken (or might be taken in the future) in the assessment of repository waste isolation performance. Second, an item might be a site characterization program activity that either has the potential for compromising one of the engineered or natural barriers meeting the first criterion or a site characterization program "activity" with the purpose of improving our understanding of the engineered and natural barriers or improving our ability to predict the long-term waste isolation performance of the

repository. This definition requires the Q-Listing of most site characterization program activities.

Chapter 7 lists the Q-Listed barriers and contains a more detailed definition of which site characterization program activities need to be Q-Listed and a discussion of the some of the implications of Q-Listing site characterization program activities.

Table 1. Summary of Basalt Repository Q-List Items. (sheet 1 of 3)

I. Surface Facility Items Important-to-Safety

A. High-level waste containment

- Shipping casks for waste materials (this is an important repository interface with external organizations that needs to be carefully controlled)
- Waste containers
- Underground transfer casks

B. Radioactive material confinement

- Primary confinement barrier
- Zone I ventilation
- Zone II ventilation

C. Handling and storage

- Criticality control systems

D. Auxiliary Systems

- Hot cell fire suppression system
- Emergency power system (provides power for monitoring, control, and confinement)
- Uninterruptible power system (provides power for monitoring and control functions)

E. Monitoring and control systems

- Effluent radiation monitoring
- Confinement function monitoring and control
- Site control room(s) (those portions that control Q-Listed functions)

F. Structures

- Waste handling building (those portions of the building that comprise the primary confinement barrier, together with building structures whose failure (e.g., during an earthquake could endanger other Q-Listed items, including Q-Listed control room functions)
- Emergency generator buildings
- Hot cell transfer port
- Missile shielding (may or may not be Q-Listed, depending on the function of the protected equipment)

Table 1. Summary of Basalt Repository Q-List Items. (sheet 2 of 3)

II. Subsurface Facility Items Important-to-Safety

A. Confinement exhaust filtration subsystem of the ventilation system

- Prevention of the creation of an unfiltered bypass airflow pathway
- High-efficiency particulate air (HEPA) filter elements
- Subsurface radiation detectors and the repository utilities and control systems that support those detectors
- Actuation systems responsible for bringing confinement exhaust filtration on-line when an actuation signal is received from the subsurface radiation detectors
- Ductwork leading to the filters, filter buildings, and airlocks
- Confinement exhaust fans and the rest of the subsurface ventilation system, depending on details of the design (i.e., can failures lead to creation of an unfiltered bypass flow)

B. Dewatering system subsystems

- Radiation detectors and associated controls and repository utilities
- Radionuclide removal function on the dewatering system flow

C. Engineered systems whose failure could cause a significant accident

- Waste handling hoist system (those functions that serve to prevent a hoist cage drop)
- Subsurface explosives magazines
- Subsurface diesel storage tank and associated fire-suppression system
- Those aspects of the waste package design that prevent nuclear criticality subsequent to waste-handling accidents

III. Important to Maintenance of Retrievability

These items will be better defined when the information being developed by the retrievability study is available. The only items so far identified that seem to be important to maintaining retrievability are:

- those aspects of waste package design that prevent systematic early failures of the waste package due to corrosion or some other mechanism
- design and fabrication of special equipment for "abnormal" retrievability

Table 1. Summary of Basalt Repository Q-List Items. (sheet 3 of 3)

IV. Items and Activities Important to Waste Isolation

A. Engineered barriers

- Waste container
- Waste package packing material
- Shaft and borehole seals
- Damaged rock zone

B. Natural barriers

- Repository isolation zone (the volume of rock and groundwater that will be relied upon to inhibit radionuclide transport to the accessible environment)

C. Site characterization program activities

- Activities that design or produce a Q-Listed item (e.g., an engineered barrier)
- Activities that modify a Q-Listed item (e.g., a natural barrier)
- Data collection and recording activities that are part of the basis for understanding the barriers or confirming their performance
- Activities that serve to define or describe the conceptual and mathematical models of the natural barriers
- The performance assessment process of formally evaluating the engineered barriers design and the natural barrier description, together with the predictions of the performance of those barriers

D. Preclosure events that can impact long-term waste isolation

- Subsurface explosives magazines (because of possibly extensive damage to the repository host rock due to a preclosure period magazine explosion).

2.0 INTRODUCTION

This document is provided as an initial step in developing a Basalt Repository Q-List in a consistent manner with the Site Characterization Plan Conceptual Design Report (CDR) (KE/PB 1987) and the Site Characterization Plan (SCP). The product of this initial step will be called the Basalt Repository Preliminary Q-List or the SCP-Stage Q-List.

A Q-List is a list of those systems, structures, and components (collectively referred to in this report as items) determined to be important to safety and those items and site characterization activities determined to be important to waste isolation. The term "important to safety" is defined explicitly in 10 CFR 60.2 (NRC 1983) and "important to waste isolation" is defined by the context of its use in 10 CFR 60.21. In addition, 10 CFR 60.111(b), 60.133(c), and 60.135 all put requirements on the ability to retrieve waste during the period of emplacement and operation of the repository. These retrievability requirements have led the OCRWM to require that systems and components important to maintaining the capability to retrieve waste also be put on the Q-List.

The items and activities on the Q-List will be subject to a formal quality assurance program based on the requirements of 10 CFR 50, Appendix B (NRC 1979), as invoked by 10 CFR 60, Subpart G.

The SCP-stage Q-List indicates items for which additional attention regarding safety analyses and design criteria must be provided for in later stages of the design process. Early identification of the Q-List is important to streamline quality assurance, design, planning, safety analysis, and licensing efforts and to ensure, to the extent practical, the Q-List is technically consistent with U.S. Nuclear Regulatory Commission (NRC) regulatory requirements and regulatory guidelines (as promulgated in NRC 1986).

The NRC defines, in 10 CFR 60, items important to safety to be 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 to 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 of the repository. For purposes of identifying items for the Q-List, only credible accidents are to be considered. At the SCP stage, due to the limited data base available and the conceptual nature of the repository design, the decision as to which accident initiators are credible has been based on engineering judgment and conservative assumptions.

Items important to waste isolation are those natural and engineered barriers that are relied on to inhibit the transport of radionuclide and that must function in a manner prescribed by federal regulations to meet the long-term waste isolation requirements of 10 CFR 60.112 and 60.113.

In addition to the above regulatory criteria, the following items will also be classified as Q-List items:

- Items whose failure could make retrieval of the emplaced waste packages impossible or impractical

- Items whose failure could have an impact during the preclosure period of the engineered or natural barriers, such that postclosure waste isolation performance is compromised. Further, any specially designed equipment required to enable retrieval of a severely damaged waste package from an underground environment requiring extensive reexcavation and opening stabilization will also be a Q-List item.

All Q-Listed items classified as important to safety will be designed to the general design requirements of 10 CFR 60.131(b). Future safety analyses will assess the need to maintain important-to-safety items on the Q-List. All items identified as important to waste isolation must support the waste isolation performance requirements of 10 CFR 60.112 and 60.113 and must be consistent with the post-closure performance allocations for the repository. Retrieval equipment placed on the Q-List will also be designed to the design requirements of 10 CFR 60.131(b).

The methodology used in preparing this report is based on a paper prepared by Roy F. Weston, Inc., for the OCRWM (Weston 1986). The methodology is intended to provide technically sound and logically traceable Q-List determinations so that the resulting SCP-stage Q-List will be based, insofar as the current design detail permits, on well-defined and documented technical bases, criteria, and assumptions. This methodology was implemented at the BWIP by means of a project directive (Rockwell 1986a). Project Directives are a mechanism for creating, in a controlled fashion, ad hoc procedures for accomplishing tasks not covered by the existing project procedures. In this case, a project directive was used because the effort to develop a preliminary Q-List was undertaken at a time when the whole structure of project procedures was undergoing a significant revision.

The Q-List will be modified, in a controlled fashion, as the design matures through the Advanced Conceptual Design (ACD) and License Application Design (LAD) stages, and as additional data from the site characterization program become available. In particular, the OCRWM Plan for the Advanced Conceptual Design (ACD) of the Repository and the Waste Package (Weston 1987) requires that conceptual design of all Q-Listed structures, systems, and components be completed during the ACD, and that detailed design of these items be completed by the end of the LAD.

Certain aspects of the waste package-related descriptions in this report are different from those presented in the Waste Package SCP-CDR (Rockwell 1987). This is the result of conceptual design efforts that started at different times. Repository and waste package descriptions in this document are based on the Conceptual Design Criteria Document, (Rockwell 1986b). The waste package descriptions in the Waste Package SCP-CDR include certain design concepts that evolved after the issuance of the Conceptual Design Criteria Document and that show merit in meeting the

performance requirements of the waste package. Further tradeoff studies, development tests, and design efforts are planned to confirm these concepts and evaluate other concepts to arrive at better configurations, handling, and performance of the waste package. These future efforts in developing the waste package design, in conjunction with similar efforts to confirm the current repository concept, will result in the establishment of a compatible waste package/repository concept during the ACD. Meanwhile, any inconsistencies noted between the Waste Package SCP-CDR and the Repository SCP-CDR are not expected to significantly affect the repository concept presented in this report, or the items placed on the preliminary Q-List in this report.

It should be noted that several other studies are under way that incorporate more recent design input and establish its impact on the overall repository design. The conceptual features developed during these studies will be considered in the Waste Package and Repository ACDs.

3.0 METHODOLOGY AND BACKGROUND INFORMATION

3.1 Q-LIST PREPARATION TEAM ORGANIZATION

The Basalt Repository Q-List was determined using a methodology (Weston 1986) provided by the U.S. Department of Energy-Headquarters (DOE-HQ) to all three high-level waste repository sites. At the BWIP, a team of approximately 15 members from BWIP Rockwell Hanford Operations (Rockwell), KE/PB, Hanford Engineering Development Laboratory, Pacific Northwest Laboratory, and Boeing Computer Services-Richland was established. Team members average more than 20 yr of nuclear or mining industry experience. A core working group of five, including a mining engineer from KE/PB, coordinated the efforts of seven other working groups and did most of the actual writing of the report. The BWIP Q-List Preparation Organization is shown in figure 1.

3.2 DESIGN BASIS

The CDR (KE/PB 1987) is the basis for the analyses described in this report. A brief description of the design is given here as background information. Assumptions have been made where necessary to conduct a meaningful analysis. These assumptions are listed in section 3.2.2 and described in detail in the appropriate places in this report.

3.2.1 Current Status

3.2.1.1 Operations Overview. Spent fuel will arrive from nuclear fuel storage facilities by truck or rail in shipping casks approved by the U.S. Department of Transportation. The fuel will be unloaded into a process cell, inspected, and either intact spent fuel assemblies or consolidated fuel pins placed in a waste container. A lid will be welded onto the container, and, after careful inspection, the container will be decontaminated and loaded into the transfer cask. A transporter will carry the cask through the transfer tunnel to the waste shaft, where it will be lowered on the shaft conveyance to the underground shaft station. The cask will be loaded on the underground waste transporter and driven to an emplacement location. At that point, the waste container will be transferred from the cask to a borehole that will have been fitted with a steel liner and preformed packing material. A shielding plug will be inserted and a cover plate bolted over the opening.

For retrieval, the emplacement process would be reversed. (Periodic removal of a few waste containers will support waste package performance confirmation studies. General retrieval may also be required under certain circumstances.) After the emplacement boreholes in an emplacement drift have been filled, the airflow will be reduced. Because of the decay heat, the temperature in the drift will rise 70 °C (160 °F). If it is necessary to re-enter the drift for retrieval porposes, 6 mo of air cooling may be required to reduce the temperature to a reasonable working level.

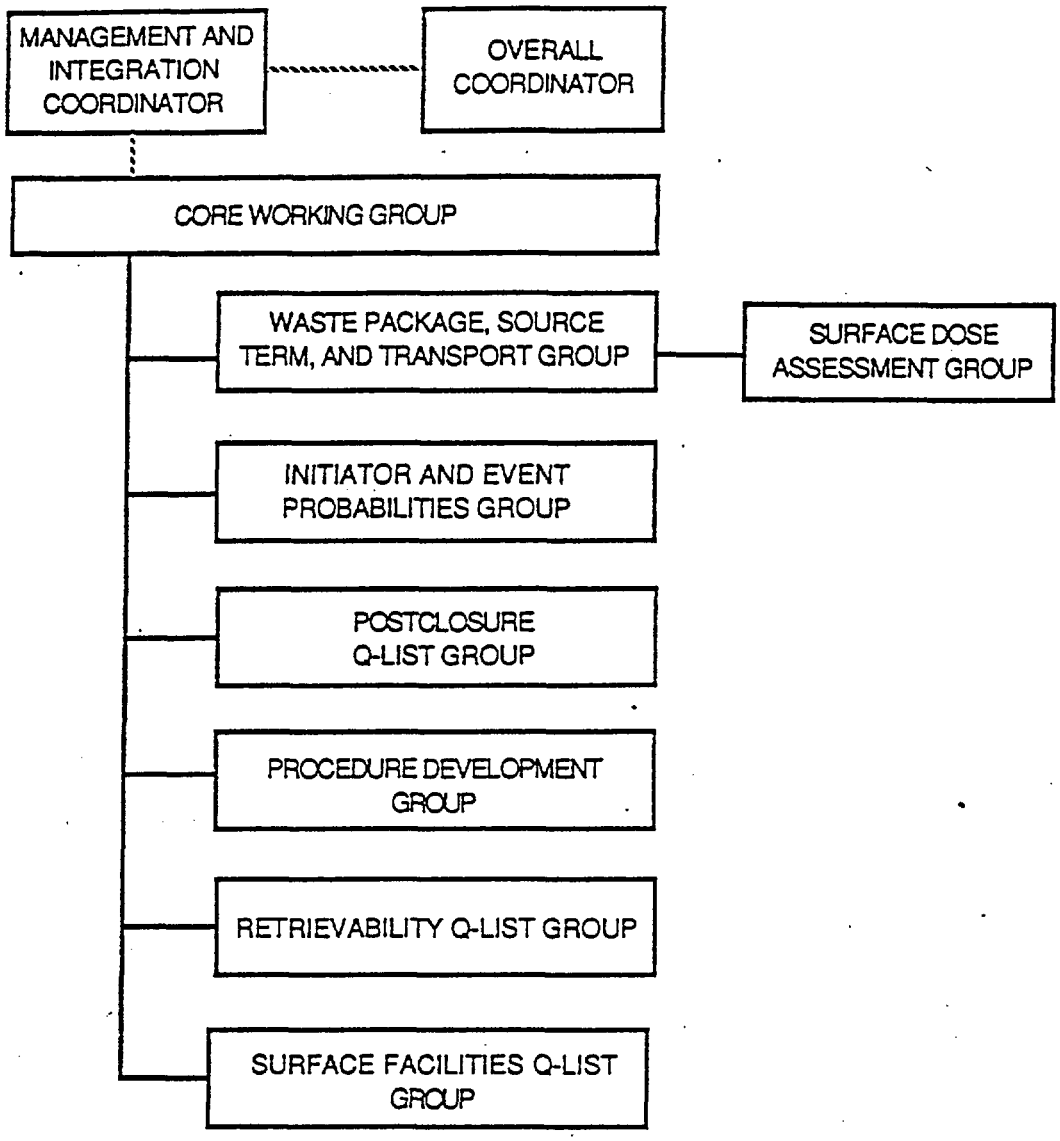


Figure 1. Basalt Waste Isolation Project Q-List Preparation Team Organization.

Upon entry to the room, personnel will check to determine whether unusual conditions (leaks, moisture intrusion, host rock failure, or radioactive contamination) exist. If so, equipment and procedures may have to be developed for abnormal retrieval. If conditions are normal, retrieval will essentially be the reverse of emplacement using the existing equipment. Some modifications of the Waste Handling Building may be required to facilitate retrieval operations and support surface storage and handling.

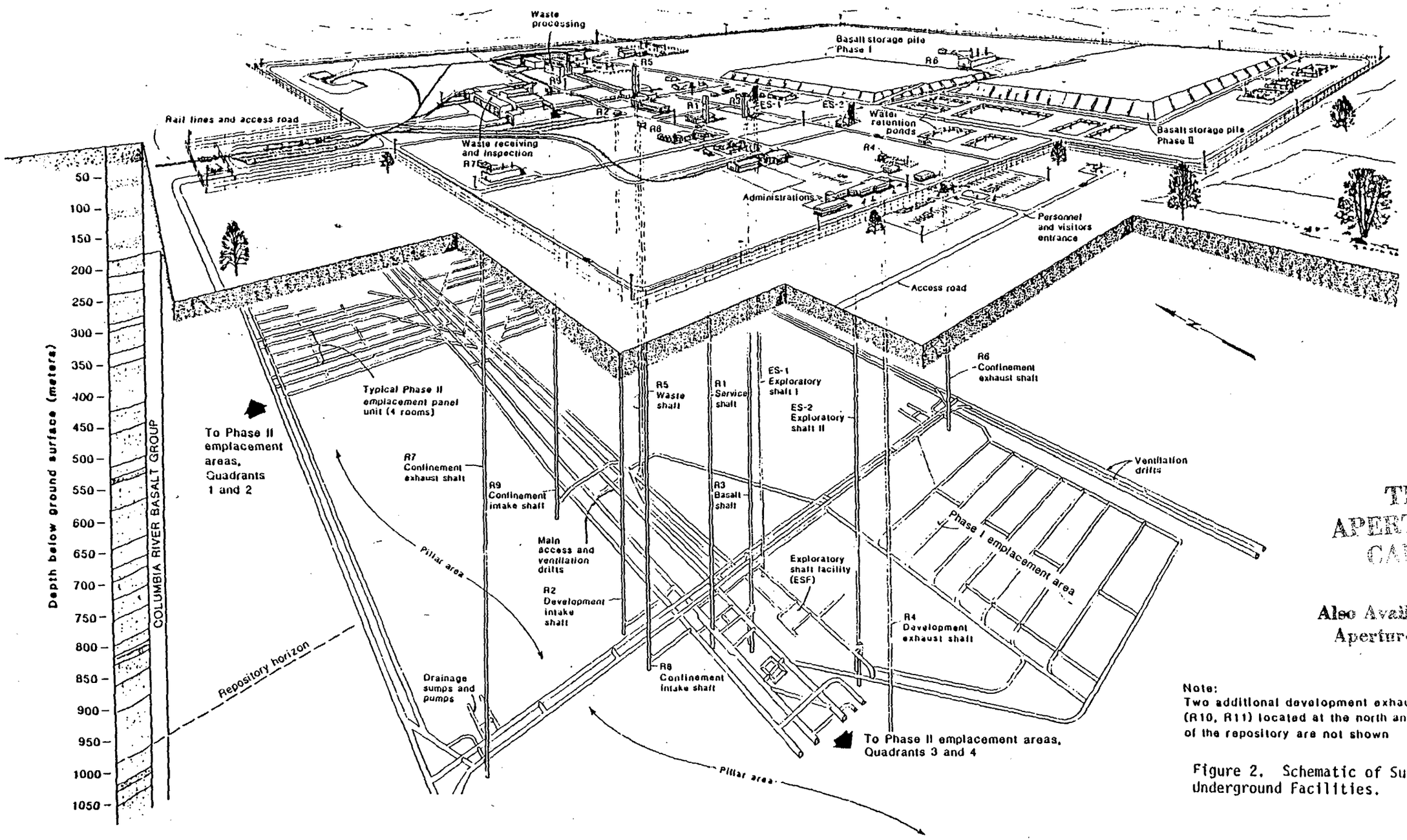
3.2.1.2 General Layout of the Repository. The repository will consist of surface facilities, shafts, and a complex of drifts at a depth of about 975 m (3,200 ft) below the surface (fig. 2). The surface facilities will include those required for processing the spent fuel as well as facilities that support all activities of the repository. There will be 11 shafts. Three of these are equipped with dedicated hoists, one for men and materials, one for mined basalt, and one for the radioactive waste containers. All shafts will provide ventilation, either for air intake or exhaust.

Underground, five main access drifts 5 m (16 ft) wide by 3.6 m (12 ft) high will run north and south of the central pillar region into which the seven of the eleven shafts will penetrate (fig. 3). These drifts will provide access and ventilation to the emplacement rooms, which will be drifts 7 m (23 ft) wide by 4 m (13 ft) high running east and west from the access drifts. In the main development (Phase II), there will be 168 rooms 792 m (2,600 ft) long, each capable of storing about 280 waste containers. The rooms will be 30 m (100 ft) apart. Two parallel drifts will be located around the perimeter and will be connected to each of the rooms for exhaust air flow.

3.2.1.3 Suspect Storage Area. Incoming trucks and rail cars will be inspected for damage, radioactive contamination, and explosive or incendiary devices. An area will be provided for storage of suspect shipments where further inspection and work may be done. This area will be behind an earthen embankment in the northwest corner of the repository area (see fig. 2).

3.2.1.4 Waste Handling Building. A small waste handling building (WHB-1) will be used early in the project. The main facility (WHB-2) will be built later, after which WHB-1 may be used for handling special waste forms. The two facilities will be nearly identical, except for size (see fig. 2). The determination of Surface Facilities Important-to-Safety items was based on the design of WHB-2; however, given the similarity with WHB-1, items placed on the Q-List for WHB-2 are similarly classified for WHB-1.

The main waste handling building, WHB-2, will consist of three floors, all above ground. There will be a series of connected hot cells on the lower two floors, each equipped for its specific task in handling and packaging the waste fuel. Two separate process lines will be included so that operations may continue if one line is down. There will be two rail/truck unloading bays, two receiving cells, four processing cells, two containerization cells, two decontamination cells, two loadout cells, and two transfer cask loadout stations. There will be a single large maintenance cell and supporting laboratories and other facilities.



Note:
Two additional development exhaust shafts (R10, R11) located at the north and south ends of the repository are not shown

Figure 2. Schematic of Surface and Underground Facilities.

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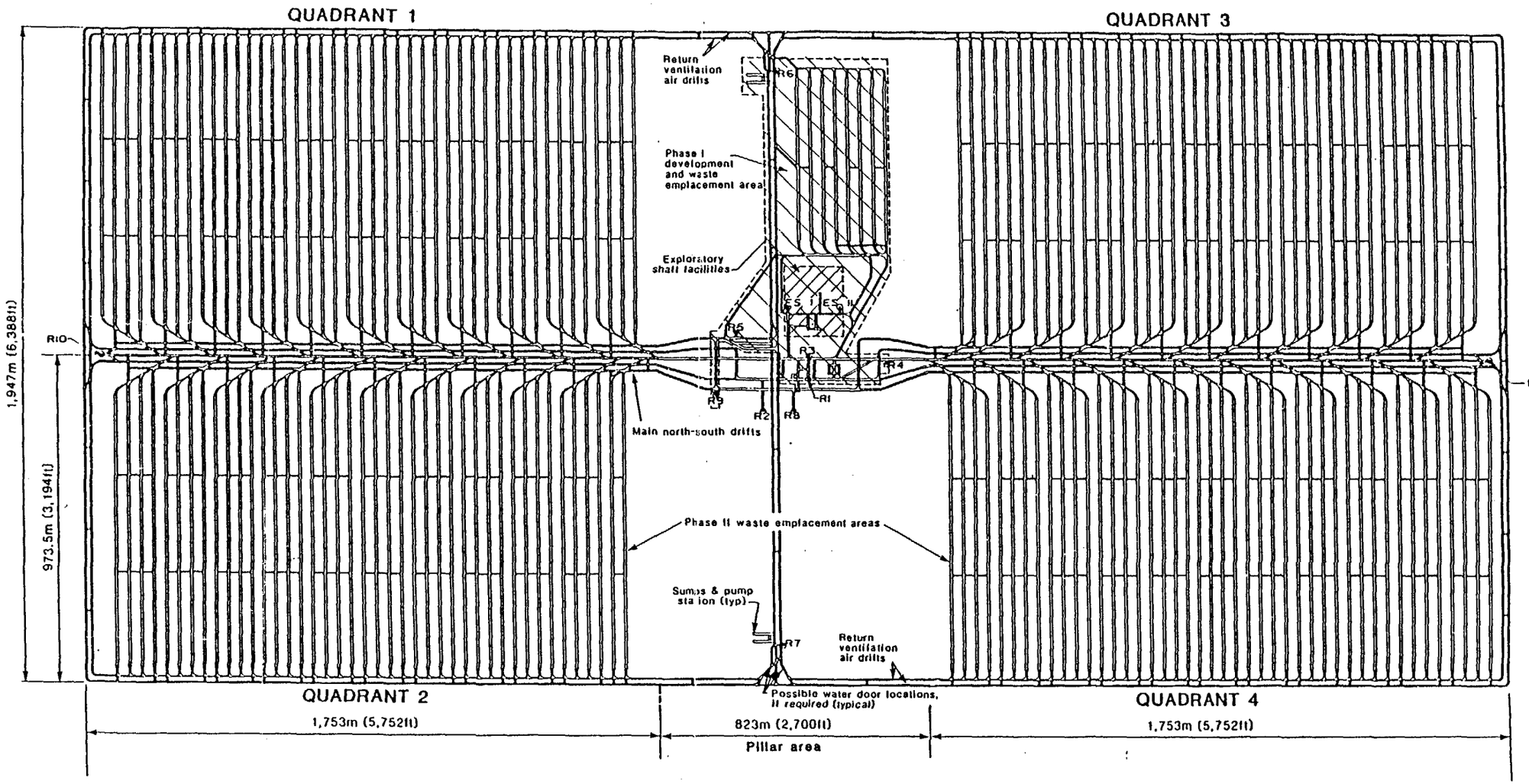
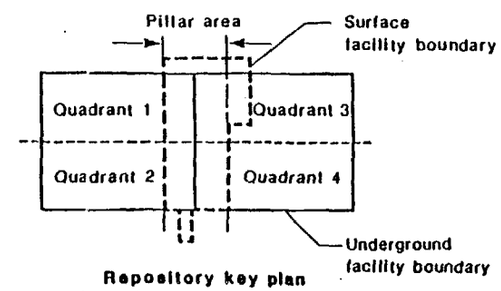


Shaft legend:

R1	Service shaft (development intake), 12' dia.
R2	Development intake shaft, 12' dia.
R3	Basalt-hoisting shaft (development exhaust), 12' dia.
R4	Development exhaust shaft, 12' dia.
R5	Waste-handling shaft, 12' dia.
R6 & R7	Confinement exhaust shaft, 12' dia.
R8 & R9	Confinement intake shaft, 12' dia.
R10 & R11	Development exhaust shaft, 12' dia.
ES I	Exploratory shaft 1, 6' dia.
ES II	Exploratory shaft 2, 6' dia.



Repository key plan
Approximate repository outline
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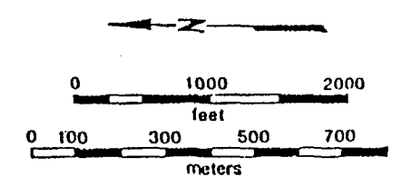

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The building ventilation system will be divided into four zones, depending on the degree of potential radioactive contamination. Zone I will include areas with the highest potential. The air pressure in each zone will be controlled so that any leakage is toward a lower numbered zone. Some Zone III air will pass through high-efficiency particulate air (HEPA) filters to Zone II areas, and some Zone II air similarly to Zone I areas. Exhaust air from Zones I and II will pass through double HEPA filters (50% filter redundancy, 100% fan redundancy). Zone III and IV exhaust will receive single HEPA filtration.

3.2.1.5 Waste Container. The waste container will be made of thick-walled carbon steel. The wall thickness will satisfy structural requirements (in particular, impact resistance) and allow for corrosion, including radiolysis-enhanced corrosion. Waste container dimension will depend upon the specific waste form. An inert atmosphere will be provided after filling with waste. Closure will be effected by welding on a heavy cap.

3.2.1.6 Shafts. The repository will have eleven shafts (plus the two exploratory shafts). Seven of these will be in the central pillar area, two will be at the east and west edges of the pillar area, and two at the north and south ends of the repository. The shaft casings will be made of ring-stiffened steel pipe of graduated thickness with a hemispherical bottom. Sections of this pipe will be welded together as they are inserted into the drill hole. The casing will be grouted in place to complete the shaft liner.

Three of the central shafts will be equipped with hoists: R1, the service shaft, will be used for men and materials; R3 will transport the mined basalt; and R5 will transport the waste containers. The primary function of the other eight shafts is ventilation. The inner diameter of the shafts will be 3.6 m (12 ft). The facilities and other attributes of the shafts are given in table 2.

Table 2. Shaft Facilities. (sheet 1 of 2)

Shaft identifier	Description of facilities
R1 (service shaft)	This shaft will have a tower-mounted Koepe hoist in a concrete structure. It will have loading and unloading facilities at the surface and at the shaft station. At the surface, there will be an air cooler and fan facility to provide inlet air for the development area.
R2	An air cooler and fan facility will be located at the surface to provide inlet air for the development area. A protective structure will cover the head.

Table 2. Shaft Facilities. (sheet 2 of 2)

Shaft identifier	Description of facilities
R3 (basalt handling)	This shaft will be equipped with a ground-mounted Koepe hoist and a steel open headframe. There will be two 9-t (10-ton) skips operating in the balanced mode. In addition, there will be a smaller skip for retrieving spillage from the spillage hopper near the bottom of the shaft. At the shaft station, there will be skip loading facility (e.g., chutes, hopper, weighing device). At the surface, there will be a scroll for dumping the skips and chutes to transport the muck to railcars. Development area air will be exhausted through ducting near the top of the shaft.
R4	Development air will be supplied through ducting near the top of the shaft. A protective structure will cover the head.
R5 (waste handling)	A tower-mounted Koepe hoist in a concrete structure will operate the waste handling cage and counterweight. At the surface and at the shaft station underground, there will be facilities for loading and unloading the waste cask. Confinement air will be exhausted through an on-line HEPA filtration facility using fans and a stack discharge. The airflow through this shaft will be relatively low.
R6	Confinement air will be exhausted through this shaft. An automatic on-demand HEPA filtration facility with fans and stacks will be located adjacent to the surface opening. A protective structure will cover the head.
R7	Same as for R6.
R8	Confinement inlet air will be drawn into this shaft through an air cooler facility near the surface opening. A protective structure will cover the head.
R9	Same as for R8.
R10	Development air will be exhausted through ducting near the top of the shaft. A protective structure will cover the head.
R11	Same as for R10.

3.2.1.7 Waste Hoist (R5). The waste hoist shaft will have a reinforced concrete tower headframe with a tower-mounted friction (Koepe) hoist. The tower will be high enough to accommodate the cage, rope attachments, the deflection sheaves, and overtravel. It will operate in a balanced mode with a counterweight. The drum will be 4.5 m (14.8 ft) in diameter. There will be six ropes, 3.2 cm (1.25 in.) in diameter, with tailropes for balance. There will be guides running the length of the shaft and into the headframe to position the conveyance in the shaft during operation. The maximum speed will be 2.5 m/s (8.3 ft/s), and the cycle time will be 21 min. The waste shaft will have a small independent cage for shaft inspection.

3.2.1.8 Electric Power Supply. Electric power will be obtained from the Bonneville Power Administration (BPA) through the Hanford Site power system. A standby power system will be equipped with diesel generators. There will be an underground substation in the central pillar area, supplied by 13.8-kV lines that will come down the R1 and R3 shafts. Transformers will supply power at 4,160 V or 480 V to pumps, borehole drills, fans, and cooling stations. There will also be an uninterruptible DC instrument power system. Other details remain to be developed.

3.2.1.9 Subsurface Ventilation System. The ventilation system will provide fresh, chilled air to maintain environmental conditions and will remove dust, methane, diesel, and blasting fumes from the development and confinement areas. In addition, it will provide for control of radiation releases.

The repository will be designed so that development and emplacement may be carried out at the same time. The rooms undergoing development will have one ventilation system, and the completed rooms in which emplacement is in progress (referred to as confinement) will have a second independent ventilation system. Separation will be maintained by use of brattices (temporary walls of wood or fabric) in the drifts during the early phases and by double doors and permanent bulkheads later. The confinement area will be maintained at a slightly lower pressure than the development area, so that any leakage will be into the confinement area.

Of the five main access drifts, numbers two and four will provide fresh air (and access) to the development rooms. Exhaust air will flow out through the central access drift (number three). Fresh air and access to the confinement area will be through access drifts number one and five (the outer drifts). The confinement exhaust air will flow out the two perimeter drifts.

Intake air will be chilled, if necessary, by cold water from a surface refrigeration plant. Forcing fans at the top of shafts R1 and R2 will provide the intake air for the development area. Exhaust will be through shafts R3, R10, and R11. For the confinement area, chilled air will be supplied through shafts R8 and R9, and the exhaust fans will be located at the top of shafts R5, R6, and R7. The waste handling shaft, R5, will have a relatively low airflow rate, and the air will be continuously filtered because the waste containers will be handled there. The bulk of the confinement air will be exhausted through R6 and R7. The air will be

filtered on demand, if radiation is detected by monitors in the exhaust drifts underground. Confinement air will be exhausted through tall stacks. All intake and exhaust structures will be reinforced-concrete buildings.

Subsystems will include the surface refrigeration plant and cooling towers (which provide chilled water), the air cooling units and fans, the filter systems, and the radiation detection system in the confinement exhaust drifts.

3.2.1.10 Dewatering System. The dewatering system will be designed to control normal water from development operations and normal seepage water plus abnormal inflows. The repository will be constructed so that there is a general dip to the south (0.5 degrees or 0.89% grade) and, further, the emplacement rooms will gently dip away from the central access drifts toward the perimeter drifts. A covered ditch in the floor at one edge of each drift will carry the water to settling sumps, one per quadrant. The settling sumps in the northern two quadrants will drain into collection sumps at the bottom of shafts R6 and R7. Water in the southern settling sumps will be pumped to the collection sumps. The shaft bottoms will also serve as sumps, the water there being pumped to the collection sumps.

The collection sumps have capacity for 3-d normal flow plus 16 h of abnormal flow. High-lift pumps (100% redundancy) will move the water to the surface in a single lift, where it will be discharged to retention ponds at the surface near shafts R6 and R7. Each shaft will have two retention ponds and one percolation pond. The water from the shafts will be sampled continuously for radioactivity. If radiation is detected, the water will be decontaminated before being moved to the percolation ponds.

For abnormal inflows of water, there will be emergency pumps available, and watertight doors may be installed at critical locations. The two collection sumps will be connected to provide operational flexibility during abnormal conditions.

3.2.1.11 Explosives. Surface storage of explosives will be in the southeast corner of the site, behind the basalt storage pile. Two magazines will be provided for explosives and one for detonators. Up to a 2-wk supply of explosive will be stored underground in two magazines in the central pillar off one of the two development intake drifts. There will be two magazines for detonators off the same drift. The magazines will be separated by at least 45.7 m (150 ft). Materials will be transported from the surface magazines via the service shaft (R1) located in the same development intake drift as the magazines.

3.2.1.12 Diesel Fuel. Storage of diesel fuel for the underground waste transporters will be provided in an underground tank. Fuel will be transported in small tanks via the service shaft (R1) to a facility in the central pillar region for refueling the transporters.

3.2.2 Design Assumptions

The conceptual design to date has concentrated on aspects of the repository that are related to process flow. Thus, items such as drift design, borehole design (related to heat load), rock stresses, hoist design (for development), and the ventilation system design (overall, related to meeting cooling requirements) are much farther along than other items. A number of utility functions not related to process flow have only been sketched. Included in the latter category are the dewatering system, the electrical power supply, and overall plant control.

The methodology used by the Q-List Preparation Team required that estimates of radiological dose to the maximally exposed member of the public, standing at the site boundary (along Route 240, approximately 1.6 km (1 m) from the nearest repository shaft), be made for a set of "credible" accidents under both mitigated and unmitigated conditions. To arrive at these quantitative estimates required considerable information about repository design. Where such information was not available in the current design, the Q-List Preparation Team was instructed by the methodology to make reasonable assumptions about the design.

The assumptions, both explicit and implicit, made by the Preparation Team are provided below. Some of these assumptions, particularly those that were the basis for not Q-Listing certain structures or systems, become implicit design commitments. If the final design does not conform to the assumption, then the structure or system not on the Q-List may need to be added to the Q-List.

3.2.2.1 Procedures. It was assumed that procedures will be in place during operations that tend to prevent hazardous situations from developing. An example is preventing the refueling of the underground transporter with a waste container onboard.

3.2.2.2 Electric Power. The electrical power system will be designed to nuclear plant standards. Specifically, Q-Listed parts of the electrical system will be segregated from the rest of the electrical system, which will be designed to rules for design that ensure it will not interfere with the performance of the Q-Listed portion of the system.

3.2.2.3 Water Treatment Facility. The repository dewatering flow will be provided with a radionuclide removal system and those sensors necessary to identify when the radionuclide removal capability needs to be on-line. Preliminary analysis indicates that the radionuclide removal system needs to be capable of lowering particulate radionuclide concentrations by at least a factor of 60 to meet the requirements of 10 CFR 60.2 In addition, as-low-as-reasonably-achievable (ALARA) requirements and the need for operational flexibility suggest the desirability of a significantly better decontamination factor for the dewatering flow. A decontamination factor of 1,000 for the dewatering flow was assumed for this report.

3.2.2.4 Radiation Detectors. Radiation detectors for confinement effluent air and for water pumped out of the repository were assumed to be part of the design, as were controls for activating mitigative systems.

3.2.2.5 Confinement Effluent Air Filtration. It was assumed that HEPA filtration of the repository confinement side air exhaust flow will be available when needed. In addition, the necessary radionuclide sensors and control and actuation systems to bring the filtration capability on-line when needed will be provided. Based on the Preparation Team's analysis and engineering judgment, a particulate removal system (HEPA) with a removal efficiency of 0.9999975 (corresponding to a decontamination factor of 4.0×10^5) was assumed to be part of the repository design.

One of the assumptions regarding the efficiency of the HEPA filters was that they would remove 50% of the volatile radionuclides reaching them. Since the analysis treated the cesium inventory as entirely volatile, the resulting site boundary dose was approximately a factor of 250 too high. This problem has been eliminated for this report by assuming the existence of a subsystem that will lower cesium concentrations in the confinement exhaust flow by a factor of 500. This problem will be eliminated in the repository, either through an improved understanding of the behavior of cesium in repository accidents or by the addition of additional cesium removal capabilities to the repository design.

3.2.2.6 Criticality. It was assumed that both repository and waste package design will incorporate design features necessary to make nuclear criticality, as a result of normal repository operations or subsequent to a repository accident or due to longterm deterioration of the waste package, an incredible event.

3.2.2.7 Radionuclide Inventory. For purposes of radionuclide release analyses, the spent fuel assumed for high-level waste was based on ORIGIN-2 analyses contained in Tape 12796 for 60,000 MWD/MTU burnup and a 5-yr cooling period (Weston 1986).

3.2.2.8 Explosives Transport. It was assumed that explosives will be transported from the underground magazines to the development face in lots of about 450 kg (1,000 lb), and that design layout and administrative controls will rigorously separate waste containers from explosives at all times.

3.2.2.9 Explosives Magazines. It was assumed that repository development explosives will be stored in subsurface magazines, designed to Mine Safety and Health Administration (MSHA) standards, in quantities sufficient for approximately 2 wk normal repository development.

3.2.2.10 Instrument and Control Qualification. Q-Listed instruments and control systems will be designed and qualified to function in any post-accident environment in which their function is required.

3.2.2.11 Missile Shielding for Q-Listed Items. Missile shielding will be provided, as necessary, for the waste package and any Q-Listed systems.

Exceptions to this are situations in which a Q-Listed system may be exposed to missile damage, but the damage could not be reasonably assumed to occur concurrently with the demand for the system function. An example of this would be tornado missile damage to radiation sensors in the ventilation system. Tornados should not cause any subsurface release of radionuclides and, therefore, shielding of the radiation sensors may not be needed for tornado missiles.

3.2.2.12 Three-Level Confinement. In the surface facilities, there will be at least three levels of confinement for waste materials.

3.2.2.13 Radioactive Material Containment. Single containment of radionuclides is acceptable, when it is provided by a "robust" container. Over-the-road shipment of radioactive waste uses this standard.

3.2.2.14 Q-Listed Controls and Monitors. "Active" Q-Listed functions requiring Q-Listed control and monitoring functions will be needed to meet safety requirements. It was assumed that there will be active facility systems that are Q-Listed and require Q-Listed control and monitoring subsystems.

3.2.2.15 Hot Cell Cooling for Waste Materials. For waste material with the design basis radionuclide inventory, heat generation rates are low enough that the waste handling hot cells will not require specially qualified cooling systems.

3.2.2.16 Transfer Port Sealing Features. Waste container transfer port sealing features provide enough of a containment and confinement function for the waste material as it moves through surface processing that the sealing features should be Q-Listed.

3.2.2.17 Hot Cell Contamination. It was assumed that the hot cells and other processing areas with cranes present will be designed to permit recovery from a crane accident dispersing significant amounts of contamination throughout the cell. On this basis, the cranes are not Q-Listed.

3.2.2.18 Road Cask Transfer Cart. It was assumed that cart integrity is not necessary to maintain the cell boundary or to protect the waste container during transfers. On this basis, the cart was not Q-Listed. Transfer port sealing features will be Q-Listed.

3.2.2.19 Surface Transporter Integrity. It was assumed that surface transporter integrity is not important to maintaining containment and confinement during waste container loading onto the waste hoist conveyance.

3.2.2.20 Hot Cell Fire Suppression System. It was assumed that a significant fire in the hot cell will affect the cell atmosphere and produce a large amount of heat, which could affect cell and ventilation integrities. On this basis, the Hot Cell Fire Detection and Suppression system needs to be Q-Listed.

3.2.2.21 Diesel Fuel Separation Requirement. For surface systems, it was assumed that separation imposed by design and administrative controls will keep any diesel fuel fire hazard away from the waste material and associated safety systems. On this basis, the surface diesel fuel supply does not need to be Q-Listed.

3.2.2.22 Radioactive Solid Waste. It was assumed that any hazardous portions of the system for handling site-generated radioactive solid waste (including radioactive waste incinerators) will be enclosed by Zone I confinement.

3.2.2.23 Radioactive Organic Waste System. It was assumed that the liquid organic chemical waste system will have suitable fire protection.

3.2.2.24 Vent Off-Gas Control. It was assumed that there will be vent off-gas control in place to protect the HEPA filters from acidic or caustic vapors, organics, or moisture.

3.2.2.25 Q-Listed Emergency Power. It was assumed that confinement functions will be required following some of the repository design basis accidents. On this basis, it is assumed that the repository requires a Q-Listed AC power supply.

3.2.2.26 Uninterruptible Power Supply Requirement. It was assumed that there are Q-Listed monitoring and control functions that require power following accidents. On this basis, a Q-Listed uninterruptible DC power system is required.

3.2.2.27 Criticality Alarms Inoperative. It was assumed that administrative controls will require the suspension of all radioactive waste handling when the criticality alarms are not functional. On this basis, the alarms are not on the Q-List.

3.2.2.28 Computer Limited to Non-Safety Functions. It was assumed that the plant computer is not involved in Q-Listed functions. On this basis, the computer is not on the Q-List.

3.2.2.29 Filtration Activation. It was assumed that a reasonable mechanism will be provided to bring the off-line filtration systems on the confinement exhaust shafts R6 and R7 on-line when they are needed.

3.2.2.30 Site Power Sources. It was assumed that two, partially independent sources of offsite AC power and two, fully independent, onsite emergency AC power systems would be provided.

3.2.2.31 Hoist Drum Protection. It was assumed that the design of the waste-handling hoist system will adequately prevent accidents (such as lubrication leaks) that could significantly reduce the friction between the hoist drum and the cables.

3.2.2.32 Underground Waste Handling. It was assumed that the underground waste handling equipment will be designed to prevent waste handling accidents that could breach the waste container.

3.2.2.33 Subsurface Fire Protection. It was assumed (on the basis of analysis described in more detail in section 5.1.1.6 and appendix A) that the only subsurface fire suppression system needed to protect public safety would be the system needed to control diesel fuel fires at the storage tank and transporter refueling area.

3.3 METHODOLOGIES

Methods followed in developing the Q-List for structures, systems, and components were focused in four areas: important-to-safety surface facilities, important-to-safety subsurface facilities, important-to-retrievability subsurface facilities, and important-to-waste isolation engineered barriers and natural systems. These methods are discussed in the following sections.

3.3.1 General Methods

The general approach to development of important-to-safety Q-Lists began with review of the conceptual design to establish general facility requirements and planned systems to accomplish functions. General hazards that would be encountered were listed with particular emphasis on potential releases of radioactive materials. A general list of accident initiators, including both natural events and system failures, was developed. Potential consequences of accident sequences were developed. A screening process was used in performing a risk assessment for accident probabilities and consequences. Events of very low probability or low consequence were screened from further consideration. Systems, structures, and components that were essential to prevention of significant offsite doses were listed as Q-List items. Design assumptions were developed as needed to support the selection of the Q-List items.

3.3.2 Surface Facilities Important-to-Safety Q-List

The method used for surface facilities Q-List development was as outlined above but qualitative assessments were made to develop the Q-List. The flow logic for surface Q-List development is provided in figure 4.

The CDR (KE/PB 1987) was reviewed and general experience was used to develop a list of the surface systems to be considered. A listing of possible accident initiators was developed and screened on the basis of experience to produce a listing of design basis events to be considered for system evaluation. A hazards analysis was also conducted to ensure that appropriate systems were in place to prevent, detect, and mitigate these hazards. The information developed in these areas was used to provide input to the evaluation process.

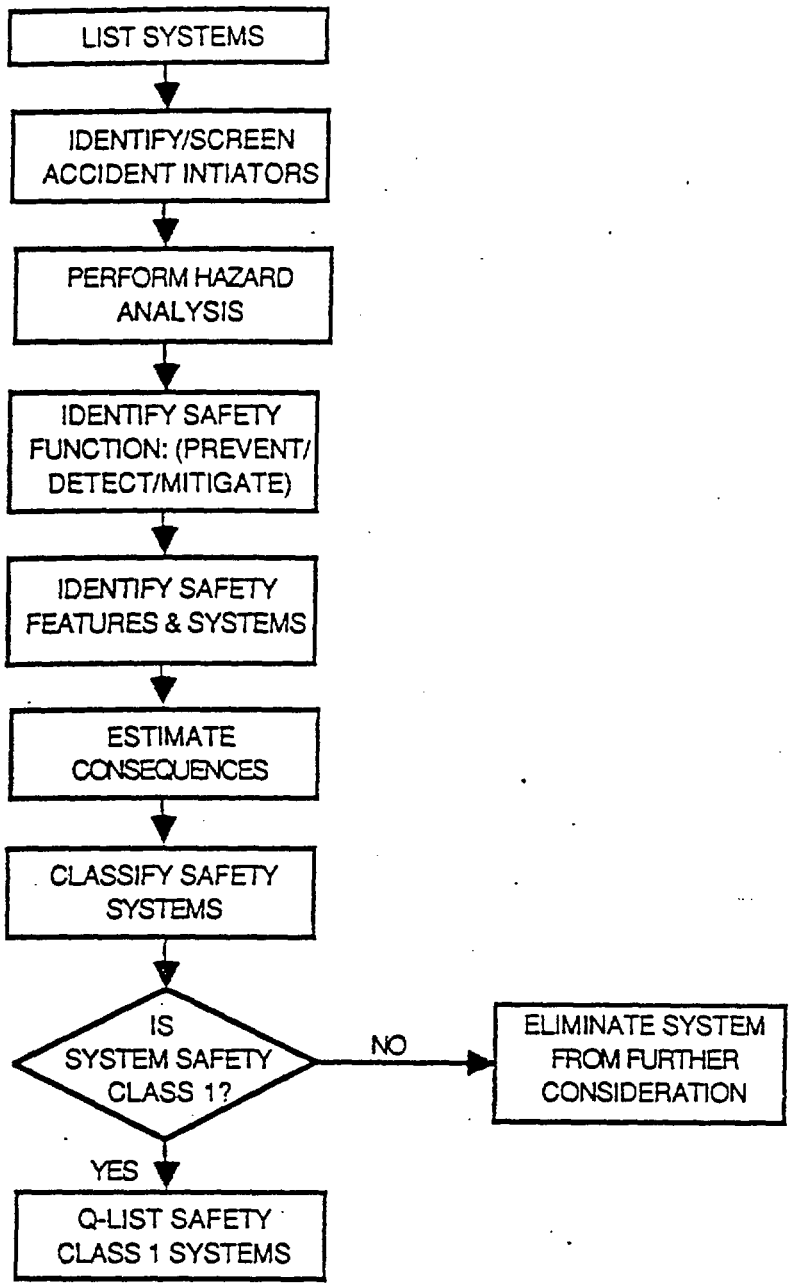

Figure 4. Surface Facilities Important-to-Safety Q-List Development.

An evaluation was performed for the most significant surface systems. Safety functions were identified to ensure that design basis events and hazards would be controlled. Safety failures of systems and offsite consequences were estimated. This evaluation formed the basis for safety classification.

Individual systems were assigned to one of three safety classes based on the estimated consequences of failure of the system to perform safety functions. Safety Class 1 system failure leads to serious offsite consequences; Safety Class 2 system failure leads to serious onsite consequences; and Safety Class 3 system failure may affect individual worker safety. The results of this evaluation process are presented in section 4.6. If a system is classified as Safety Class 1, it is Q-Listed. The surface systems Q-List is presented in section 4.7. Important assumptions that support the evaluation and classification process are presented in section 3.2.2.

3.3.3 Subsurface Facilities Important-to-Safety Q-List (Methodology A)

Subsurface Q-List development included the use of event trees to perform a risk assessment for systems, structures, and components important to safety. Those elements listed on the event trees that are needed to prevent or mitigate accidents to within acceptable limits were Q-Listed.

The flow logic for Methodology A is provided in figure 5. The design data base was drawn from the CDR (KE/PB 1987), Conceptual Design Criteria Document (Naiknimbalkar 1986), and other reference documents, including DOE orders. Accident initiators were based on the OCRWM methodology (Weston 1986). These accident initiators were augmented with site specific initiators. Initiator frequencies were established using the best data available and documented (see table 7) to establish the basis for risk estimates to be developed. Where accident initiators were below the threshold frequency of 1.0×10^{-7}/yr (or 1.0×10^{-5}/yr with 95% confidence), they were eliminated from further consideration for this Q-List activity. The subset of credible initiators for further consideration was then used as an input for consequence screening.

Coarse hazards analyses were performed to ensure that important hazards to the public, the environment, the contained waste, the operators, and the safety systems were addressed in the evaluation process. This helped ensure that significant hazards to the public were recognized, in addition to the obvious hazard from radioactive material releases.

A screening process was used to focus attention on events with a significant level of risk based on seriousness of consequences. If the consequences of the event resulted in less than 100 mrem whole body dose at the site boundary, it was dropped from further consideration.

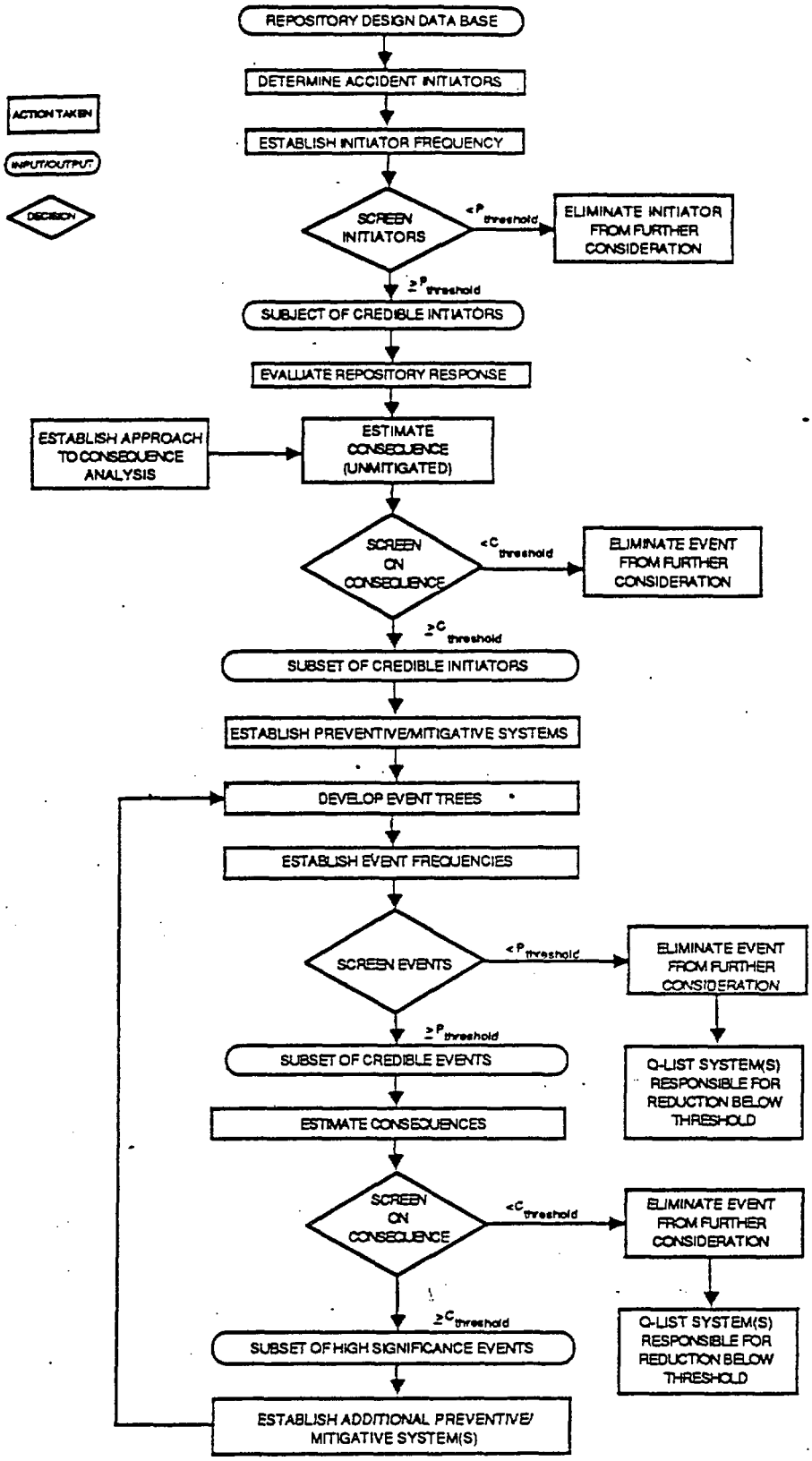

Figure 5. Flow Logic for Methodology A.

A subset of unmitigated events that passed both frequency and consequence screening was developed. The preventive and mitigative systems for the repository were developed from the design database as augmented by assumptions where needed. Event trees were developed for each event as shown in figure 5. Where preventive systems were assumed to operate and event sequences can be shown to be incredible, the event was considered to be controlled and was eliminated from further evaluation. Systems responsible for reducing event frequencies below the threshold frequency were then considered to be important to safety and were accordingly Q-Listed.

Where events were still sufficiently frequent to merit further consideration, mitigating systems were considered in the event trees. Where consequences could be mitigated to below the threshold consequence, the event was considered to be adequately controlled and was not considered further. Systems responsible for reducing consequences to below the threshold value were considered to be important to safety and Q-Listed. Where consequences remain too high, additional preventive or mitigative systems were assumed and new event trees devised until all events had been suitably controlled.

General functions of systems as developed in the event tree were based on engineering experience, conceptual design criteria, or the conceptual design report. In most cases, the main functions of the system are inherent in the name of the system. In cases where this was not so and the system has an important safety function, these functions were described.

Several qualitative design objectives were used in developing and evaluating the event trees used to support Q-List development. Among these are defense in depth, avoiding challenges to safety systems, and Prevention of accidents (in preference to mitigation of accident effects). This latter objective would favor the selection of a system that prevents an accident for Q-Listing as compared to listing a mitigation system. An example of this would be to Q-List a crane that prevents dropping material. However, both systems may be listed to satisfy the defense-in-depth objective, or only the mitigative system may be Q-Listed for safety or risk-assessment reasons.

3.3.4 Retrievability Q-List Development (Methodology B)

The flow logic for Methodology B is provided in figure 6. The procedure for Methodology B was the same as that for Methodology A with the following two exceptions:

- The consequence screening was done on the basis of technical judgment whether the initiating event or the event sequence (taking preventive/mitigating systems into account) renders retrieval impractical

- The selection of preventive and mitigating systems may be different when considering the impact of an event sequence on retrievability.

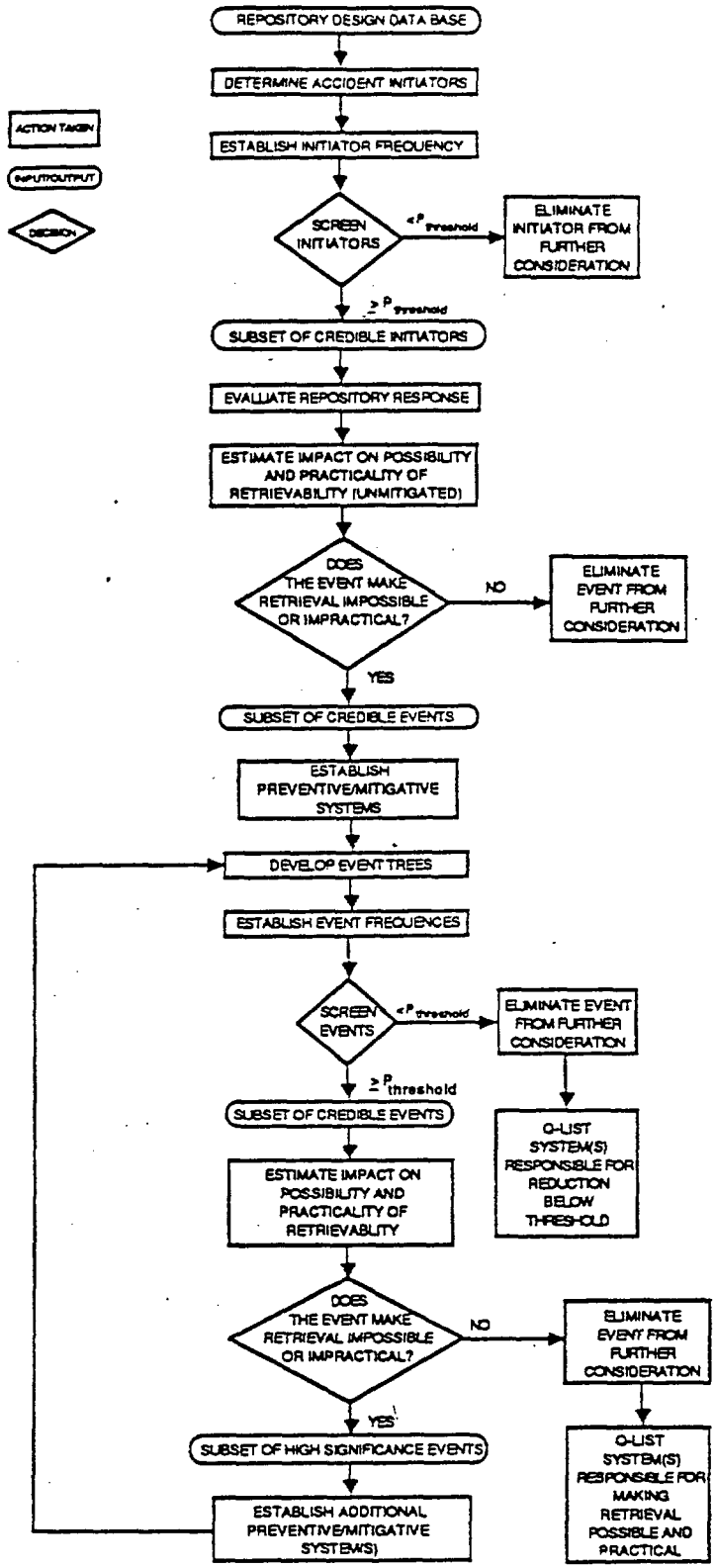

Figure 6. Flow Logic for Methodology B.

An evaluation was made regarding the importance of system, component, or structure to the practicality of retrieval, given certain events have occurred. If the failure of such a system, component, or structure makes retrieval impossible or impractical, then the causal item was judged to be important to waste isolation and Q-Listed.

It will be important to assure, in the event that retrieval of high-level emplaced waste becomes necessary, that it can be retrieved safely, while maintaining radioactive material releases within acceptable limits. Equipment needed to maintain radiation doses within acceptable limits would normally be important to safety and Q-Listed. In most cases, the conditions for "normal" retrieval will be identical with those for emplacement, so the determination of the emplacement important-to-safety Q-List items will provide most of the items for the retrieval list.

There could be situations in which the repository systems and geologic features are damaged or the waste packages are damaged, thus requiring retrieval of high-level waste. Retrieval under these circumstances may require the use of specialized equipment to deal with a hostile environment, abnormal configurations, and radioactive material releases, while providing adequate protection for operators and the public. This specialized equipment will be included on the Q-List, if it is required to protect the public or to ensure retrievability.

3.3.5 Important-to-Waste Isolation Q-List Development (Methodology C)

The postclosure portion of the Q-List consists of barriers important to waste isolation and the activities that must be controlled to ensure that those barriers will satisfactorily inhibit radionuclide release to the accessible environment. Figure 7 provides the flow logic for Methodology C (which is quite different than Methodologies A and B).

The procedure began with identification of engineered and natural barriers in accordance with 10 CFR 60.112 and 60.113. Barriers were chosen conservatively, such that all barriers potentially required were included. Where necessary to identify particular barriers based on current design and performance assessment/ performance allocation considerations, it was performed by technical consensus of three qualified persons.

It is also necessary to identify the activities that must be properly conducted to ensure that the barriers will perform satisfactorily in service and to include performance assessment activities where they are necessary to demonstrate satisfactory performance. Consideration must be given to potential preclosure and retrieval activities and events, as they might affect postclosure isolation.

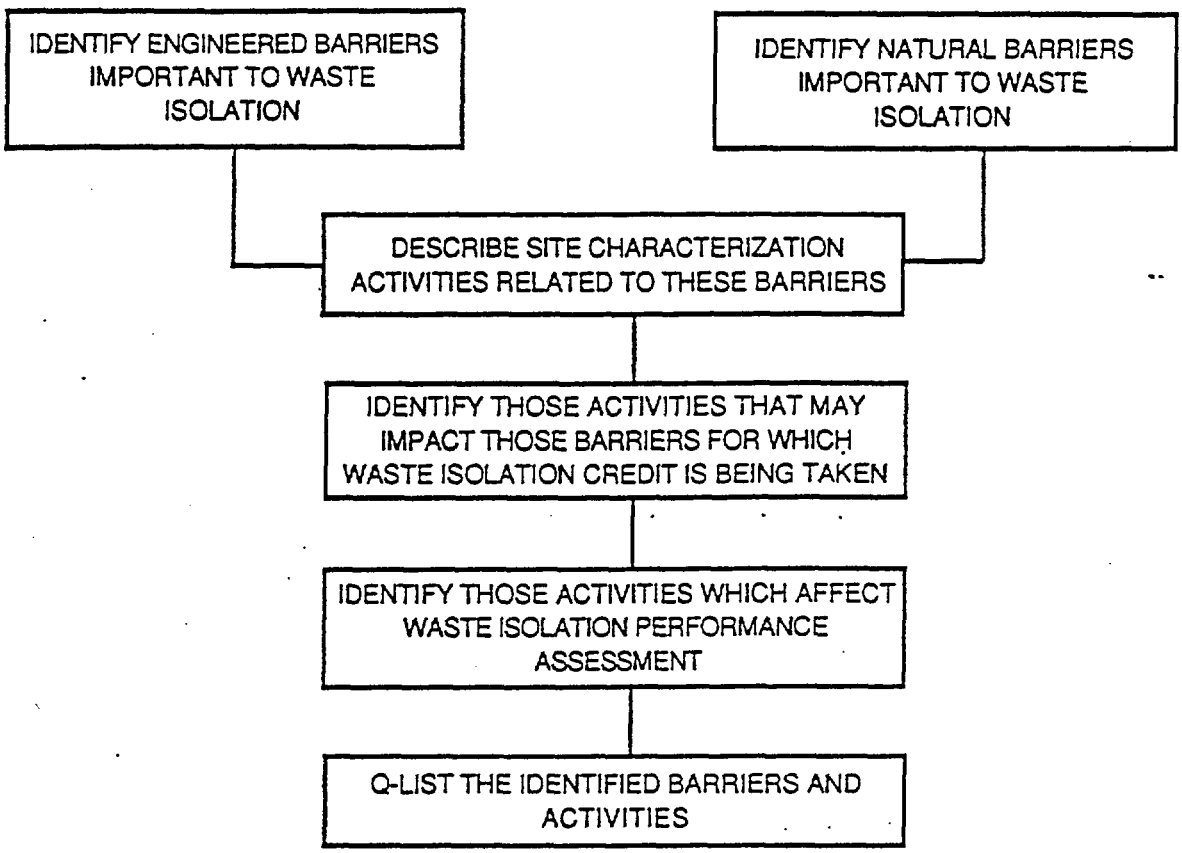

Figure 7. Flow Logic for Methodology C.

3.3.6 Limitations of the Current Analysis

The present Q-List Preparation Team effort for initial determination of the basalt repository Q-List follows the well-defined OCRWM methodology (Weston 1986). This methodology uses some of the tools of probabilistic risk assessment at a level of detail appropriate to the conceptual nature of the current design. The intention is that this methodology will be interactive with the results of the analysis at each stage used to modify the Q-List and feed back into the design process to help establish the reliability and functional requirements for repository structures, systems, and components.

This initial iteration for the basalt repository Q-List is being done under several significant constraints. The first, and expected, constraint is the early, conceptual nature of the design. Some repository systems are only sketched at this time using broad, functional requirements. This impacts both the assumptions that can be made about the mitigating capabilities of a system as well as the assumptions to be made about the availability or the reliability with which it performs its mission. The approach of the Preparation Team to this constraint, in most cases, was to use estimates of function and reliability that are representative of the state-of-the-art. In a few cases, it will be important that a system meet functional or reliability requirements that are not specified in the current design or that are beyond those specified by the design. In those cases, the assumptions of the Preparation Team become implicit functional or reliability requirements for the system.

The second major constraint is related to the relatively tight schedule on which this activity is being carried out. The only computer code analyses used by the Preparation Team are the radiological dose assessments, using codes maintained by Pacific Northwest Laboratory (see subsection 5.4.2.1). These have been used to calculate the dose to the hypothetical maximum exposed individual at the site boundary. These codes are in place, verified and validated, and under appropriate quality assurance controls. Other techniques were used to produce an estimate of the radiological source term of the surface release produced by a subsurface accident. The schedule did not permit developing and verifying codes that would allow a more sophisticated treatment of the physical processes producing subsurface releases or the physical process removing or retarding radionuclides on their way to a surface release point. In fact, codes for dealing with the transport and retardation processes in the context of a repository (rather than a reactor containment) are thought not to be well advanced and may require a code development effort before they can be applied to repository accident analysis.

In view of these constraints, one working group composed of four technical people with backgrounds in material properties, reactor accident phenomenology and code development, reactor operational safety, and spent fuel waste package design was used. This working group has combined discussions of accident sequences, literature search, and some bounding calculations to arrive at a consensus of which repository accident sequences are significant and credible, and the appropriate subsurface and surface

source terms for those accident sequences. A similar technique was used by another working group to arrive at estimates for the probabilities of accident initiators and the event tree branch point probabilities.

4.0 SURFACE SYSTEMS Q-LIST

4.1 INTRODUCTION

The surface facilities were evaluated using qualitative methods to establish systems with safety functions, safety classifications, and systems important to safety. Systems typical to a waste handling facility were assumed to supplement information in the CDR (KE/PB 1987). Accident initiators were developed and preliminary hazard analysis was performed for each system. Safety classes were assigned on the basis of safety functions to be performed. Those systems with Safety Class 1 designation were Q-Listed. Important design criteria needed to support the Q-Listed equipment were also developed.

4.2 METHODOLOGIES

4.2.1 Surface Facilities System List Development

For the purposes of this review, surface systems were taken to be the systems for receiving, storing, processing, containerizing, and transferring the waste to the waste-handling shaft. General utilities and emergency power generators were also listed as surface systems. A brief review of underground systems (review encompassed entire underground systems) was conducted, including those portions of the underground systems that are physically located on the surface. Examples of these are the hoist head frames and the confinement exhaust filtration system. (The classification of these systems, including their surface portions, is reported in chapter 5.)

A listing of surface facility systems was developed using information from the CDR and general experience with fuel-handling facilities. The systems listing developed for the surface Q-List was only intended to list systems with significant hazards for important safety functions and was not intended to be a complete listing of all surface systems. The systems list used for evaluation is provided in section 4.3.

The development of the surface facilities Q-List was performed on a qualitative basis (as contrasted with the quantitative approach for underground facilities). The systems list was reviewed for systems that presented significant hazards or systems with important safety functions. On the basis of engineering judgment, it appeared that some listed systems would not have important safety functions.

A preliminary safety classification was assigned to each system on the basis of its relevance to safety as part of the evaluation process. The classifications are defined as follows:

- Safety Class 1: An item whose failure could cause or permit the release of radioactive material that could result in a whole body dose of 0.5 rem at the site boundary

- Safety Class 2: An item whose failure could cause or permit the release of radioactive material that could result in a significant dose or other serious consequences to onsite personnel

- Safety Class 3: An item whose failure could result in occupational level radiation doses or typical industrial risks to personnel.

Assumptions regarding repository systems and safety functions of those systems are described in section 3.2.2. Based on experience with such systems at other nuclear facilities and on safety analyses done for those facilities, safety classifications were assigned to all repository systems. Actual design evaluations and safety analyses may show later that the consequences of repository accident sequences will not exceed limits. This would permit reassigning the safety classification.

In most cases, structures are considered to be secondary systems that house, support, and protect engineered systems and thus would not directly provide primary safety functions. However, when the structural element is a containment or confinement barrier that limits the release of radioactive material to the environment, it is considered to provide a primary safety function.

Systems were identified as Safety Class 1 when it was estimated that the direct failure of the system could lead to significant offsite exposures. The assessment was qualitative in that appreciable source terms were assumed and system configurations were assumed or were based on the CDR design (where available). Significant consequence (offsite release) assessment was based on engineering experience and judgment. Important assumptions made to support these judgments are discussed further in section 3.2.2.

A brief listing of hazards was made on the data sheets as a basis for evaluating each system. In this context, hazards originating within the system and external to the system are considered. Thus, the system is considered as a potential source, as well as a target, of hazards. "Hazards" is used here in its full sense; to include hazards to people, the radioactive waste, and other systems. As discussed in section 4.4, a listing of facility accident initiators was developed for use in the evaluation. Those that could significantly affect the evaluated system are listed. Failures of structures, systems, or components that have the potential for affecting the safety function were also listed.

As part of the overall system of safety assurance, safety class functions of all systems are listed as a means of supporting the Q-Listed items. As an example of this process, confinement exhaust flow for the Waste Handling Building is required for all conditions. Therefore, the exhaust system is Q-Listed. To have flow, emergency power must be provided for design basis events where normal power is disrupted. Thus, the emergency power system and the structures containing it are also Q-Listed.

When most of a system is required to provide a safety function that prevents or mitigates radioactive releases to the environment, then the system is Q-Listed. Similarly, if most of a structure has a primary safety function (as does the Waste Handling Building), it is Q-Listed. However, when only one or two features of a system are needed to provide Safety Class 1 functions, then only these features are listed.

A basis statement for each safety classification was developed on the data sheets to show why the safety classification was made. In many cases, important qualifying assumptions were necessary to support both the classification made and the basis statement. These assumptions have been compiled separately and appear in section 3.2.2.

4.2.2 Identification of Events and Accidents

A list of initiating events and accidents was developed for the surface facilities (see section 4.4). These events included natural phenomena such as earthquakes and tornados as well as internal facility events such as fires and explosions. System failures were also considered. These events were used in the evaluation to establish hazards and needed safety functions.

A qualitative assessment was made to establish systems and safety functions that would be needed to respond to identified events and accidents. Needed qualifications for systems to survive initiating events and accidents were also considered. Engineering judgment was used to establish Q-Listed systems that would provide primary safety functions assuming that initiating events and accidents occur.

4.2.3 Hazards Analyses

The hazards evaluated in developing the Q-List included hazards arising both within the system and external to the system. Hazards that could impact the evaluated system and other systems were reviewed and identified on the data sheets. A hazard is considered to be a source of risk peril for persons, material, and hardware. In this case, of particular importance are the hazards to the public, to high-level waste and waste containers, and to safety systems. A summary of the hazards analysis is provided in section 4.5.

In a manner similar to the approach to initiating events and accidents, hazards were evaluated to ensure that safety functions and systems were in place to respond to hazards and also that other systems would be qualified to function with such hazards present. Systems needed to respond to significant hazards were considered for the Q-List. In cases where the hazard could be readily separated from a safety system, this was recommended rather than Q-Listing the system. An example of this would be to separate the surface explosives magazine from all safety systems (except those dealing with the magazine) and to separate the magazine from radioactive waste materials.

4.2.4 Systems Evaluations and Safety Classifications

As a result of the evaluation process, the important surface facilities systems that will likely be needed for the repository were listed and evaluated. Only those systems that were evaluated are discussed in section 4.6. Safety classes were assigned to each system.

4.2.5 Q-List Summary

Systems identified in the evaluation as being Safety Class 1 were Q-Listed. A table of these systems is provided in section 4.7.

4.2.6 Design Criteria to Support Q-List

As a part of the system evaluation and safety classification, a number of questions were raised about the planned design criteria or design concepts. Several assumptions were needed to determine appropriate safety classification. It is intended that design criteria be developed to address these points. These are described in section 3.2.2.

4.2.7 Reviews

The system listings were reviewed with assistance from KE/PB personnel to ensure that significant systems either presenting hazards or providing important production or safety functions had been considered in the evaluation process.

4.3 IMPORTANT SURFACE FACILITY SYSTEMS

No conceptual design for the surface facilities exists. Thus, as a part of the evaluation process, the important surface facilities systems that will likely be needed for the repository were listed based on experience with nuclear facilities. Only those systems that pose significant hazards or are likely to be safety class systems were assumed for the evaluation. The system list is provided in table 3.

4.4 INITIATING EVENTS FOR SURFACE FACILITY Q-LIST

As a part of the evaluation process, initiating events potentially affecting the various systems were considered. A preliminary list of events was considered based upon events that are typical for present Hanford Site

Table 3. Potential Surface Systems for Evaluation.
(sheet 1 of 2)

System	Subsystem
Yards	Railyard Roadyard Storage Parking Basalt pile
Structures	General structures--process General structures--minor Explosives magazines Radiation shielding Primary confinement barrier Secondary confinement barrier Tertiary confinement barrier Waste Handling Building Site Control Room Equipment support Storage/warehousing Office buildings
Waste Containment	Unconsolidated spent fuel Unconsolidated spent fuel--containerized Consolidated--containerized fuel West Valley High-Level Waste Shipping casks Underground transfer cask Waste container Emplacement container
Material Handling	Cask off-load system Cask unloading system Lag storage Container packing and welding Container inspection Defective waste handling Container decontamination Container loadout Underground cask transporter Cask receiving crane (150T) Cask preparation crane (20T) In-cell hoists Cask transfer crane Hoist cage loader Receiving cask cart Waste consolidation equipment Cleaning systems

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Table 3. Potential Surface Systems for Evaluation.
(sheet 2 of 2)

System	Subsystem
Ventilation Systems	Zone I Ventilation (Surface) Zone II Ventilation (Surface) Zone III Ventilation (Surface)
Auxiliaries	Hot cell fire protection system Heat removal system Inert gas supply Flammable gases: propane, H2 Diesel fuel supply Mining explosives supply Site-generated high-level solid radioactive waste Radioactive waste incinerator Low-level liquid radioactive waste High-level liquid radioactive waste Vent offgas radioactive waste
Electrical	Emergency power system Uninterruptible power Main preferred power Alternative preferred power Lighting High voltage DC power
Instrument/Control	Accountability and criticality control Effluent radiation monitoring In-cell radiation monitoring Personnel radiation monitoring Criticality alarm system Fire detection/alarm Water leak detection Seismic monitoring Confinement function monitoring Communications
Computer and Data Processing	Operations computers

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facilities. Initiating events considered for evaluation of the surface facilities are:

EXTERNAL EVENTS

- Design Basis Earthquake (OCRWM Seismic Event)*
- Tornado*
- Flood - Natural Event
- Flood - Pond Breakout
- Volcanic Ashfall
- Helicopter Crash**

INTERNAL EVENTS

- Fire*
- Explosion*
- Water Leak
- Loss of Electrical Power
- Radioactive Material Containment Breach (OCRWM Waste Handling Events)*
- Radioactive Material Confinement Failure (OCRWM HVAC System Failure)*
- Nuclear Criticality
- Human Error

4.4.1 Initiating Events and Accidents

4.4.1.1 Earthquakes. The facility can experience a wide spectrum of earthquakes. For the purposes of the evaluation, however, it was assumed that important facilities will be designed for the Uniform Building Code level earthquake for Zone II.

Facilities important to safety will be categorized as Seismic Category I and will be designed to survive the design basis earthquake. Expected consequences of the design basis earthquake include failures of unqualified structures and systems as well as loss of main electrical power.

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*Weston (1986, table 4-1) lists these events. (Breach of fuel pin clad is also a listed event, but was not considered to be a determining factor for this qualitative Q-List for surface facilities, since the hot cell atmosphere control system was assumed to be designed to accommodate such events as anticipated events.)

**A helicopter crash is listed as an event since security forces routinely use helicopters for patrolling. An aircraft crash is not listed since commercial and general aviation airfields are too far away from the site to pose any significant hazard.

4.4.1.2 Tornado. The important-to-safety structures should be designed to withstand tornado winds and tornado-generated missiles. It should be noted that protection can be afforded in most circumstances by hardening the outer boundary of structures so that everything inside is protected. A potential effect of a tornado is a loss of main electrical power.

4.4.1.3 Flood. For typical plant sites in other regions, a natural flood is used for design purposes. At the proposed site of repository surface facilities, natural floods are of low concern because the site is located on high ground well away from the major rivers. However, the fact that water will be pumped out of the ground to an impoundment presents a potential flooding hazard from a pond breakout. Design criteria providing locations well away from the surface facilities and shaft collars can eliminate or greatly reduce this hazard. The only significant flooding potential is due to a very infrequent flood of Cold Creek (an ephemeral stream located on the site). The probable maximum flood for Cold Creek is considered for the evaluation.

4.4.1.4 Volcanic Ashfall. Active Cascade volcanos can cause several inches of volcanic ash to falling on the site. Potential effects of this ashfall include structural loading of buildings, failure of machinery due to abrasive effects of ash, access difficulties, health effects for personnel, clogging of filters, and electrical shorts and failures. Many of these impacts can be mitigated by proper design criteria.

4.4.1.5 Helicopter Crash. Security forces use helicopters to patrol all of the Hanford Site and its facilities. Procedures, equipment, and training programs have been devised to minimize the chance of helicopter crash into Hanford Site facilities. However, because of the frequency of these flights, this event is sufficiently likely to be considered in the hazard analysis. The hazards include the mechanical impact to structures and exposed equipment as well as a resulting fire or explosion.

4.4.1.6 Fire. Fire preventive and mitigative design features of the facilities are very important. Significant efforts will be focused on preventing fires, but the availability to detect and suppress fires is necessary for industrial facilities and especially for nuclear facilities. If a fire spreads, the potential for radioactive material release is great. Not only can the fire breach barriers and damage equipment, but it can produce more dispersible forms of material.

4.4.1.7 Explosion. It is a good practice to design a facility to eliminate as many of the potential explosion sources as possible. Separating explosion sources from potential targets and shielding such targets may also be effective. Care should be taken to protect the confinement barriers against explosions or missiles generated by explosions.

4.4.1.8 Water Leaks. While externally generated floods are not of great concern on the present site, there is a possibility that internal water systems in the facility could compromise safety functions. Design criteria for water systems, structures, and safety systems should be used to reduce these risks.

4.4.1.9 Loss of Electrical Power. In addition to loss of electrical power due to Bonneville Power Administration network failures, there are many events onsite that could also lead to loss of power events. These include natural events such as earthquakes and tornados, construction activities, and system failures. Important safety functions, monitoring, and control activities should be evaluated for the impact of loss of electrical power.

4.4.1.10 Radioactive Material Containment Breach. Most of the radioactive material in the facility will be in containers that prevent releases to the environment. In the event of a containment breach that releases radioactive material, functions and systems will be provided to limit releases to the environment and the public.

4.4.1.11 Radioactive Material Confinement Failure. During handling, processing, and packaging operations, some of the waste will be in confinement systems that rely on the combination of a barrier and active ventilation/filtration systems to maintain safety. Failures of boundaries and/or ventilation systems can lead to releases of radioactive material. Functions and systems should be in place to limit releases to the environment and the public.

4.4.1.12 Nuclear Criticality. The facility and systems will be designed to prevent nuclear criticalities from occurring. If a criticality does occur, the primary hazards are to operating personnel. Some damage to equipment and releases of radioactive effluents to the environment may also occur for certain types of criticalities. A more complete discussion of criticality concerns is provided in section 4.6.3.5.

4.4.1.13 Human Error. Human errors can initiate or aggravate accident sequences, including some of the events listed above. These human errors in certain cases may dominate system unreliabilities. For this stage of the evaluation, human error as an initiator is recognized but not explicitly evaluated. Human error will be carefully considered as the design is developed, design requirements are established, and safety evaluations are made in greater detail. Design provisions will be made to reduce the likelihood that human errors will initiate significant accidents. Q-Listed features may be needed to either reduce the likelihood of human-induced accidents or reduce the consequences of such events.

4.4.2 System Failures

Potential failures of engineered systems were considered when performing the evaluation. The most significant failures for this facility are system breaches or leaks that can allow the release of radioactive materials to the environment. This can occur as a result of operator error, structural failures, or damage to containment and confinement barriers. For this study, failures that could result in the failure to perform safety functions were emphasized.

An important distinction is whether these failures are failures of active or passive components. Radioactive material in a waste container is

in a passive containment. Failure protection is usually achieved by preventing damage to this containment barrier. Many times, this protection can be achieved by other passive elements. Passive design is preferred because it tends to be simple and stable. Active elements tend to be complex, dynamic, and less easily understood and controlled. However, economics or other reasons often require the use of active systems. The most important active system in the surface facility is the radioactive material confinement system. This system combines a passive (semipermeable) barrier and an active ventilation system, which maintains appropriate pressure differentials and flows and passes all effluents through filters. Failure of this system can result in the release of radioactive materials. Since the system is active, several failures could impair the safety functions, including not only structural failures and leaks, but also failures of monitoring, control, and power supplies. The possibilities of such system failures have been considered in these evaluations.

A defense-in-depth philosophy was used so that system failures can be tolerated. In most cases where a single failure of a component could lead to significant offsite releases, the item was Q-listed. However, it was not necessarily assumed that one Q-List element was sufficient to protect against system failures. For example, redundant confinement is provided for the active confinement systems in that both the Zone I and Zone II confinement ventilation systems are placed on the Q-List. This compensates for the fact that active systems are generally less reliable.

4.5 HAZARDS ANALYSES

Hazards usually take the form of an uncontrolled release of energy adversely affecting a "target." The hazards recognized for the surface facilities include effects of radiation that can cause immediate harm to humans (usually by ionizing radiation) or long-term health effects (usually when inhaled into the body or otherwise incorporated into body tissues). The release of unacceptable amounts of radioactive material is the prime hazard in this evaluation. Most of the other hazards could cause or permit such releases as a consequence. These include explosions, fire, and falls of heavy objects, which could damage radioactive material containers and confinements and disperse the materials to the environment. Other hazards could affect the ability of safety systems to perform their vital functions.

Excessive heat or mechanical loads constitute hazards to the high level waste containers. Explosions could directly breach confinement barriers or produce missiles that could pierce these barriers. Fires in cable trays could damage vital monitoring, control, and power cabling. High radiation fields can damage or disrupt control systems that are not adequately protected. Elastomers and polymers typically used for compliant seals can also be damaged by radiation fields. While water may be useful in fighting fires, it can also damage electrical equipment or provide moderation and reflection that could lead to a nuclear criticality.

The methods to control hazards and their preferred order of application are as follows:

- Eliminate the hazard
- Separate the source from the target
- Install engineered systems to contain/control the source
- Install engineered systems to protect targets.

These approaches were considered when doing evaluations and quite often led to important assumptions about the repository designs (see section 3.2.2).

4.6 SURFACE SYSTEM EVALUATION AND SAFETY CLASSIFICATION

The surface systems were evaluated by developing data sheets for all surface systems that presented hazards or would provide safety functions. As a result of that process, several systems that provide safety functions and might be candidates for the Q-List are listed in table 4. A discussion of the evaluation of each system is provided. Systems that are Safety Class 1 are Q-Listed. The Q-List is provided in section 4.7.

4.6.1 High-Level Waste Containment

4.6.1.1 Shipping Casks (Over-the-Road, Rail). The shipping casks for radioactive material are listed as a surface system since they form a very important control element, even though the casks may originate elsewhere. It is assumed that these casks provide all the protection, containment, and cooling needed for the waste, and that the manner of their routing, handling, and storing will present no hazards greater than over-the-road (or rail) shipment. Also, it is assumed that the parking and storage of casks does not produce an unacceptable concentration of sources.

4.6.1.2 High-Level Waste Container. The high-level waste containers provide the principal containment barrier that prevents release of radioactive material. This is the first and most important barrier and has a key principal safety function. It is classified as Safety Class 1 and Q-Listed.

4.6.1.3 Underground Transfer Cask. The underground transfer cask that receives the waste container is assumed to provide mechanical protection for the waste container against missile, rockfall, fire, and explosion hazards, so that design basis accidents (DBA) are appropriately resisted by the combination of cask and container. The waste container is assumed to provide the leak-tight sealed containment boundary for the radioactive waste. Due to the important safety function of protecting the waste container, the underground transfer cask is classified as Safety Class 1 and Q-Listed.

Table 4. Surface System Evaluations. (sheet 1 of 2)

Systems	Function	Safety class
High-level waste containment		
Shipping casks for waste materials[a]	Radioactive material containment	1
Waste container	Radioactive material containment	1
Underground transfer cask[b]	Radioactive material containment	1
Radioactive material confinement		
Primary confinement barrier	Radioactive material confinement	1
Secondary confinement barrier	Radioactive material confinement	2
Tertiary confinement barrier	Radioactive material confinement and tornado protection	3
Zone I ventilation	Radioactive material confinement	1
Zone II ventilation	Radioactive material confinement	1
Zone III ventilation	Radioactive material confinement	2
Handling and storage		
Receiving area cranes	Cask handling	2
Waste handling shaft transfer cranes	Cask handling	2
Hot cell crar	High-level waste handling	2
Waste proc equipment	Prevents damage to fuel cladding	2
Criticality control systems	Maintains criticality control	1
Cask parking storage	Maintains safe storage	2
Hot cell lag storage	Maintains safe storage	2
Auxiliary systems		
Hot cell fire detection and suppression system	Protects Zone I and II confinement	1
Heat removal system	Protects Zone II high-efficiency protective air	2
Emergency power system	Provides power for monitoring, control, and confinement	1
Uninterruptible power	Provides power for monitoring and control	1
Hot cell cooling systems	Cools high-level waste, maintains boundary integrity	2

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Table 4. Surface System Evaluations. (sheet 2 of 2)

Systems	Function	Safety class
Monitoring and control		
Effluent radiation monitoring (accident)	Monitors releases to environment	1
Seismic monitoring and alarm	Ensures alarms for significant earthquakes and records	2
Confinement functions monitoring and control	Ensures vital confinement functions are being maintained	1
Site Control Room	Monitors and controls facility processes	1
Structures		
Waste Handling Building	Protects confinement and control room	1
Emergency generator buildings[c]	Protects and supports emergency power system	1 and 2
Hot cell transfer port features	Maintains confinement during transfers	1
Missile shielding[d]	Protects waste and safety systems from missiles	1 or 2

[a]This is an important system that will be furnished by others as part of the transportation system.
[b]This is an underground system with important surface interfaces.
[c]The Q-Listed function of the structure is that it not collapse and damage the Q-Listed emergency generators.
[d]Safety class is commensurate with the safety class of item protected and the missile hazard.

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4.6.2 Radioactive Material Confinement

As noted earlier, confinement systems consist of the combination of a physical barrier and a controlled ventilation/filtration system.

4.6.2.1 Ventilation Zones. Ventilation zones in the building are provided according to the likelihood of radioactive contamination. Zone I areas are assumed to be contaminated; Zone II areas have a low potential for contamination; Zone III areas have a very low potential for contamination; and Zone IV areas have no greater contamination potential than areas outside the building.

4.6.2.2 Primary Confinement Barrier. The primary confinement barrier surrounds Ventilation Zone I, which is assumed to have high levels of radioactive contamination due to the handling of radioactive waste. While good design and operating practices should be followed to reduce such contamination, significant contamination in these hot cells is an expected operating condition. Due to the importance of providing first barrier confinement of radioactive materials, the primary confinement barrier is classified as Safety Class 1 and Q-Listed.

4.6.2.3 Secondary Confinement Barrier. Outside the hot cells are the Zone II operating galleries. The Zone II area normally should not be contaminated, and design measures should be followed to maintain it as a clean area. However, it should be recognized that some contamination potential exists, and design features should be in place to decontaminate. The Zone II ventilation area is a backup support for Zone I ventilation. It is also expected that the main confinement exhaust system will be Zone II. This ventilation function supports radioactive material confinement, and therefore, the Zone II barrier is classified as Safety Class 2.

4.6.2.4 Tertiary Confinement Barrier. Tertiary confinement is provided for the Zone III ventilation, which has a very low contamination potential. The Zone III barrier is classified as Safety Class 3. The tertiary barrier may also serve the function of protecting the secondary and primary barriers from tornado missiles and winds. The design should be optimized to provide this protection in the most dependable and cost-effective way.

4.6.2.5 Zone I Ventilation. Zone I ventilation is the other half of the Zone I confinement system. Since this zone is expected to be contaminated, this function is essential and is classified as Safety Class 1 and Q-Listed.

4.6.2.6 Zone II Ventilation. Zone II confinement ventilation may only occasionally be contaminated and at much lower levels than Zone I. If it is contaminated, it suggests that some breach of the Zone I has occurred. For that reason and to provide defense in depth, the Zone II confinement ventilation function is classified as Safety Class 1 and Q-Listed. Both the primary and secondary ventilation systems must have tornado protection provided to ensure that pressure differentials generated by the tornado neither damage the safety class portions of the system nor cause unacceptable pressure differential changes that could lead to the spread of contamination.

4.6.2.7 Zone III Ventilation. Zone III ventilation will be controlled to provide appropriate pressure differentials with respect to outside atmosphere and Zone II areas. Since it supports the Safety Class 1 functions of the Zone II areas, the Zone III ventilation is classified as Safety Class 2.

4.6.3 Handling and Storage

4.6.3.1 Receiving Area Cranes. The cranes used for handling casks containing wastes (or empty casks) will be regarded as Safety Class 2. These cranes will be classified to be similar to the cranes used at individual waste source locations to load casks onto rail cars and truck beds. In most cases, heavy lift crane standards as used nationally should be sufficient and will meet U.S. Department of Transportation regulations.

Radioactive material casks for over-the-road shipment are qualified for accidents such as falls, impacts, and fires. The design criteria for the facility will ensure that the risks to and from casks while in the facility will be substantially less than comparable risks while the casks are in transit. Therefore, cask qualification will remain valid to protect the waste until it is removed from the casks into the hot cells. The facility will be designed to minimize the distance that the cask will have to be moved from the trailer or car. Maximum height of lift will be strictly limited to be as low as possible. Physical arrangement of unloading facilities will minimize the possibility of one cask being dropped on another.

The principal hazards of concern are failures that could result in dropping a cask, resulting in cask breach or injury to handling personnel. Fail safe features such as deadman switches, overtravel limits, and braking actuation upon loss of power will be utilized to ensure crane safety. The cranes and their support structures will be seismically qualified to ensure that heavy cranes could not fall on casks in the event of a design basis earthquake. Similarly, the building design will ensure that other design basis events such as floods, fires, and tornados could not cause the cranes to fall on the casks.

4.6.3.2 Waste Handling Shaft Cranes. Cranes used to transfer the underground transport casks between transporters and the waste hoist will be carefully designed to ensure that they cannot cause excessive damage to waste container, casks, hoists, or confinement barriers. Therefore, these cranes may require special design provisions to ensure that they cannot significantly damage casks in the event of malfunction. However, the present design of the waste container includes steel walls that are 7.6 cm (3 in.) thick, providing substantial resistance to breaching events. It is expected that this container and its surrounding cask are robust enough to withstand abuse.

Since the waste container and cask are Q-List items, they will be fully qualified for handling activities that will include normal handling operations, anticipated failures, unlikely events, and very unlikely events. At this time, it is expected that these containers and casks can be qualified by assuming normal industrial crane safety provisions. If this proves to be difficult, safety requirements on the cranes may require upgrading at a later time.

Since failures of the crane will not cause a release of radioactive material to the environment, the transfer cranes are classified as Safety Class 2.

4.6.3.3 Hot Cell Cranes. Hot cell cranes and hoists will be used to move waste and waste containers around in the hot cells for packaging and processing operations. Principal hazards are mechanical damage to fuel elements and the release of radioactive material to the cell. Risks include the possibility of damaging waste and containers such that radiation is spread into the cell, breaching the hot cell confinement barrier due to impacts of cranes or loads, or criticality induced by dropping waste material on storage arrays.

It will be a design goal and operating practice to maintain contamination of the hot cells at the lowest level consistent with practical processing needs. However, it is assumed that the cells may be substantially contaminated, and that facility systems will be designed to fully accommodate substantial cell contamination. Therefore, a contamination event resulting from dropping waste in the cell will be fully within the design scope of the facility. For this reason, preventing such contamination is regarded as a Safety Class 3 (ALARA) concern.

The hot cell boundaries will be designated as confinement boundaries and classified as Safety Class 1. An essential part of this designation is to ensure structural integrity and limited leakage for a wide range of normal and accident conditions. Design requirements for these hot cell walls will include evaluations of crane malfunctions and operator errors that could lead to impacts on cell walls, floors, and ceilings. Criteria will be developed that minimize the number of penetrations that could be affected by drops and impacts, provide overtravel and overspeed limit devices, and provide other protective features. As with most overhead heavy masses, the in-cell cranes and hoists will be structurally qualified to resist design basis earthquakes and other design basis accidents. It is expected that the evaluations and design controls used to protect the confinement barriers will be sufficient to maintain safety without listing the in-cell cranes and hoists as Safety Class 1.

Criticality prevention within hot cells has been listed as a Safety Class 1 function. Normal as well as accident conditions are evaluated to ensure that nuclear criticalities are adequately prevented. The potential for waste to be dropped or otherwise mishandled in the presence of other waste packages will be considered in the criticality analysis. Waste storage locations will be sufficiently robust that dropping waste containers

on them will not result in collapse. Using these and other criteria, it is not necessary to classify the in-cell cranes and hoists as Safety Class 1; instead they will be classified as Safety Class 2.

These hot cell cranes will be structurally qualified for design basis accidents and will have suitable interlocks and other safety devices and controls to minimize risks. In addition, to minimize the maintenance frequency for in-cell cranes, high reliability requirements will be imposed on these cranes. This has a secondary benefit of improving the reliability for safety considerations.

4.6.3.4 Waste-Handling Equipment. A preliminary evaluation was made on the equipment to be used to handle high-level waste. In early phases, this equipment is relatively simple and is used to place waste fuel elements into waste containers. In later stages, more complicated equipment may be used to consolidate the waste by cutting non-fuel portions of the waste away from the waste itself. Malfunctions of the cutting equipment could result in fuel being damaged with significant releases of radioactive material to the hot cell. It is expected that these operations will result in significant contamination inside the hot cells, even though good design practices should be used to reduce such contamination. On this basis, cell atmosphere processing equipment should be provided to intercept releases of noble gases, halogens, volatiles, and particulate. If such equipment is provided, inadvertent damage to fuel portions of the waste could be readily accommodated. With this assumption, the waste-handling equipment is classified as Safety Class 2.

4.6.3.5 Criticality Control Systems. This event assumes that an inadvertent nuclear criticality occurs in the repository surface facilities due to process mishaps, equipment failures, or operator errors. It should be noted that spent fuel is generally less reactive than fresh fuel due to burnup of fissile material and creation of neutron poisons by the fission process. In addition, almost all of the fissile material handled in a waste repository is solid; this avoids some of the complications of material handling and analysis that arise when processing liquid fissile material.

This event is considered unacceptable and specific design requirements are included in 10 CFR 60 regulations (NRC 1983) governing repository design. Because of this, the current analysis has assumed that design provisions will reduce the probability of occurrence of this event below the threshold probability being used to screen initiating events in (the subsurface portion of) this study. In addition, a safety analysis will be performed during repository design to ensure that a nuclear criticality is not credible. Because of the above considerations, the consequences of nuclear criticality have not been explicitly analyzed in this study, although some criticality-related aspects of repository design have been placed on the Q-List.

Criticality control features and activities should be regarded as Safety Class 1 and thus on the Q-List at this time. At a later time, it may be possible to establish that the offsite consequences are not severe enough to warrant such a classification, and criticality control may be redesignated as Safety Class 2.

The primary controls preventing criticality in the hot cells are expected to be storage racks that will maintain stored waste in criticality-safe configuration for a wide range of accidents, including inadvertent introduction of moderation. Secondary controls are expected to include monitors and administrative limits that prevent handling of an excessive amount of waste outside the storage positions.

4.6.3.6 Cask Parking Storage. Over-the-road and rail waste casks arriving at the repository will be parked in a yard on their vehicles prior to being brought into the waste handling building. The yard will be designed to provide good separations from other casks and from significant hazards, as well as to provide good physical security. In this status, the casks are considered to be still within the transportation system.

When casks are brought into the Waste Handling Building, it will be a part of the process operation to decontaminate, prepare, and unload casks from their vehicles. While it is not planned to store many of the casks in the building, it is expected that space for a small number of casks may be provided to permit flexibility in handling operations. It is recommended that a principal safety design criterion be developed to ensure that casks can be safely offloaded and brought to their respective processing stations without transferring them over stored casks. This will minimize the danger of dropping one cask on another. Assuming that design layout can meet the above requirements, the cask storage system is classified as Safety Class 2.

4.6.3.7 Hot Cell Lag Storage. High level waste will be transferred from over-the-road casks into the hot cells for handling and processing. An over-the-road cask will hold three to four times as much material as can be placed into a waste container. As part of the container loading and closure process, individual containers may become defective and have to be reprocessed. Throughput requirements may necessitate some parallel processing operations. For these reasons, lag storage for high-level waste within the hot cells will be provided. However, cell layout, process design, and scheduling will be used to minimize the need for, and size of, such hot cell lag storage.

Waste stored in an array within the hot cells in lag storage locations is a potential target for accidents such as the dropping of heavy loads on the lag storage area by hot cell crane malfunction or operator error. It is expected that the lag storage location can be made robust enough to resist such accidents with little radioactive material release. Such releases in the contaminated environment of the hot cell should be expected and should be part of the hot cell design criteria. Criticality control features will ensure that such accidents will not cause a nuclear criticality. For these reasons, lag storage in the cell is classified as Safety Class 2.

4.6.4 Auxiliary Systems

4.6.4.1 Hot Cell Fire Detection and Suppression. A fire detection and suppression system for the hot cells to protect Zone I and Zone II

confinement barriers and ventilation functions is assumed to be necessary. A key design criterion should be to limit the inventory of combustible and flammable materials in and around the cells. However, processing necessities may require the use of substantial quantities of these materials, and a design basis fire is assumed for evaluation purposes. The fire detection and suppression systems should be classified as Safety Class 1 and Q-Listed, because a fire could both liberate dispersible radioactive material and damage confinement functions.

4.6.4.2 Heat Removal System. If substantial heat is developed in the design basis fire that elevates the temperature of the exhaust flow to the main HEPA filters, a heat removal system would be needed to protect these filters. This has been assumed for this review, and the system classified as Safety Class 2. Safety Class 2 is considered to be a high enough safety class for this system, since it is a third line of defense after fire prevention and fire suppression. Analysis may show that with a sufficiently small design basis fire, that exhaust temperatures would not become excessive, and the filters would not be challenged. In this event, the system might be deleted from the design or downgraded to Safety Class 3.

Note that this system differs from the hot cell cooling system listed below, which is designed to remove the normal heat load from the cell due to waste decay heat, lighting, and equipment heat. The heat removal system is to be provided for fire conditions.

4.6.4.3 Emergency Power System. An emergency power system is essential to provide electric power to the main exhaust fans and it is therefore a Safety Class 1 and Q-Listed item that is required to provide a primary safety function (confinement exhaust flow). It is also expected that Safety Class 1 monitoring and control functions for confinement and other functions may also require assured power. However, these latter functions would likely require much less power, which might be furnished from local sources such as batteries and may not need central power systems. It is assumed that this emergency power system would be qualified to Institute of Electrical and Electronics Engineers (IEEE) 1-E requirements and would be very similar to nuclear power station installations. The emergency power system is classified as Safety Class 1 and Q-Listed.

4.6.4.4 Uninterruptible Power System. A separate system (which also provides emergency power) made up of batteries and chargers is assumed. This system provides uninterruptible power to certain key monitors and controls. The emergency generators would require a few minutes to start, come up to line voltage, and receive sequenced loads. The Uninterruptible Power System (UPS) would provide continuous power during this period for confinement monitoring and effluent monitoring. This system is assumed to be essential and therefore is classified as Safety Class 1 and Q-Listed. Detailed design evaluation may show that only Safety Class 2 and lower functions need to be supported. In this case, the system would be removed from the Q-List.

4.6.4.5 Hot Cell Cooling Systems. The anticipated heat loading in hot cells should be estimated as early as possible to establish the degree of cell cooling that may be needed during normal and accident conditions. The principal cell heat load is expected to be waste decay heat, followed by heat from lights and equipment. It is possible that this heat could be removed simply by high airflow into the confinement system. However, this implies a high flow rate, which implies that large openings, large fans, and large filter assemblies would be needed for the primary confinement system. These requirements are contrary to the requirement for small throughput to minimize suspension of particles by airflow and to minimize the size of openings in the cell walls to limit radioactive material escape paths. Nevertheless, adequate exhaust must be available to control heat and combustion product buildup in the event of a cell fire.

The use of in-cell coolers may be preferable to trying to control heat only by exhaust flow. However, if these coolers employ liquids, they may represent a criticality hazard if leakage of liquids into waste storage spaces were to occur. This provides additional incentives to provide criticality control designs that can tolerate optimal moderation and reflection.

It is important to assess the consequences of heat buildup in the cell, especially following design basis accidents that could lead to loss of power and cooling function. If the resultant heat buildup could lead to a loss of cell differential pressure (or to the breaching of seals), then confinement functions would not be satisfied. If the cooling function becomes a Safety Class 1 requirement, this implies adding many systems to the Q-List and adding substantial capacity for the emergency generators. Both of these are important impacts on the safety provisions in the design and should be explored as soon as possible.

Unless the cooling function is determined to be essential, the cooling systems are classified as Safety Class 2.

4.6.5 Monitoring and Control

4.6.5.1 Effluent Radiation Monitors. Effluent radiation monitors will be required on the plant to monitor normal and accidental releases to the environment. This system is classified as Safety Class 1, and Q-Listed since it provides essential information about hazards to the public. If releases are above limits, the system will acuate automatic control and/or alert operators to take recovery actions. Only the portions of the system that monitor for accident level releases need be Q-Listed. Portions monitoring for normal or occupational release levels will be categorized in lower safety classes.

4.6.5.2 Seismic Monitors. It is assumed that much of the plant will be seismically qualified for UBC Zone II accelerations or the design basis earthquake. Q-Listed items will by definition be Seismic Category I and qualified to resist the design basis earthquake. However, there will be

portions of the plant with no seismic qualification, and some levels of earthquakes may do substantial damage. In the event of an earthquake, it is important for the operators to be able to assess its strength and therefore be able to take the appropriate actions. These decisions are related to safety, because immediate cessation of operations and evacuations of plant personnel may be required. For these reasons, the seismic monitors are classified as Safety Class 2.

4.6.5.3 Confinement Function Monitoring and Control. The key elements in preventing releases of radioactive materials rest on material containment (essentially leak-tight passive barriers) and material confinement (a dynamic activity requiring barriers and differential pressures with filtered flow). It is assumed that monitoring and controlling the confinement ventilation functions is essential, and it is therefore Q-Listed. In a sense, this does not necessarily represent a separate Q-List item in that it can be seen as an essential extension of the Q-Listed Zone I and Zone II ventilation. If loss of function or excessive effluent release is detected, the confinement system should have qualified exhaust system controls to correct the problems. The confinement function monitors and controls are classified as Safety Class 1 and Q-Listed.

4.6.5.4 Control Room. It is assumed that a qualified control room that contains the needed monitoring indicators and controls will be necessary to verify safe operations of the repository, diagnose off-normal conditions, and plan and direct recovery from accidents. To protect these vital functions, the control room is classified as Safety Class 1 and Q-Listed.

At this stage of design development, functions that need to be controlled under off-normal conditions are not completely defined. However, monitoring and control that will be needed does include monitoring for excessive radioactive effluent, operation and control of exhaust fans and HEPA filter banks, as well as cessation of handling and emplacement operations. Further definition in this area should be an objective of the ACD.

4.6.6 Structures

4.6.6.1 Waste Handling Building. The Waste Handling Building has been placed on the Q-List because much of the building will contain the Q-Listed Zone I and Zone II confinement barriers and ventilation systems. In many cases, the integrity of these confinement barriers depends directly on the structural integrity of the main building. Close analysis of the entire building will show that there are Zone III areas that do not need to be protected and certain elements that may not need to be seismically qualified since their failure would not have serious consequences. When these details become available, appropriate component breakdowns can be used to focus on only those building elements that are needed for the Q-List. However, at this time, the entire building is classified as Safety Class 1 and is Q-Listed.

4.6.6.2 Emergency Generator Buildings and Uninterruptible Power System Building. The emergency generator buildings house the systems needed to furnish emergency AC power to both vital surface and vital underground facilities in the event of a loss of main preferred offsite power (and any alternate offsite power). These buildings are not needed to provide direct protection to radioactive materials, but they provide protection to a very important safety system. The main function of the buildings is to provide seismically qualified support to the generators, auxiliaries, cabling, panels, and controls. Another important function is to protect the safety class systems inside from tornadoes, missiles, and severe weather effects. Finally, maintaining access to the generators for startup, operation, switching, and maintenance may also be important.

Early criteria discussed for safety classification of structures assumed that the generator buildings would not have to be classified as Safety Class 1 as long as appropriate Safety Class 1 functions were provided. As long as the hazard of radioactive contamination of the emergency generator areas could be minimized, it may be advisable to include the emergency generators as part of the Q-Listed Waste Handling Building. If this is selected as an option, it would be advisable to locate the individual generator facilities in widely separated areas to maintain independence and provide protection against common-cause failures or challenges. On the basis that the safety functions needed to preserve the emergency power safety functions are features listed for the emergency power system, these buildings are classified as Safety Class 2.

4.6.6.3 Future Q-Listed Structures. At the current conceptual stages of design, only preliminary assessments of the need for Q-Listing structures and portions of structures can be made. As the advanced conceptual design proceeds and the Q-Listing of systems and components is made more complete, requirements for Q-Listing supporting and housing structures will become more evident. It is expected that future versions of the Q-List will have additional structural elements in addition to the Waste Handling Building.

4.6.7 Components

4.6.7.1 Hot Cell Transfer Port Features. When waste is moved into or out of the hot cells, it is essential to maintain containment and confinement of the waste. It is assumed that the waste container is not robust enough to resist all DBA's at this critical transfer point and, therefore, the transfer port features are classified as Safety Class 1 and Q-Listed. When casks are mated to the cell port, the waste is vulnerable (e.g., earthquakes could impose shearing forces and moments that could both damage the waste and breach the Zone I barrier). These transfer port features are key elements of the Zone I barrier, since it is clear that interfaces with other key systems (casks) and hazard potentials need to be factored into the design.

4.6.7.2 Missile Shielding. It is assumed that there are design basis accidents that can produce missiles (equipment overspeed, pressure system rupture) capable of damaging waste packages or disabling Safety Class 1

systems. While good design practice may eliminate many such sources, it is expected that some will remain. Similarly, key design criteria should be used to require separating as many targets from missile sources as possible. It is recommended that missile shields become part of the protection package as established by careful design reviews of significant missile hazards. The missile shields should be classified as Safety Class 1 or 2, depending upon the nature of the risk involved and the safety class of the item to be protected.

4.6.7.3 Future Q-Listed Components. The present Q-List effort has focused primarily on safety systems due to the conceptual nature of the design, particularly for surface systems. As the ACD is developed, efforts should focus on identification of Q-Listed components. Most of these components will be the constituents of Q-Listed systems, but some may be Q-Listed features and components that will be outside of Q-Listed systems. Examples of these might include isolation valves, separations barriers, and tornado missile shields.

4.7 Q-LIST SUMMARY

The Q-List for surface facilities is presented in table 5. The scope of the list includes those systems necessary to receive, store, process, package, and transport radioactive waste to the waste handling shaft. It does not cover systems that provide underground functions. These underground functions are developed in the subsurface Q-List, and it is intended that surface extensions of these facilities (e.g., hoist head frame, underground confinement exhaust) will be evaluated for inclusion on the subsurface Q-List.

Table 5. Q-List for Surface Systems and Structures.

Systems	Safety function
High-level waste containment	
Shipping casks for waste materials[a]	Radioactive material containment
Waste container	Radioactive material containment
Underground transfer cask[b]	Radioactive material containment
Radioactive material confinement	
Primary confinement barrier	Radioactive material confinement
Zone I ventilation	Radioactive material confinement
Zone II ventilation	Radioactive material confinement
Handling and storage	
Criticality control systems	Maintains criticality control
Auxiliary systems	
Hot cell fire detection and suppression system	Protects Zone I and II confinement
Emergency power system	Provides power for monitoring, control, and confinement
Uninterruptible power system	Provides power for monitoring and control
Monitoring and control	
Effluent radiation monitoring (accident)	Monitors releases to environment
Confinement function monitoring and control	Ensures vital confinement functions are being maintained
Control Room	Monitors and controls facility processes
Structures	
Waste Handling Building	Protects confinement and control room

[a]This is an important system that will be furnished by others as part of the high-level radioactive waste transportation system. It appears on the Q-List because it involves an interface with external organizations that needs to be carefully controlled.

[b]This is an underground system with important surface interfaces. PST87-2155-3

5.0 SUBSURFACE FACILITIES IMPORTANT-TO-SAFETY Q-LIST

Actions related to subsurface structures and facilities that potentially impact preclosure safety will be taken during the site characterization program. The methodology used for determination of the subsurface facility preliminary Q-List is based on a quantitative assessment of the radiological consequences of a selection of possible accident sequences. Because of the conceptual nature of the current design of repository facilities and the lack of design detail in a number of areas, this quantitative analysis required a number of assumptions and involves significant uncertainties. This report attempts to explicitly describe the design assumptions (in section 3.2.2). The uncertainties have been dealt with implicitly by incorporating factors of conservatism in the definition of the threshold screening probability and the threshold radiological consequence. In addition, estimates of the radiological source terms and physical transport mechanisms have been conservatively based on earlier analyses of similar accidents and related experimental data.

The OCRWM methodology for the Subsurface Facilities Important-to-Safety Q-List, described in section 3.3.3, requires that a collection of potential subsurface accident initiators be evaluated to determine those which are "credible" (i.e., those whose estimated probability of occurrence is greater that the threshold screening probability). This credibility screening is described in section 5.1.1. Subsequently, the radiological consequences of the credible accident initiators are evaluated under the assumption that no mitigative or preventive systems are in place. Those accident sequences for which the unmitigated consequences are less than the threshold screening consequence level are eliminated from further consideration. This threshold screening process is described in section 5.1.2.

For the accident initiators that remain after the probability and consequence screening steps, event trees are constructed that take into account the possible success or failure of mitigative or preventive repository systems. The OCRWM methodology requires that all accident sequences (i.e., all event tree branches) resulting from the event tree analysis be either "incredible" or "inconsequential." To put it another way, additional level of prevention or mitigation must be provided to eliminate all of the possible sequences remaining after the unmitigated accident screening. Finally, those preventive and mitigative systems whose successful functioning was responsible for the elimination of any of the remaining sequences must be put on the Subsurface Facilities Important-to-Safety Q-List. The event tree representation of the accident sequences is described in section 5.2. The estimation of probabilities for event tree branch points is treated in section 5.3, while the results of the consequence analysis for mitigated accident sequences are presented in section 5.4. Finally, the Subsurface Facilities Important-to-Safety Q-List is summarized in section 5.5.

5.1 PROBABILITY AND CONSEQUENCE SCREENING OF UNMITIGATED ACCIDENTS

5.1.1 Initiating Event Identification and Screening Probabilities

A set of initiating events was considered which included both external and internal events. The events selected for evaluation present differing challenges to the waste package and safety systems as well as enveloping lesser challenges. Potential initiators that are enveloped by other initiators are discussed. In some cases, the probability of the event occurring is developed, while in other cases the bases for engineering judgment have been discussed.

Potential repository accident initiators considered for evaluation are given in table 6.

5.1.1.1 Earthquakes. There are two major issues to be dealt with in the consideration of seismic events as possible accident initiators at a repository in basalt. First, what will be the response of the repository to a design basis earthquake? Second, what are the probability and likely consequences of a beyond-design-basis seismic event which results in the creation of a new fault? The second question is important because the available evidence, although somewhat anecdotal, indicates strongly that subsurface damage from earthquakes is almost entirely associated with relative ground motion along the fault itself, imposing shearing and/or compressive loads on the underground structures. This fact together with the implicit commitment of the basalt repository project to avoid capable faults during the development of the emplacements drifts implies that significant seismic damage to more than one waste container requires a seismic event that creates a new fault intersecting a series of emplaced waste containers.

5.1.1.1.1 Design Basis Earthquake. The design basis earthquake for the Hanford Site is a surface wave magnitude 6.5 earthquake occurring on the Wallula-Rattlesnake Fault 5 km (3.1 mi) from the nearest edge of the planned repository. The earthquake is described in a study of seismic hazards at the Washington Public Power Supply System nuclear powerplant WNP-2 (Slemmons 1982) and produces a peak horizontal surface acceleration of 0.25 g at WNP-2 and the DOE Fast Flux Test Facility (FFTF), located approximately 16 km (10 mi) and 8 km (5 mi), respectively, from the proposed basalt repository. Repository designers are using a ground surface acceleration of 0.4 g as the seismic design basis for surface facilities. Seismic ground accelerations decrease with depth and, for the 975-m (3,200-ft) depth of the drifts and emplacement locations, the peak horizontal acceleration assumed for design purposes is 0.2 g.

Lawrence Livermore Laboratory has produced a seismic hazard model for DOE sites (Coats and Murray 1984). For the Hanford Site, the recurrence period for a 0.2-g earthquake is 10,000 yr. For a 0.1-g earthquake, the recurrence period is 700 yr. Slemmons (1982) estimated a recurrence period of 50,000 yr for a magnitude 6.5 earthquake occurring near the repository along the Rattlesnake-Wallula alignment.

Table 6. Potential Repository Accident Initiators.

Natural Events
*1. Earthquake
1.1 Design basis earthquake level
1.2 Earthquake fault creation
2. Tornado
3. Flood
3.1 Surface flood
3.2 Groundwater intrusion into repository
3.3 Dewatering pond breakout
4. Volcanic ashfall
5. Host rock failures
5.1 Rock bursts
*5.2 Rock falls and drift collapse
*5.3 Shaft failure
Internal Events
*6. Fire
6.1 Transporter fire involving cask
6.2 Diesel fuel tank fire involving cask
*7. Explosion
7.1 Methane-caused explosion
7.2 Explosives--transport to development face
7.3 Magazine explosion
8. Thermal effects
8.1 Loss of drift cooling
*8.2 Areal power density overload
9. Local water leak
10. Loss of electric power
11. Hoist cage events
*11.1 Hoist cage overtravel at headframe during retrieval
*11.2 Hoist cage drop of waste package
12. Radioactive containment breach
13. Radioactive confinement failure
14. Nuclear criticality
*15. Improper construction techniques
*16. Operator error during preclosure operations
*17. Coupled effects
*18. Retrieval of failed waste packages

*These events are listed in Weston (1986, table 4-2).

The primary possible effects of the design basis earthquake include damage to unqualified surface and subsurface systems, loss of main grid electric power, and displacement of or damage to waste packages in the process of being moved or emplaced. Seismic experience in the mining community shows most damage occurs to surface facilities and that their seismic effects are seldom felt or recognized underground even for strong motion earthquakes.

The evidence on earthquake effects from the mining industry is buttressed by the articles in a book by Ariman (1983) on the effects of earthquakes on buried pipelines and tunnels. Measurements by Hamada et al. (Ariman 1983, p. 100) verify reduced accelerations and small liner strains in a tunnel 120 m below ground level. The paper by O'Rourke and Tawfik (Ariman 1983, p. 125) notes that welded steel pipeline in the 1976 Guatemalan earthquake survived displacements of 0.46 m without failure. Singhal and Benavides (Ariman 1983, p. 295) note that

> "Ductility of the pipe material is very important factor [sic] for the seismic resistant design of buried pipelines. Ductility of a few percent may help a piping system follow the movement of the surrounding ground and escape severe damages during even large scale earthquakes. A characteristic example of this conclusion is the proven seismic resistance of welded steel pipes."

The general conclusions of a number of papers in the Ariman book may be summarized as outlined below.

1. Earthquake damage to pipelines is most often due to relative ground movement at fault zones.

2. Ductile pipelines have generally much better survivability than brittle pipelines.

3. For many types of earthquakes, even large fault movements do not impose large shearing forces on underground structures because of correspondingly large fault widths.

4. Structures in bedrock are subjected to smaller displacements than structures in soil.

Finally, a December 11, 1967 earthquake at the Koyna hydroelectric project in the Deccan Trap basalt flows in India had measured surface accelerations on the order of 0.4 g (horizontal and vertical components). The earthquake destroyed many structures in the nearby village of Konyanagar and caused structural damage to the dam itself. The associated underground powerplant is 8 km (5 mi) from the dam with a few hundred meters (note: "several hundred feet" in the original source) of overlying Deccan Trap rock above it. It suffered no damage of consequence, as detailed by Berg et al. (1969):

> "A few fine cracks were found in the walls of the generator room, and the turbines and generators were shaken badly enough to require adjustment of their alignment, but there was no damage."

The large-scale layering of the basalt flows might reasonably be expected to attenuate seismic energy from "distant" earthquakes. In addition, the facts that expected fault movement in earthquakes near the repository is small by comparison with the large-motion earthquakes discussed in (Ariman 1983), and that the waste container is ductile and short (by comparison with pipelines), strongly suggest that the waste packages would survive even an earthquake creating a new fault intersecting a number of waste package locations.

5.1.1.1.2 Earthquake Fault Creation. While most earthquakes occur at existing faults, a potential effect of strong earthquakes is the development of additional faults in rock masses. This event is defined as being the development of a new fault that affects emplaced waste packages. As the site is developed for emplacement, existing faults in the rock will be avoided. Emplacement holes will not be driven where there are existing capable faults, so this event assumes that the earthquake creates a new fault that cuts across several waste emplacement locations. This event is considered because it is a credible subsurface accident initiator that can potentially cause release of radionuclides from more than one waste container.

An estimate for the probability of earthquakes producing new faults was developed for the reference repository location. The calculations supporting this estimate appear in appendix A. The estimated probability, p, for an earthquake that creates a new fault in the vicinity of the repository subsurface facilities is bounded by

$$7.1 \times 10^{-8}/\text{yr} < p < 7.1 \times 10^{-6}/\text{yr}. \qquad (1)$$

As described in appendix A, this range is a 95% confidence interval for the probability, p, assuming a log normal distribution with standard deviation of the log-transformed variate equal to 0.5. This estimate of the probability of new fault creation is in the range bounded by the 4.0×10^{-11}/yr in Bartlett (1977) and the 2.0×10^{-12} new faults per year per square kilometer in Bertozzi (1977) for a salt site in a tectonically stable zone and 4.0×10^{-4} new faults per year for the Nevada tuff repository site (Hunter et al. 1982).

The probability of a new fault being created which intersects several waste package locations is even lower. Displacement along the fault would produce loading and potential distortion of the waste package which could challenge its integrity. It is expected however, that the package would retain its containment integrity. Even if this integrity is lost, the escape of noble gases and halogens to the emplacement room should be quite slow with little thermal or other driving force available. It is not expected that significant particulate material from the fuel would escape into the emplacement drift. Events that damaged the waste package while still in the emplacement drift or other areas during the emplacement process would be expected to produce much more significant releases than this event.

Because of these calculations and the evidence cited in this section and the previous one, a significant radiological accident at the repository due to underground effects of a seismic event is not considered to be a credible event, based on establishing that its probability of occurrence is less than 1.0×10^{-5} at the 95% confidence level.

5.1.1.2 Tornado. Tornadoes are quite rare in the Hanford Site area, and the very few that have been recorded have been of low intensity and duration. Lawrence Livermore Laboratory has developed extreme wind and tornado hazard models for DOE sites (Coats and Murray 1985). A tornado with wind speeds of 67 m/s (150 m/h) has been considered for the Hanford Site. The return period for a 67-m/s tornado at the Hanford Site is 1,000,000 yr (Coats and Murray 1985). Straight winds of up to 36 m/s (80 m/h) are also described. The return period for a 36 m/s (80 m/h) wind is 40,000 yr.

The design basis tornado defined for the reference repository location (based on NRC 1974) has a 85 m/s (190 m/h) rotational speed with a 2.24 to 22.4 m/s (5 to 50 m/h) translational speed. A pressure drop of 10,340 Pascals (1.5 lbf/in^2) in 2.5 s is specified. No return period is specified, but it is probably in excess of 1,000,000 yr by comparison to Coats and Murray (1985).

Tornado effects would primarily be felt at the surface with wind pressure, tornado-generated missiles, and pressure transients affecting unqualified structures. It is possible that main grid power sources would be lost. As far as underground systems are concerned, there is potential for damage to ventilation systems due to tornado-generated missiles and transient underpressures acting on supply and exhaust systems which could damage equipment or cause air flow reversals.

For surface structures, seismically qualified buildings will generally resist tornado winds up to 67 m/s (150 m/h) without additional significant design provisions (Coates and Murray 1985). In addition, it has been assumed that surface ventilating systems (for both surface and subsurface facilities) are equipped with tornado dampers to prevent damage from the transient under-pressure.

This event is not expected to have any impact on the waste hoist or on waste containers in the subsurface facilities and, hence, is not expected to produce any public radiological hazard.

5.1.1.3 Flood.

5.1.1.3.1 Surface Flooding. The repository is located in a dry site that is well removed from surface water channels, rivers, and streams. Natural flooding that occurs from storms, high spring snowmelt runoff, or even dam failure is not expected to affect the site.

The analysis leading to the conclusion that the proposed basalt repository is not at risk from surface flooding on the Yakima or Columbia River is described in detail in Environmental Assessment for the Reference Repository Location (DOE 1986). The environmental assessment also describes

the probable maximum flood on Cold Creek. The proposed repository sits on an alluvial fan of Cold Creek, and the Cold Creek probable maximum flood would crest at 2.3 m (7.5 ft) along the southwestern border of the reference repository location. This event has never been observed, its recurrence interval cannot be theoretically predicted, it is expected to be of short duration, and it would possibly involve repository surface support facilities.

Surface flooding is not considered to involve a significant public radiological safety risk for the repository subsurface facilities because of its short duration, the availability of numerous mitigating actions, and the fact that the waste containers are designed to survive hundreds of years immersed in a saturated environment.

Since natural floods cannot affect the site directly, the most severe effect would be the potential disruption of utilities such as the main power grid, transportation routes, and communications outside the site.

5.1.1.3.2 Groundwater Intrusion into Repository. The repository subsurface facilities are below several aquifers from which groundwater potentially can intrude. Development of the underground facilities will provide for known water sources by appropriate seals and pumping capacity. Therefore, appreciable flooding of the underground facility due to this cause requires a combination of the natural source and failure of the engineering features installed to deal with this potential source.

The design basis groundwater intrusion assumes that the flow top immediately above the repository is inadvertently penetrated as drifts are being developed. It is taken to be 12.9 m^3/min (3,400 gal/min), initially followed by long-term flow of 7.57 m^3/min (2,000 gal/min) (Baker 1985) based on a deterministic hydrologic analysis using homogeneous hydrologic parameters. A more recent analysis (Clifton and Arnett 1985) using spatially heterogeneous hydrologic parameters and a Monte Carlo analysis of the groundwater intrusion indicates a 90th percentile initial flow from the same event of 4 m^3/min (1,050 gal/min) and a 90th percentile flow rate 1 wk later of 0.7 m^3/min (185 gal/min).

The repository dewatering system is designed as two 100% capacity independent systems sized to accommodate the design basis groundwater intrusion. Due to damage to the waste handling shaft liner caused by the falling waste cask/waste container, one of the accident sequences considered (in section 5.2) in the development of the waste hoist cage drop accident involves the assumption that the repository will be exposed to inflow from other, more capable aquifers. This event is more troublesome because the flow rates are higher and it occurs concurrently with significant radioactive contamination of the repository subsurface facilities.

The probability of the design basis groundwater intrusion is estimated to be 1.0×10^{-3}/yr, assuming that good mining engineering practices have been employed in dealing with this threat. Flooding of the underground facilities, if not mitigated, would be expected to interfere with utilities and could hinder efficient ventilation of the subsurface facilities. However, flooding would not damage waste containers and, hence, does not constitute a radiological hazard to the public.

5.1.1.3.3 Dewatering Pond Breakout. During all of the development period and through most of the emplacement period, substantial amounts of water will be pumped out of the subsurface facilities to settling/percolation ponds. Some potential exists for a localized "flood" from a breakout of the water in these ponds.

It is assumed that the dewatering pond design will include features that separate the ponds from critical facilities, placing them some distance away and at a lower elevation. The ponds should be designed so that even a total failure of dikes or earthworks would not affect either surface or subsurface facilities. No probability has been assigned to this event on the basis that the appropriate design criteria and implementation will eliminate this hazard.

5.1.1.4 Volcanic Ashfall. The May 1980 eruption of Mount St. Helens produced an ashfall of 1.7-cm (0.7-in.) depth at the Hanford Site. There are several volcanoes in the Northwest that could deposit ash on the site should they erupt. The design basis ashfall is taken to be 1.2 cm/h (0.5 in./h) for 6 h followed by about 0.3 cm/h (0.1 in./h) for an additional 18 h for a total depth of about 10 cm (3.9 in.) of ash as estimated in Naiknimbalkar (1986).

Glacier Peak, Mount Mazama, and Mount St. Helens have all produced significant ashfalls within the past 12,000 yr, so it is estimated that the return period for the ashfall discussed above is on the order of 6,000 yr.

The principal effects of a volcanic ashfall are interruption of electrical power supplies due to shorting effects caused by buildup of ash on insulators, transformers, and other equipment. It is also difficult to maintain transportation to bring in supplies, personnel, and equipment under these conditions. It might also be difficult to maintain confinement ventilation both for subsurface and surface facilities due to clogging of supply filters and abrasive effects on rotating machinery bearings.

This event is not expected to have any impact on the waste hoist or on waste containers in the subsurface facilities and, hence, is not expected to produce any public radiological hazard.

5.1.1.5 Host Rock Failures. Events in this classification include failures of the rock comprising the shaft walls, drift boundaries, support and emplacement rooms, or emplacement hole boundaries.

5.1.1.5.1 Rock Bursts. Rock bursts are defined as energetic failures of rock walls or roofs where small amounts of rock are forcefully ejected into open volumes. These usually occur as a result of high stress in the rock which suddenly relieves. The rock volume discharged is typically limited to approximately 1 m^3 (35 ft^3) but the rock burst can cause significant damage to utilities and severe injury or fatalities to personnel.

Most rock bursts occur near the development face within a week of excavation of material. A calculation using data on rock bursts in a hard rock mine in the Coeur d'Alene mining district in Idaho leads to an estimate

of 0.08 serious rocks bursts per year occurring away from the repository development face. Incorporating the conditional probability that a loaded waste transporter will be near enough to a rock burst when it occurs for the rock burst to impact the waste container leads to the estimate (see appendix A for the calculation) that probability of rock burst hitting loaded transporter is equal to 1.62×10^{-6}/yr.

The waste containers derive some protection from rock bursts from the heavy shielding that surrounds them when they are in the underground facilities. The containers themselves are designed to withstand the effects of severe blows. Because of this physical robustness and the low probability that the waste container would be exposed to a rock burst, it is extremely unlikely that a radioactive material release would occur as a result of a rock burst. In the unlikely event of a release of radionuclides, there would be little energy or thermal driving force available for significant dispersion of the radionuclides. The consequences of any such release would be bounded by Initiators B and C of section 5.1.2.

Utilities such as dewatering lines, communications, or electrical equipment could be damaged locally by such a rock burst.

5.1.1.5.2 Rock Falls and Drift Collapse. Rock falls and drift collapse are defined as events where a significant portion of the drift or emplacement room roof gives away. A drift collapse is understood to be a larger rock fall. As site characterization and repository development proceeds, an important part of the effort will be to assess the structural adequacy of the drifts. In areas where rock weakness is observed or where rock strength is indeterminate, rock bolts, wire mesh, and shotcrete reinforcement will be applied as needed. In extreme cases, shoring may be installed. These efforts should limit the likelihood of rock falls.

The actual risks in basalts at the Hanford Site will be specific to those basalts and should become better understood during the site characterization program. There are several aspects of the basalts at the Hanford Site and the repository design that lead to the conclusion that rock bursts and falls that can challenge the waste package will be relatively low probability events. Rock bursts and falls will tend to announce themselves; the repository design includes provisions for monitoring of rock stress relief. Basalts at the Hanford Site have a higher strength-to-stress ratio than the rock involved in significant rock burst incidents (DOE 1984, table 6-23). Also, rock bursts tend to occur more frequently in multilevel mines and mines with high extraction ratios, neither of which applies to the basalt repository. Because of the fractured nature of the Hanford Site basalts, rock bursts and falls will tend to consist of a number of relatively small pieces of rock. The steel-clad lead waste transfer cask and the steel waste container comprise a ductile laminated structure that will tend to distribute rock burst/fall forces and deform in a ductile fashion rather than breach. As noted earlier, a waste package breach caused by a rock burst/fall would have little energy available to disperse radionuclides. The consequences of such an event would be bounded by Initiators B and C and would be mitigated by those repository systems that mitigate the releases associated with Initiators B and C.

The estimated probabilities of occurrence of rock falls and drift collapses are given in table 7 (see section 5.1.2). On the basis of the small amount of data currently available on the ground support properties of a basalt repository (Blake 1984), these probabilities are considered to be conservative. In addition, Appendix A details a calculation of the conditional probability that a waste container will be present at the time and place of occurrence of a rock burst or rock fall or drift collapse; that conditional probability is approximately 1.8×10^{-5}.

Local disruption of utilities is expected in the event of rock falls that could lead to a loss of dewatering capability, communications, or electric power. Where the rock fall is sufficient to completely plug up the drift, local ventilation flow would also be stopped or severely limited.

5.1.1.5.3 Shaft Failure. This event is defined as a collapse of shaft walls with significant damage to the shaft liner that breaches the liner or shaft seals. Since the shafts are being drilled rather than blasted, damage to the rock immediately surrounding the shafts will be limited. In addition, if weak areas of shaft wall are discovered during the drilling process, engineering measures will be taken to counteract these effects.

The shafts and the shaft liners will be designed to accommodate the long-term effects of rock creep. A shaft liner failure in the waste-handling shaft could prevent operation of the waste-handling hoist. Liner failures in that and other shafts could lead to groundwater intrusion. Both of these results would require a recovery program, but would not be expected to lead to a breach of waste containers and a hazard to the public. Repository utility service could be disrupted by liner failures; however, this would not be expected to lead to a radionuclide release.

5.1.1.6 Fire.

5.1.1.6.1 Transporter Fire Involving Cask. This event is based on the reference conceptual design in which the transporter is powered by a diesel engine with a 190-L (50-gal) fuel tank. It is assumed that this fuel catches fire and burns completely, with the fire involving a waste transfer cask and waste container.

Using trucking industry accidents data (Bradley 1985; Dennis et al. 1978) we can conservatively estimate the probability of a diesel fire as being equal to 1.2×10^{-9} fires per kilometer (1.9×10^{-9} fires per mile) driven (see appendix A for the details of this calculation). Assuming 2,500 waste emplacements per year with a maximum of 6.4 km (4 mi) driven with the transporter loaded with a waste package (per waste emplacement) the estimated probability of a waste transporter fire involving a waste package is 1.9×10^{-5}/yr. This estimate should be conservative since the trucking statistics it is based on assume no special design provisions or special operator training, and include accidents involving flammable cargos and gasoline-engine trucks. Such a fire would be expected to heat up the waste container if it is on board. Bounding calculations (see appendix A) establish that, even assuming that the total heat of combustion from 190 L (50 gal) of diesel fuel is put into the waste cask and waste container,

Table 7. Probabilities of Accident Initiators.

Initiator	Expected frequency (per yr)	Level of confidence in expected frequency*
1. Earthquake		
a. Design basis earthquake	2.0 E-05	High
b. Quake with new fault through waste-emplacement panel	6 E-08 to 6 E-10	Medium
2. Tornado (241 km/h (150 mi/h))	1 E-06	High
3. Surface flooding		
a. Columbia River		
• 100 yr	1E-02	High
• Probable maximum flood	<1 E-03	Medium
b. Cold Creek probable maximum flood	1 E-06	Low
c. Dewatering pond breakout	1 E-05	Medium
4. Water influx from aquifer	1 E-03	Medium
5. Volcanic ashfall	3 E-04	Medium
6. Rock failure events (hitting cask)		
a. Rock bursts	0.8	Medium
b. Rock falls (rock <45 t (50 tons))	<10	Medium
c. Drift collapse	1	Low
7. Shaft liner failure		
a. Rock creep and hydrostatic	1.0 E-02	Medium
b. Cask drop or other external	1.0 E-05	Medium
8. Fires		
a. Transporter fire	1.9 E-05	Medium
b. Diesel storage tank	1.6 E-07	Medium
9. Explosions		
a. Methane	1 E-03	Medium
b. Explosives transported	3 E-07	Medium
c. Magazine explosion	1 E-04	Medium
10. Loss of electric power		
a. Main grid AC power loss	1.5 E-02	High
b. Loss of all AC power	1.0 E-04	High
11. Hoist events		
a. Hoist cage overtravel at headframe	1 E-04	Medium
b. Hoist cage drop of waste package	2 E-08	Medium
12. Containment loss		
a. Small radioactive containment breach	1 E-01	Medium
b. Large radioactive containment breach	1 E-06	Medium
13. Nuclear criticality	<1 E-07	Medium

*High = Substantiated by data in available literature (clearly applicable to the Basalt Waste Isolation Project (BWIP)).
Medium = Substantiated by related data applied to the BWIP by engineering calculation or engineering judgment.
Low = Estimate based on engineering experience and judgment.

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pressure inside the waste container would only increase by a factor of approximately two and a half, not enough to fail the container. Since the waste container remains intact, this event will not create a hazard to the public. Local interruption of utilities would be expected in the vicinity of the fire. In addition, the ventilation system would have to handle the products of combustion and the resultant heat.

5.1.1.6.2 Diesel Storage Tank Fire Involving Cask. This event is defined as the burning of the underground storage tank of diesel fuel with a waste transfer cask and waste container involved.

By using trucking statistics (Bradley 1985; Dennis et al. 1978) to estimate the number of diesel truck refuelings per year and the number of fires resulting from those refueling, an estimate of the probability of an inadvertent fire occurring during a refueling was developed. In addition, it is relatively simple to estimate the number of repository waste transporter refuelings that will be required each year. Combining these two numbers, an estimate for the probability per year of a fire involving the diesel fuel storage tank and a waste transporter of $p = 1.6 \times 10^{-7}$/yr is obtained (see appendix A for the details of this calculation). This probability is already quite close to the "credibility" threshold screening probability.

As before, the trucking industry data involve no special design provisions or special training, and includes gasoline-engine trucks. Either design provisions making it impossible to fuel the waste transporter while it is loaded with a waste container or strong administrative controls aimed at preventing that from occurring will be established. In addition, it may be possible to locate the diesel fuel storage tank on the development side of the repository and have the fuel pumped through the rock to a fueling station on the confinement side. It is expected that the repository design will include a fire supression system at the diesel storage tank or at the fueling station. The combination of all these factors renders the storage tank/waste package fire scenario incredible.

The effects of this fire, should it occur, on a waste container could be very significant, including a breach of the waste container and volatilization of a significant fraction of the "volatile" radionuclides, particularly cesium and iodine. The heat of the fire and convection currents created could cause much of the material released to be dispersed throughout the atmosphere of the confinement area. The heat and combustion products from the fire could severely challenge the capabilities of the exhaust ventilation system. Ventilation equipment and filters could be damaged or destroyed. All underground systems could be jeopardized by a fire of this size. In addition, this fire could cause some localized host rock failures because of thermal shocking and vaporization of water inclusions.

The OCRWM methodology for determining the repository Q-List requires consequence analysis and consideration of preventive and mitigative systems for engineered systems whose failure could lead to the initiation of a significant accident. On this basis, the consequences of the diesel storage tank/waste package fire are analyzed as Initiator D and the fire event tree.

5.1.1.7 Explosion

5.1.1.7.1 Methane-Caused Explosion. Preliminary evidence suggests that the levels of methane that will be encountered during development are not high, but some methane is expected to be released from the groundwater seeping into the repository during development. If the concentration of methane in the repository atmosphere reaches explosive limits, an explosion would be expected. The repository will be well ventilated so that even if higher than expected methane release rates are encountered, the methane can be kept well below explosive limits. If ventilation is stopped, the repository subsurface facilities should be evacuated, as this hazard would increase over time as methane comes out of solution in groundwater that seeps into the repository. It is estimated that the probability of a methane-caused explosion for the basalt repository is on the order of 1.0×10^{-3}/yr.

Since the explosion would be distributed, its effects are primarily a hazard to personnel. However, an explosion behind a bulkhead could cause missiles to be generated which could cause interruption of utility services. It is not expected that these missiles could penetrate the waste container and, therefore, there is no significant hazard to waste material.

5.1.1.7.2 Explosive--Transport Event. As development work proceeds, explosives would be transported to the development face in 454-kg (1,000-lb) lots. This event is envisioned as an accidental explosion adjacent to a waste transporter.

Strict separations criteria and physical barriers will be in place to separate development activities (including the transport of explosives) from emplacement activities. On no occasion should explosives and waste transporters inhabit the same corridor. This accident could only occur if physical barriers which separate the emplacement and development activities were violated. The incidence of accidental transport explosions in the underground facilities is estimated to be less than 7.5×10^{-4}/yr at a 95% confidence level (see appendix A for details of the calculation).

If an explosion accidentally occurs near a waste container, the heavy bulk of the radiation shielding should provide some protection for the container. If the container is breached in this event, the amount of radioactive material ejected would be comparable to that from a fire. Local disruption of repository utilities would be expected. Since this would be a failure of an engineered system resulting in a significant accident, the consequences of this event are analyzed in the remainder of the chapter as Initiator E and the explosion event tree.

5.1.1.7.3 Magazine Explosion. A maximum of a 2-wk supply of explosives will be stored underground to support the development activity. In the event considered, one of the magazines is assumed to explode. The explosion is assumed to damage the overlying rock to the extent that heavy flooding of the repository occurs.

Explosives magazines will be constructed according to stringent safety regulations which include, among other provisions, strict separations criteria such that several magazines are preferred to a single one. The blasting caps and explosives will be kept in separate magazines, as required by federal and state regulations. These magazines will be located away from fire sources and potentially sensitive equipment. The explosive to be used is water-gel, which is extremely stable. As with inadvertent transportation explosions, magazine design provisions and administrative controls are sufficient that magazine explosions "don't occur." Twenty years of non-occurrence of magazine explosions in United States underground metallic mines provides a 95% confidence level estimate of the true probability per year of an underground magazine explosion of 1.0×10^{-4}/yr (see appendix A for details of the calculation). There is no "good reason" for a loaded waste transporter to be near the explosives magazine in the sense that the explosives magazine is on the development side of the repository. The waste transporter is intended by design of the process flow to stay on the emplacement side, and there are physical barriers between the two (to control ventilation air leakage from development to emplacement side). The probability of the event that a magazine explosion occurs with a waste container near enough to incur significant damage falls belows the "credibility" threshold value. The magazine explosion may, however, be important from a long-term waste isolation standpoint.

5.1.1.8 Thermal Effects.

5.1.1.8.1 Loss of Drift Cooling. This event is not expected to cause any problems. It is planned that the air flow in an emplacement drift be reduced after all boreholes are filled, and the temperature be allowed to rise to an equilibrium level. Thus, loss of drift cooling is part of the design basis for the emplacement drifts and borehole spacing.

5.1.1.8.2 Areal Power Density Overload. This event (one of the minimum set of accident initiators (Weston 1986, table 4-2)) consists of some error of design or construction, which leads to the heat generation load in (an) emplacement drift(s) being significantly greater than the design basis. This is not really an accident initiator, but rather a design constraint on the design of the emplacement drifts and the waste containers. As such, the relevant design activities are already appropriately controlled. In addition, major aspects of the site characterization program will be devoted to confirming the thermal behavior of the host rock and the validity of the thermal design codes. Those activities will be appropriately controlled as Site Characterization Plan activities.

The only likely preclosure safety consequence of a greater-than-design-basis heat load would be an increased tendency to host rock failure. Consequences of this have been discussed in section 5.1.1.5.

5.1.1.9 Local Water Leak. This event assumes that some repository system fails so as to cause localized flooding of the subsurface facilities. Neither this event nor any possible secondary failures caused by the flooding should have any significant impact on the integrity of waste containers. Consequently, there should not be any preclosure public safety impact.

5.1.1.10 Loss of Electric Power. These events can occur as a primary failures, due to repository system failures or offsite events, or as secondary failures following a seismic event, tornado, ashfall, or some host rock failures.

5.1.1.10.1 Loss of Main AC Power Supply. This event is the loss of main grid power from the Bonneville Power Administration grid or a repository system failure that causes the same effect on subsurface AC power. This would cause an interruption of all electrically powered repository functions (except for those control and monitoring functions provided with uninterruptible DC power) for a brief period until emergency AC power supplies came on line and began accepting load. It is not expected that all repository functions will be furnished with emergency power, so some repository functions will be lost for the duration of the outage.

Using historical reliability data (Harris et al. 1985), the probability of a greater than 1-hr loss of both offsite AC power sources is on the order of 0.015/yr.

There should be no impact on the integrity of the waste containers from this event and, hence, no preclosure public safety impact. There is likely to be some loss of ventilation and cooling functions that could lead to secondary failure of other repository systems due to overheating.

5.1.1.10.2 Loss of All AC Power Supplies. This event assumes the event described in the previous paragraph followed by a failure of the emergency AC power supplies. Thus, there would be no motive power available for any repository function (although it is assumed that the uninterruptible DC power continues to function). Ventilation, cooling, and repository dewatering would be lost. There would be a gradual increase in methane levels in the subsurface facilities.

The probability of losing all AC power source for an extended period of time is on the order of 6.0×10^{-5}/yr (see appendix A for details of the calculation).

This event could involve serious risks to repository personnel but should not have any impact on the waste containers and, hence, no impact on public safety.

5.1.1.11 Hoist Cage Events. The fact that the waste-handling hoist is a Koepe hoist implies that certain hoist accidents that are a serious concern with drum hoists are either impossible or much less serious. Specifically, a standard drum hoist accident involves a descending hoist cage hanging up in the shaft with cable continuing to pay out. If the cage finally breaks loose, it can fall at least the length of slack cable that accumulated while it was stuck. This cannot happen with a Koepe hoist when the total friction between the drum and the cable is only sufficient to overcome the difference in weight between the cage and payload on one side of the drum and the counterweight on the other side. If the cage hangs up, the Koepe drum is incapable of lifting the counterweight by itself; without lifting the counterweight, there is no way for slack cable to accumulate on the cage side of the drum.

Similarly, with a drum hoist, if the cage fails to stop at the surface station during its ascent, it is possible for the drum to pull the cage up tight against the headframe of the hoist or against whatever protective structure is underneath the headframe. If the drum is able to exert sufficient force, it may damage the payload or the hoist cables. In the case of the Koepe hoist, the force that can be exerted on the headframe protective structure, the cables, the cage, and its payload is limited to the total friction between the cables and 180° of Koepe drum. By design, this force only needs to be sufficient to overcome the weight imbalance between the counterweight and the loaded (or unloaded) cage. In the case of the waste-handling hoist, that maximum force should not be enough to threaten the structural integrity of either the cage and its payload or of the hoist cables.

5.1.1.11.1 Hoist Cage Overtravel at Headframe during Retrieval. This event is based on the possibility that the hoist malfunctions permit a full cask being brought up the waste handling shaft to continue on past the surface station until it crashes against the protective beams underneath the hoist drum.

As described in the previous section, this event is not the serious concern with a Koepe hoist that it is with a drum hoist. If the Koepe hoist pulls the cage up against the crash beam, the maximum force the Koepe drum can exert on the cage and the crash beam is on the order of 4,893 to 9,786 Newtons (5 to 10 t force). If additional torque were applied to the drum, it would just slip underneath the cable.

This event is a possible cause of a hoist cage drop accident, although the conditional probability of the hoist cage overtravel proceeding on to a cage drop should be quite small, because of the physical limitations previously discussed and the number of ways of terminating the overtravel accident. The limitation on the force that the drum can exert on the cables should prevent a breach of the waste container and, thus, prevent any impact on public safety. The probability of the hoist cage overtravel accident is estimated as 1.0×10^{-4}/yr.

5.1.1.11.2 Hoist Cage Drop. This event assumes that the hoist cage separates from the hoist cables (or because of drum failure both cage and cables fall), falls freely the length of the shaft, and impacts on the bottom at terminal velocity. A fault tree analysis for a similar hoist cage drop accident in a salt repository (Pepping et al. 1981) calculated an estimate of 9.4×10^{-9}/yr for the probability of occurrence of the hoist cage drop. The probability of a similar accident at the Waste Isolation Pilot Project site (Banz et al. 1985) was calculated as 1.7×10^{-8}/yr. The basalt repository waste hoist is expected to incorporate design features justifying roughly equivalent estimates for the probability of a cage drop accident. For purposes of this analysis, the probability of this event was estimated to be 2.0×10^{-8}/yr.

This event is assumed to be hypothetical, in the sense that the cage, cask, and waste container can not reasonably be expected to fall freely the length of the shaft. There is not much clearance in the shaft, long cylinders will not fall stably without tumbling, and the guide structures in

the shaft are not strong enough to keep a 54-t (60-ton) object falling at 90 m/s (190 mi/h) from tumbling. The upshot is that the cask/container could be expected to impact the shaft liner a number of times on its way down the shaft, possibly breaching the liner and possibly causing enough damage to the cask/container to breach it before it reaches the bottom of the shaft. If the shaft liner breached at the level of the most capable aquifers, there could be a groundwater influx into the repository well above the design basis. It is possible that cask debris could be hung up at some intermediate point in the shaft.

While this event is hypothetical, it is also a worst case event that can be used to test the depth of the repository safety systems and can be used to establish worst case radiological consequences. It is judged that the radiological consequences of the hypothetical cask drop will bound all other events, with the possible exception of the diesel storage tank fire involving a waste container. As an example of the failure of an engineered system whose failure can cause a significant accident, the consequences of this accident are analyzed as Initiators A and F in sections 5.1.2 and 5.4.

5.1.1.12 Radioactive Containment Breach. This event is considered to be any significant release of radioactive material due to a breach of the waste container from any cause. This is not really an accident initiator, but rather is a consequence of other initiators treated in this section (e.g., host rock failures, seismic events). Another mechanism for producing a breach of radioactive containment is a failure of equipment or procedures related to subsurface process flow. Examples would be a drop of the waste container while it is being transferred from the hoist cage to the underground waste transporter or too much force being applied to the waste container by the emplacement equipment.

Since the waste container walls are thick because of the requirements for long-term corrosion resistance and it is being designed for impact resistance, an appropriate way to deal with process-related breaches of the waste container is to design the waste transporter and emplacement equipment so that they cannot exert enough force to threaten the waste container.

A range of probabilities would be expected for containment breach, ranging from 1.0×10^{-1} for small breaches to 1.0×10^{-6} for large breaches with substantial releases. The approach used in this analysis is to bound the consequences of all accidents involving containment breach. To this end, Initiators B and C posit a radionuclide release equivalent to that assumed for the cask drop accident, arbitrarily considered to originate at two locations within the underground facilities chosen as the worst case locations for transport to the surface and proximity of the surface release point to the site boundary.

5.1.1.13 Radioactive Confinement Failure. There are two different subsurface radioactive confinement subsystems. The first is the confinement exhaust ventilation filtration subsystem, and the second is the aspect of subsurface ventilation design that ensures that leakage is always from the development side of the subsurface facilities towards the confinement side. The filtration subsystem ensures adequate removal of radionuclides that might be present in the confinement exhaust because of a containment

failure. The controlled leakage feature ensures that radionuclides from the confinement side do not leak to the development side and escape to the surface through its unfiltered exhaust.

This event is also not, strictly speaking, an accident initiator. A failure of subsurface radioactive confinement will not cause any public radiological exposure, unless there is a failure of subsurface radioactive containment prior to repair of the confinement failure. Estimates of the probability of failure of the filtration subsystem enter this analysis as event tree branch point probabilities. They are described in section 5.3. The current design of the subsurface ventilation system provides robust assurance of maintaining the proper direction of leakage between the confinement side and the development side. On the development side, all of the ventilation fans are supply fans; on the confinement side, the fans are exhaust fans. The fans, by design, cannot be reversed. Thus, even with the failure of any subset of the fans, the differential pressure between development and confinement sides will still provide leakage in the proper direction. If some of the permanent or temporary barriers between the two sides fail, the result will be increased leakage flow but still in the right direction. Ventilation system failures could cause a variety of problems, but the assurance of proper direction of leakage should be quite robust. Since a sequence of events and systems failures resulting in a bypassing of radionuclide contamination through the development side has not been identified, this has not been incorporated into the event trees.

5.1.1.14 Nuclear Criticality. This event assumes an inadvertent nuclear criticality occurring subsequent to a subsurface accident initiator or subsequent to waste containment failure. The radiological consequences of this event have been evaluated. However, either version of the event is considered to be unacceptable and specific design requirements are included in 10 CFR 60 regulations governing repository design. Because of this, this analysis has assumed that design provision will reduce the probability of occurrence of this event below the threshold probability being used to screen initiating events in this study. In addition, a safety analysis will be performed during repository design to ensure that a nuclear criticality is not credible, either during surface processing of the waste, as a sequel to some subsurface accident, or as a long-term consequence of waste package deterioration. Because of the above considerations, nuclear criticality has not been incorporated in this study, either as an initiating event or as an event tree branch point, although some criticality-related aspects of repository design have been placed on the Q-List.

5.1.1.15 Improper Construction Techniques. This was not considered to be an accident initiator during this analysis. For systems and subsystems judged important to safety and placed on the Q-List, both design and construction will be subject to appropriate QA controls. For other systems the QA controls used will be appropriate to the importance of the system to occupational safety and to achieving of programmatic goals. Several site characterization activities are intended to develop information about optimal subsurface facility design and construction.

Since the estimates of the probability of occurrence of other accident initiators are based, either qualitatively or quantitatively, on industry experience with similar events in the past, they implicitly incorporate the impact of improper construction techniques upon the occurrence of those past events. In addition, Initiators B and C are intended to bound the consequences of any credible accidents caused by improper construction techniques.

5.1.1.16 Operator Error during Preclosure Operations. As with improper construction techniques, operator error was not considered in this analysis to be an initiating event. It was considered to be one of the possible conditions affecting the probability of occurrence of the "primary" initiating events that were analyzed. It is considered appropriate and desirable to apply human factors design principles to repository design in order to minimize the contribution of operator error to the occurrence of accident initiators. As before, historical system failure data and statistics on the occurrence of accident initiators are considered to incorporate the impact of operator errors. Insofar as historical experience reflects designs that placed less emphasis on human factors than will be the case with repository design, it could be expected that the resulting failure probability estimates will be conservative. In addition, Initiators B and C are intended to bound the consequences of any credible accidents caused by operator error.

5.1.1.17 Coupled Effects. This is also considered to be misplaced as an accident initiator. The phrase coupled effects denotes the tendency of a primary accident initiator to cause secondary failures whose impact must be taken into account in evaluating the consequences of the primary initiator. This analysis has attempted to do that, either by incorporation of the secondary failures within the event tree (with explicit assignment of conditional probabilities of occurrence) or by simply assuming occurrence of the secondary failures as a worst case analysis of the consequences of a particular initiator.

5.1.1.18 Retrieval of Failed Waste Packages. For purposes of this analysis, it has been considered that the safety of retrieval of a failed waste package is adequately covered by one or both of the following:

- The analysis of items important to safety for normal emplacement
- The commitment that the design and fabrication of equipment for special retrieval (i.e., retrieval of failed waste packages) will be Q-Listed.

As rationale for this approach, all of the items important to safety for the normal emplacement of waste containers will be equally important to the safe retrieval of a failed waste container. In addition, the commitment to Q-List the design and fabrication of special retrieval equipment requires a careful consideration of the adequacy of the surrogate containment provided by that equipment. That consideration should include a safety analysis that imposes public safety criteria in the use of the special retrieval equipment at least as strict as the criteria used for determining Q-List items.

5.1.2 Unmitigated Consequences Screening of Accident Initiators

The accident initiators chosen for radiological dose assessment were as follows:

- Initiator A, cask drop accident
- Initiators B and C, accident occurring within the repository drifts resulting in a radionuclide release equivalent to the cask drop
- Initiator D, underground diesel storage tank fire enveloping a waste package
- Initiator E, underground explosion in close proximity to a waste package
- Initiator F, a cask drop accident into water with radionuclides being pumped to the surface by the dewatering system.

Of these, Initiators A, D, E, and F are analyzed (even though their estimated probabilities of occurrence are approximately equal to the threshold screening probability), because they represent failures of engineered systems that can cause significant accidents. Initiators B and C do not represent any identifiable repository accident; rather, their consequences found the consequences of repository accidents involving a breach of a waste container, ranging from high-probability low-consequence accidents to low-probability high-consequence accidents. Estimates of the radionuclide source term for each of these accident initiators are presented in table 7.

The accident initiators for which subsurface and surface source term estimates have been made were chosen using engineering judgment applied to a combination of the accident initiator probabilities and consideration of which accident consequences were likely to envelope the consequences following from other initiators. For each chosen initiator, it was necessary to estimate the subsurface source term and to estimate those physical removal or retardation mechanisms that determine the surface release source term for that accident initiator. Finally the surface source term was used in conjunction with worst case weather conditions based on Hanford Site meteorological data, and the radiological dose to the maximally exposed member of the public was calculated using standard Hanford Site radiological dose assessment codes.

Although the cask drop accident is judged to be an extremely low probability event, its consequences were analyzed using a series of conservative assumptions about the nature of the accident and of the corresponding release. This was done because it was felt that the cask drop accident consequences bound those of all other plausible accident initiators. If the mitigating systems handle the cask drop accident adequately, it is reasonably assured that they will do the same for all other initiators. As a corollary to this approach, accident consequences for some of the accident initiators considered in section 5.1.1 were not

explicitly evaluated, because of confidence that the associated surface radionuclide releases will be less than the hypothetical bounding releases (Initiators B and C) that were analyzed. Also, repository systems that suffice to mitigate the bounding accidents will also suffice for lesser releases. Specifically, Initiators B and C are intended to bound the consequences of all subsurface process-related, rock-failure, and seismic-event-initiated accidents that can result in a breach of a waste container.

The results of the unmitigated dose assessments are reported in table 8 along with the mitigated dose. Briefly put, all of the unmitigated doses were unacceptable, requiring further consideration of mitigating systems in sections 5.2 and 5.3.

5.1.2.1 Bounding Accident Initiators. The accident initiators considered for analysis of unmitigated radionuclide release are a waste-handling shaft waste cask drop accident, a large fire enveloping a waste transfer cask, and a large explosion occurring next to a waste transfer cask. In addition, since the waste cask drop accident produces the largest estimated particulate release, and since it is the particulate release that produces the limiting exposures (to bone and lung) to the maximumally exposed individual at the site boundary, it was arbitrarily assumed that an accident producing the same subsurface release occurred at two other locations within the repository. One of those locations was chosen because its proximity to the confinement side ventilation exhaust shaft might result in large particulates being entrained in the high-velocity exhaust flow and carried to a surface release point. The other location was chosen because its surface release point (the other confinement exhaust shaft) is significantly closer to the site boundary than the waste handling shaft. This location is, however, far enough away from the confinement exhaust shaft that large particulate will drop out of the subsurface flow before it enters the high-velocity confinement exhaust.

These accident initiators are considered to be hypothetical for the reasons outlined below.

1. The waste cask drop accident (Initiators A and F) assumes that a waste-handling shaft hoist cage (or alternately, a waste cask itself) suffers some accident or equipment malfunction that causes it to fall freely the entire length of the shaft and to impact an unyielding structure at the bottom of the shaft with full terminal velocity. A conservative analysis of the spent fuel fragmentation produced by the resulting energy density (i.e., at the moment of impact) was used to determine the particulate release produced by this accident. An equivalent form of the accident involves a waste cask already at the bottom of the shaft impacted by the hoist counterweight falling freely from the top of the shaft.

   The waste-handling hoist will be a Koepe hoist with the cage supported by six cables and the cage maintained in position by two guide rails. The waste transfer cask and waste container are both long cylinders. The guide rails will not be strong enough to maintain the alignment of a 64 t (70-ton) cage/cask falling at 90 to 135 m/s (200 to 300 mi/h). Similarly, experience teaches

Table 8. Surface Release Source Terms Assumed for Analysis of Unmitigated Accident Consequences. (sheet 1 of 2)

Description of initiator	Assumed releases to surface environment
Initiator A: Waste cask drop down the waste-handling shaft	Noble gases: 100% Halogens: 10% Other volatiles: 10% Solid fuel mass released as particles: 2.0 wt% of the fuel mass released as respirable particles (diameter less than 10 μm) The assumed subsurface release of halogens and other volatiles is 20%, with 50% removal during transport to the surface release point.
Initiator B: Cask drop release arbitrarily assumed to occur at a Phase I emplacement location only 60 m (200 ft) from the confinement exhaust shaft R6	The subsurface release is the same as that for Initiator A, with the exception that an additional 9.5 wt% of the solid fuel mass (consisting of particles with diameters between 10 and 150 μm) is assumed to be entrained by the ventilation flow in the drift and the confinement exhaust shaft and thus carried to the surface release point. The surface release point for this initiator is the confinement exhaust shaft R6, which is 2,315 m (7,600 ft) from the nearest point of Route 240.
Initiator C: Cask drop release arbitrarily assumed to occur at a Phase II emplacement location 609 m (2,000 ft) from the confinement exhaust shaft R7	This subsurface release is the same as that for Initiator C, with the exception that the 9.5 wt% in suspendible particles fall out of the ventilation flow before reaching the high flow velocities of the confinement exhaust shaft. Thus, only the 2.0 wt% in respirable particles are carried to the surface release point. The surface release point for this initiator is the confinement exhaust shaft R7, which is 1,116 m (3,400 ft) from the nearest point on Route 240.
Initiator D: A diesel fuel fire causing overpressure failure of the waste container	Noble gases: 100% Halogen: 50% (100% released; 50% lost in transport to the surface) Other volatiles: 50% (100% released; 50% lost in transport to the surface) Solid fuel mass released as particulates: 0.5 wt% (pre-existing respirable particles) The surface release point for this initiator is the confinement exhaust shaft R7.
Initiator E: An explosion causing partial fragmentation of a waste container	The surface source term for this accident initiator is the same as that for Initiator E. The additional mass of respirable solid particles generated by explosion-caused fragmentation is calculated to be significantly smaller than the 0.5 wt% of particles already present in the spent fuel due to thermal cycling, spalling, etc. The surface release point is also shaft R7. Because the plausible locations for this accident are all distant from the confinement exhaust shafts, the suspendible and larger particles are assumed to fall out of the ventilation flow before reaching the high flow velocities of the exhaust shafts.

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Table 8. Surface Release Source Terms Assumed for Analysis of Unmitigated Accident Consequences. (sheet 2 of 2)

Description of initiator	Assumed releases to surface environment
Initiator F: Cask drop into water	This accident initiator assumes a cask drop source term entrained in water and pumped out of the subsurface to the surface retention pond by the dewatering system, without first passing through the dewatering radwaste treatment system. The analysis of this initiator uses reasonable assumptions about radionuclide transport through the unconfined aquifer to the Columbia River, incorporation of radionuclides into the food chain, retention pond evaporation, and wind-borne resuspension and transport of radionuclides. The following subsurface source term with no plateout removal during transport to the surface is assumed: Radionuclides — In air — In water Noble gases — 100% of inventory Halogens — 20% of inventory Volatiles — 20% of inventory Particulates — 50 wt% of spent fuel mass The particulate mass is greater than for Initiators A through C, because it is assumed that larger particles are suspendible in the dewatering system liquid flow than in the confinement exhaust shaft ventilation flow.

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that long cylinders will not fall without tumbling in air unless they have fins, or unless they are spinning on their long axis. Since the cask/container is longer than the shaft is wide, a falling cask will dissipate energy as it falls by striking the sides of the shaft. The result of these collisions with the shaft liner may be a significant level of damage to the waste cask and the waste container, but it is judged that the energy densities produced by these multiple collisions will not produce the amount of spent fuel fragmentation that has been assumed (based on engineering judgment backed up by a literature review) for the idealized shaft drop accident. Even a complete shredding of the cask, waste container, and spent fuel rods would result in spent fuel pellets falling with relatively low terminal velocities (because of much lower mass to cross-sectional area ratio).

In addition to being a physically unrealistic hypothetical accident, because of design features of the Koepe hoist and its control systems and in light of historically good experience with mine hoist systems, the accident is judged to have an extremely low probability of occurrence.

2. The postulated severe fire assumes that the 26,500-L (7,000-gal) subsurface diesel fuel storage tank burns with a waste transfer cask on a waste transporter nearby becoming enveloped in the fire. This postulated severe fire is an extremely unlikely event because of the relatively low flammability of diesel fuel and procedural and design controls that would prohibit the refueling of waste transporters while they are carrying waste containers.

3. Similarly, the postulated explosion assumes that an explosives transfer vehicle carrying 456 kg (1,000 lb) of explosives from one of the magazines to the repository development face explodes while it is next to a passing waste transporter. Indeed, the calculations assume that the explosive was at the surface of the waste cask itself. This accident can be considered to be extremely low probability because, during most of the repository development phase, the explosives magazines, the transport pathways to the development face, and the development face itself are all found on the development side of the repository, while the waste transporters, waste transfer casks, and the emplacement rooms are all found on the confinement side of the repository. The two sides of the repository are strictly separated, thus rendering this postulated accident extremely improbable.

4. Initiators B and C are explicitly hypothetical. They are not tied to the analysis of any identifiable repository accident, but are intended to bound the consequences of all identifiable repository accidents involving the possible breach of a waste container. To that end, Initiators B and C are defined by arbitrarily assuming that the subsurface radionuclide source term estimated for the cask drop accident (Initiator A) is produced at the two locations within the repository subsurface with the potential for maximizing the surface release. Initiator B assumes the greatest transport

of radionuclides to the surface, and Initiator C assumes the surface release point closest to the boundary of the restricted area.

5.1.2.2 Catalog of Conservatisms in the Present Analysis. Because the unmitigated exposures to the maximum exposed individual located at the repository site boundary are significant, it is useful to enumerate some of the conservatisms in the current analysis. In addition, it should be recognized that this analysis is being used solely to establish those repository systems which, because of their potential importance to public safety, are worthy of special attention in the establishment of quality assurance and design requirements.

1. The threshold accident consequence being used in this analysis, 100 mrem dose to the whole body or any organ, is a factor of five more conservative than the 500 mrem required by the DOE-HQ methodology A, which is itself a factor of ten more conservative than the 5 rem specified for other nuclear fuel cycle facilities.

2. The threshold probability being used in this analysis, 1.0×10^{-7} best estimate or less than 1.0×10^{-5} with 95% confidence, is significantly more conservative than the 1.0×10^{-5} required by methodology A. This results in requiring mitigation for rarer accident sequences that would be discarded as incredible using the higher threshold probability.

3. Individuals inhabiting the site boundary of the basalt repository would normally inhabit it at a velocity of 25 m/s (55 mi/h). The nearest site boundary point, 1.6 km (1 mi), is Route 240 as it passes through the Hanford Site. The nearest residences (scattered ranches) are approximately 15 km (10 mi) distant from the repository. The nearest town is 30 km (20 mi) distant.

4. Air velocities in the waste-handling shaft are designed to be approximately 0.5 m/s (1 mi/h) up the shaft. That implies that an accident occurring at the bottom of the shaft would have no impact on the surface for 20 min. An efficient system for closing the stretch of Route 240 passing through the Hanford Site would eliminate much of the public exposure.

5. The current subsurface source terms are intentionally conservative and almost certainly overestimate the actual subsurface particulate source term and underestimate the removal of subsurface source term radionuclides during their long and, in some sequences, tortuous pathway to the surface.

6. The analysis assumes spent fuel with (60,000 MWd/t) burnup, which overestimates the burnup of United States light-water-reactor spent fuel. In addition, a 5-yr cooldown period after removal from the reactor was assumed, which will be a significant underestimate of the actual average cooldown period. Both of these lead to an overestimate of the radionuclide inventory present in the spent fuel. The shorter cooldown period

explicitly overestimates the radionuclide inventory because of continuing radionuclide decay. The assumption of greater burnup biases the radionuclide inventory toward equilibrium concentrations of longer half-life radionuclides.

7. The radiological dose assessment uses 95th percentile worst case meteorological conditions. Since winds on the Hanford Site are most often out of the west, for most surface releases, the wind would blow the release toward the interior of the site and away from Route 240. Sixty to seventy percent of the time the nearest member of the public would be more than 21 km (15 mi) distant from the repository.

5.2 DEVELOPMENT OF EVENT TREES

In the preceding sections, the initiating events were identified and screened on the basis of their credibility of occurrence and potential public radiological impact if accident sequences were not prevented and were unmitigated. For credible accidents, important-to-safety mitigative systems must be provided and described in terms of the way in which the repository responds to the set of initiating events. This can be achieved by using event tree modeling techniques. The following sections discuss the use of the event tree methodology and the repository-specific event trees developed for this technical analysis.

5.2.1 Methodology

The event tree methodology is an inductive process by which accident initiating events are combined with functional successes or failures of repository systems, structures, or components that respond to the accident. These combinations of possible responses form accident sequences that can then be evaluated to determine if the repository systems adequately prevent or mitigate the initiating event. To determine that an accident is adequately prevented or mitigated, all accident sequences must be shown to be either incredible (i.e., the systems are of a high enough reliability that the probability of the sequence occurring is below the threshold probability of 1×10^{-7}/yr best estimate or 1×10^{-5}/yr with 95% confidence) or inconsequential (i.e., the systems operate in such a way that the radiological dose to the public, at or beyond the repository site boundary, is below the consequence threshold of 100 mrem).

The process of developing an event tree consists of the following steps:

- Statement of event tree objective
- Determination of information sources
- Identification of preventive or mitigative functions or systems
- Event tree construction.

The objective of the event tree development is to depict, for a given accident, the possible responses of the repository systems to prevent the breaching of a waste package and mitigate the subsequent dispersion of radioactive material. The event tree must, therefore, display the functional dependences between systems (i.e., cases where failure of one system means that it is not possible for another system to perform its function successfully). An event tree also shows where a system's success or failure has no impact on the radioactive release associated with a given accident sequence (these situations are depicted on the event tree by omitted branch points).

The event tree sequences end in one of three general end states: success, consequence analysis (CA), or probability analysis (PA). Those sequences ending in success represent scenarios in which the waste package is not affected by the accident and, therefore, result in no releases of radionuclides. Sequences with CA end states represent scenarios where all of the mitigative confinement systems function correctly. These sequences require consequence analyses to verify that the mitigative confinement systems are sufficient to limit the radionuclide releases to acceptable levels (i.e., below the consequence threshold value). The consequence analyses are provided in section 5.4. The PA end states represent scenarios where the radionuclide releases are not confined to the underground facility because the mitigative confinement systems fail to function correctly. These sequences could potentially result in excessive levels of radionuclide releases and, therefore, the sequences must be shown to be incredible (i.e., below the probability threshold values). To determine the credibility of a sequence, an analysis of the probability of the sequence must be done. The probability analyses of these sequences are provided in section 5.3.

The source of information on which the event trees were developed was the Site Characterization Plan CDR (KE/PB 1987), augmented where necessary by direct communication with architect/engineer personnel. Because an event tree is an inductive analysis process, based on the current state of understanding about the repository functions and systems, changes are expected in the event tree descriptions. This is particularly true at this early design stage, since the design will be undergoing numerous changes. Upon completion of this task, future design changes can be evaluated and the event trees can be updated as necessary.

Identification of preventive or mitigative functions is one of the key objectives of the event tree development process. These functions are important in protecting the public from radiological doses resulting from accidents within the repository subsurface. More accurately, most functions operate to either prevent or mitigate the breaching of a waste package while it is in the subsurface facility. The preventive functions operate so that breaching does not occur, while the mitigative functions operate to limit the consequences should breaching occur.

During normal repository operating conditions, a waste package breach is prevented by passive or active engineered systems. The repository design

contains a variety of active and passive preventive systems, such as the following:

- Redundant waste-handling hoist control systems (acting to prevent a hoist cage drop accident)
- Fire suppression systems (at the subsurface diesel fuel storage tank area)
- Design features that prevent nuclear criticality
- Waste package container integrity (which provides thick-walled containment against the release of radionuclides into the subsurface facility).

The mitigative functions consist of subsurface radionuclide containment features, limitations on the transport of radionuclides to the surface, and confinement of radionuclides that reach the surface facilities. In the present design, the first two functions are primarily performed by naturally mitigative phenomena. However, active mitigative systems could be provided, if necessary (e.g., shutting off the confinement ventilation exhaust fans). For the third function, the main mitigative systems are currently considered to be the dewatering treatment and ventilation filtration systems. The dose to the public from the dewatering pathway is considered to be only possible in the event of a massive containment breach, as in a waste hoist cage drop accident. For all other accidents, the dewatering pathway is considered not to be a significant contributor to the public dose. The function of confinement ventilation filtration is generally considered adequate to mitigate all accidents of concern that result in airborne releases.

Construction of an event tree is basically the graphical representation of the sequence of events for a given accident, taking into account mitigative and preventive features. The subsurface repository systems respond to accidental radionuclide releases in essentially the same way, regardless of the initiator of waste container breach. The event trees can thus be grouped on the basis of potential general effects on the repository and its systems. The event trees developed on this basis are as follows:

- Waste Hoist-Cage Drop (Initiators A and F)
- Explosion (Initiator E)
- Fire (Initiator D)
- Rock Failure (Initiators B and C)
- Seismic Event (Initiators B and C).

Table 9 provides the initiators grouped for each type of the above event trees. Sections 5.2.2 through 5.2.6 discuss each event tree and resulting end states in detail. The intermediate event probabilities, for each event branch point in the sequences, are discussed in section 5.3.

Table 9. Event Tree Grouping.

Event trees developed	Corresponding initiating events
Waste hoist cage drop event tree	Waste hoist cage drop Waste hoist cage overtravel
Explosion event tree	Magazine explosion Commercial explosion associated with the transportation of explosives to a development face Methane explosion Complete loss of ventilation Complete loss of all electrical power
Fire event tree	Fire associated with a waste transporter Fire associated with a diesel storage tank
Rock failure event tree	Drift collapse Rock fall Rock burst
Seismic event tree	Earthquake

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5.2.2 Waste Hoist Cage Drop Event Tree

The waste hoist cage drop event tree is shown in figure 8 and is represented by individual success and failure responses to the four questions listed below.

- Does a waste hoist cage drop occur?
- Is the integrity of the waste package challenged?
- Is the dewatering pathway treated?
- Is the waste handling shaft ventilation airflow filtered?

The waste hoist cage drop event tree includes a waste hoist cage overtravel incident and a waste hoist cage drop accident. The overtravel incident should have little impact on the waste package due to the low velocities involved and travel limitations on the hoist in order to protect the cables. A worst case event would be for the hoist cage to impact the headframe, break the cables, and then drop down the waste handling shaft (R5), impacting the bottom of the shaft. This results in essentially the same effects as, and is considered to be included in the determination of the probability of, a waste hoist cage drop event. The waste hoist cage drop initiating event is discussed in section 5.1.1.11. The event tree applies to both normal and retrieval operations.

5.2.2.1 Sequence 1. Sequence 1 illustrates a success path (i.e., there is no damage to a waste package). During a waste hoist cage drop event, the most likely way for a waste package not to be damaged is for the waste package not to be on the cage when the drop occurs. This condition would

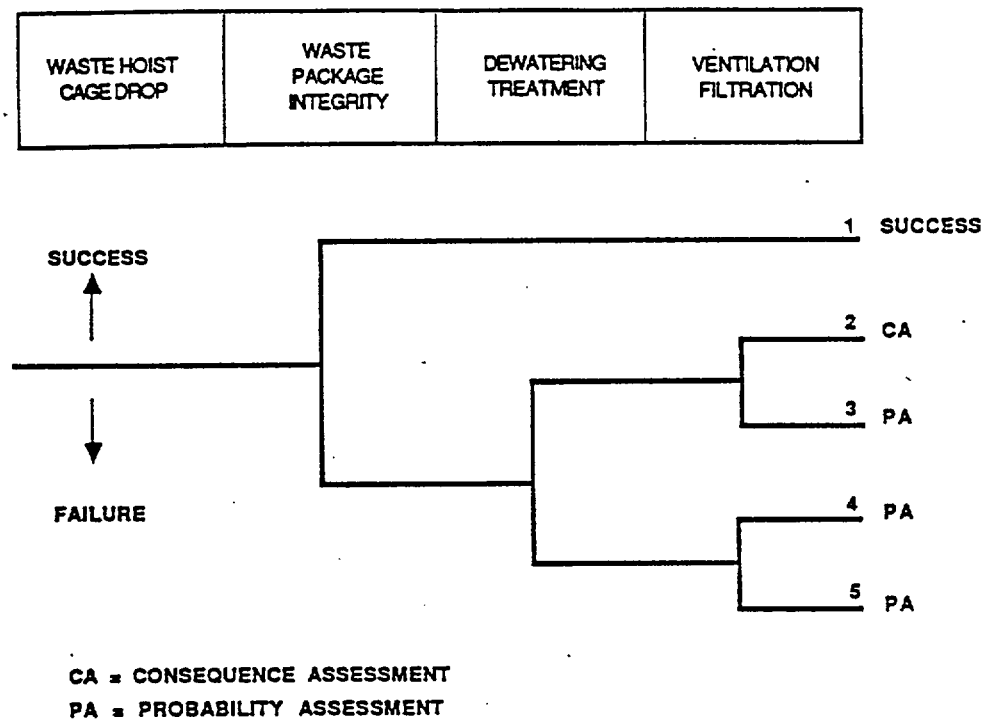


Figure 8. Waste Hoist Cage Drop Event Tree.

exist during the emplacement phase when the cage is traveling up the shaft, after having unloaded a waste package at the subsurface station. During the retrieval phase, this condition would exist when the cage is traveling down the shaft, having already unloaded the retrieved waste package at the surface. This condition is depicted by the "up" branch under the waste package integrity top event block. This sequence does not depend on the effects of the other top events, since there is no potential for a radionuclide release regardless of how the mitigative systems function.

5.2.2.2 Sequences 2, 3, 4, and 5. Sequences 2, 3, 4, and 5 are represented by the scenario in which the cage, loaded with a waste package, drops down the shaft. The sequences can vary, depending on the combination of success and failure of the dewatering treatment and R5 ventilation filtration mitigative functions. If the dewatering pumps are not damaged by the impacting hoist cage, the radionuclides entrained in any shaft sump water could be immediately pumped out of the shaft. For the dewatering treatment to function successfully, it must be able to detect the radiation and promptly respond to the event by treating the contaminated water. Any airborne radionuclide release could be swept up the shaft in the R5 ventilation airflow. The R5 ventilation filtration is primarily a passive system and is continuously on-line during repository operations.

Sequence 2 represents the successful operation of both the dewatering treatment and ventilation filtration functions. This should be adequate to limit the radionuclide releases to the environment to acceptable levels, but a consequence analysis must verify that the radiation dose to the public is below the consequence threshold value of 100 mrem. Sequences 3 and 4 represent the success of one of the mitigative functions and the failure of

the other. Since this could result in a release to the environment by the failed pathway potentially above acceptable limits, the sequences must be shown to be incredible (i.e., the mitigative function must be reliable enough to make its probability of failure low enough to make the sequence's probability of occurrence below the probability threshold value of 1×10^{-7}/yr best estimate or 1×10^{-5}/yr with 95% confidence). Sequence 5 represents the failure of both mitigative functions and must be resolved in the same way as Sequences 3 and 4.

5.2.3 Explosion Event Tree

The explosion event tree is shown in figure 9 and is represented by individual success and failure responses to the three questions listed below.

- Does an explosion occur?
- Is the integrity of the waste package challenged?
- Is the confinement side ventilation airflow filtered?

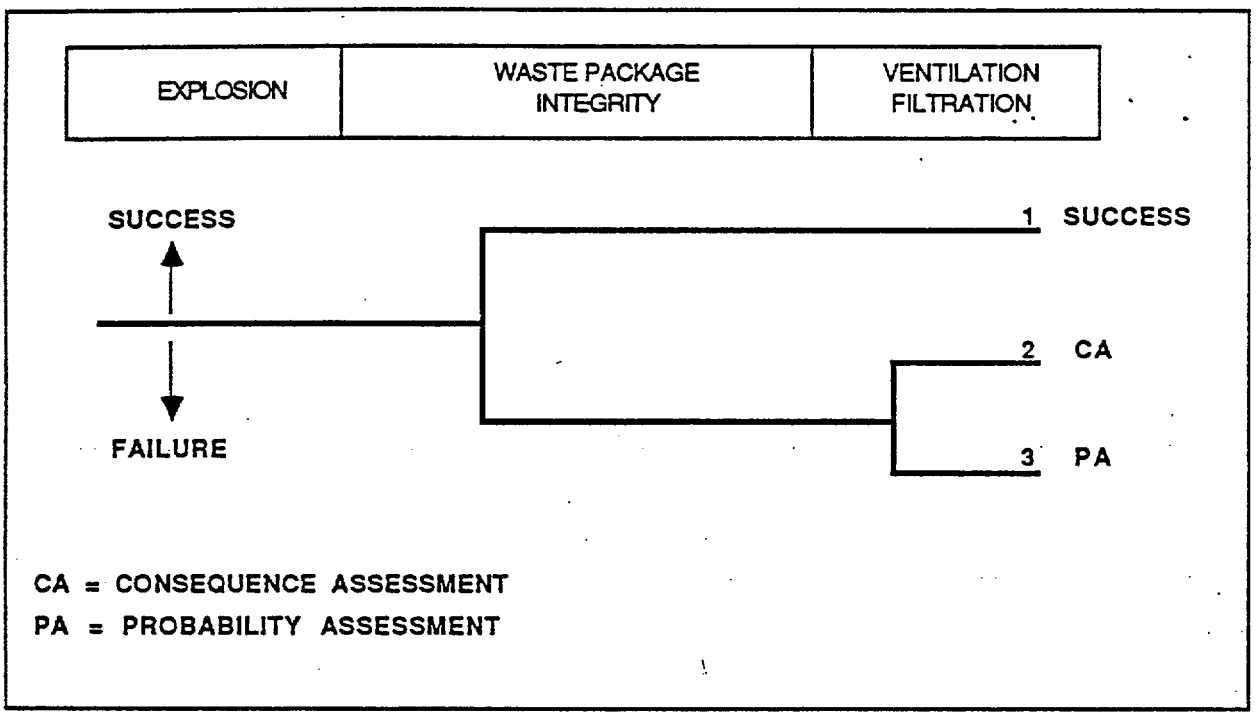


Figure 9. Explosion Event Tree.

The explosion event tree considers the potential impacts from a magazine explosion, inadvertent explosion during transportation of explosives to the development face, methane explosion, complete loss of all electrical power, and complete loss of the ventilation system. The first three incidents are obviously explosion accidents, and the last two incidents can result in situations that could lead to a methane explosion; hence their inclusion in this event tree category. The effects of the first three accidents are discussed in section 5.1.1.7. The methane explosion is not expected to have a significant impact on the waste package, since its effects would be distributed. Similarly, the explosion of a magazine is not considered to have a credible potential for impacting the waste package, since the magazines are located on the development side and the waste packages are on the confinement side at all times. However, an inadvertent detonation of explosives being transported to a development face could potentially have an impact on the waste package if the explosives travel through the confinement side drifts (refer to section 5.1.1.7 for further discussion of these accident initiators). Therefore, the explosion event tree is developed for the inadvertent detonation of explosives being transported to the development face, since it is considered the only credible event with a potential impact on the waste package.

5.2.3.1 Sequence 1. Sequence 1 illustrates the accident scenario that results in no adverse impact to the waste package and is therefore considered to be a success path. The sequence considers the detonation of explosives during transport, which either does not occur on the confinement side and/or does not occur near a loaded waste transporter. Since the explosion does not have a direct impact on the waste transporter, the scenario results in no impacts on the waste package. This sequence is considered a success path, since it results in no radionuclide releases.

5.2.3.2 Sequence 2. Sequence 2 assumes the waste transporter is near the inadvertent detonation of the explosives. This could only occur if the explosives were transported through the confinement side, in violation of administrative procedures. The waste package is assumed to be damaged by the explosion resulting in a release of radioactive material from the waste package. Some particles could be picked up by the confinement side ventilation airflow and be carried to the confinement ventilation exhaust shafts (R5, R6, and R7). The R5 shaft ventilation exhaust is continuously filtered during operations. However, the R6 and R7 shaft ventilation exhaust is not normally filtered because of velocity requirements on these shaft exhaust airflows. Therefore, the R6 and R7 filters are normally bypassed. Upon detecting the release of radioactive particles in the subsurface, dampers would be actuated to divert the airflow through the R6 and R7 filters. Loss of ventilation airflow would not cause a failure of the filtration function. Instead, a decrease in ventilation airflow could result in a greater plateout and deposition of radioactive particles in the subsurface facilities. In this scenario, the ventilation exhaust airflows are assumed to be properly aligned and filtered. This should be adequate to limit the radionuclide releases to the environment to acceptable levels. However, this must be confirmed by a consequence analysis.

5.2.3.3 Sequence 3. Sequence 3 is similar to Sequence 2, with the exception that the ventilation filtration function fails to prevent the radionuclide release from reaching the environment. The failure could occur in any of the three ventilation exhaust shafts (R5, R6, or R7). The airflow to the R6 and R7 filters must be aligned; whereas, the R5 filters are normally in the airflow. Since the R6 and R7 filters require actuation of dampers and operation of supporting systems (e.g., radiation detectors and electrical power), they are more likely to fail to confine the release than the R5 filtration function, which is basically passive. These functions need to be reliable enough so that the probability of this sequence is sufficiently low to be considered incredible. This will be demonstrated by a probability analysis in section 5.3.

5.2.4 Fire Event Tree

The fire event tree is shown in figure 10 and is represented by individual success and failure responses to the four questions listed below.

- Does a fire occur?
- Is the fire suppressed?
- Is the integrity of the waste package challenged?
- Is the confinement side ventilation airflow filtered?

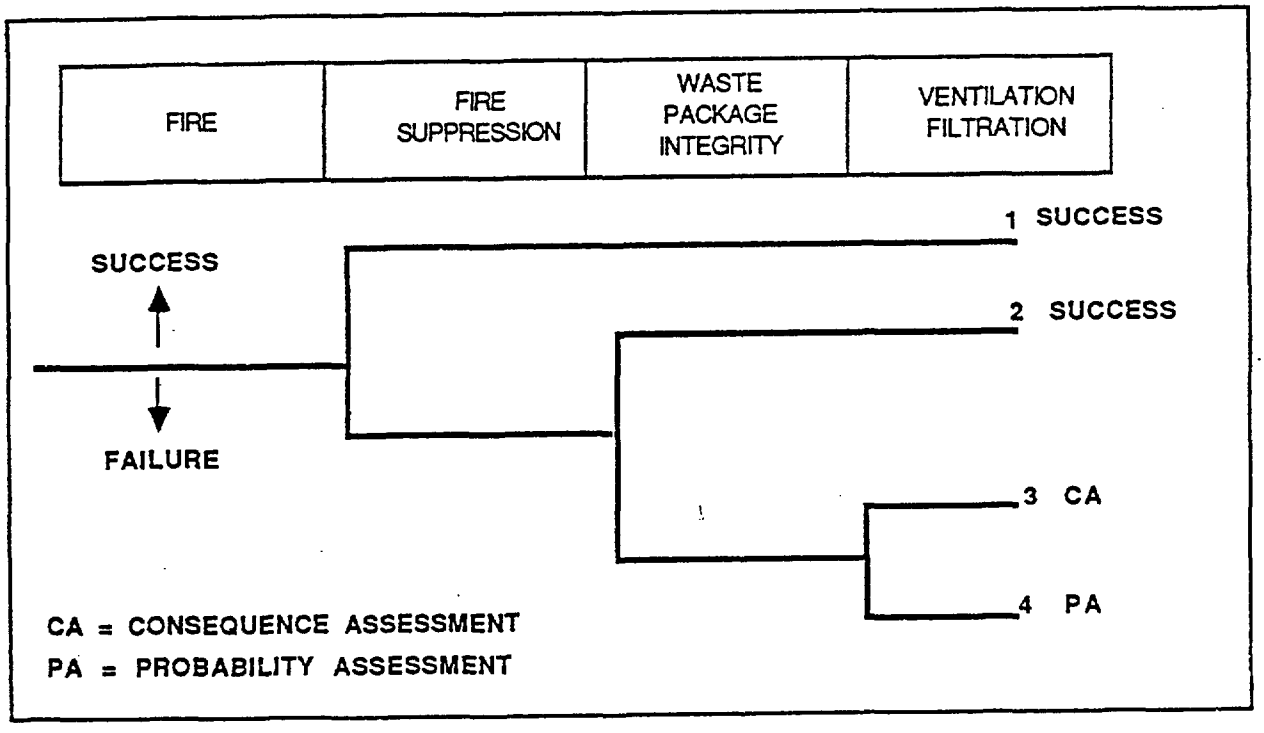


Figure 10. Fire Event Tree.

The fire event tree considers the potential impacts of a fire associated with the waste transporter and a fire in the diesel storage tank area. These fires are discussed in section 5.1.1.6. A bounding calculation (given in appendix A) establishes that a fire associated with the transporter would not contain enough energy to overpressurize the waste container. However, a fire in the diesel storage tank area, which is located on the confinement side, was considered to have the potential for causing an overpressure failure of the waste package. Therefore, the fire event tree is developed for the initiating event represented by a fire in the diesel storage tank area. This event tree should apply to both normal and retrieval operations.

5.2.4.1 Sequences 1 and 2. Sequences 1 and 2 represent the accident scenarios that result in no adverse impact to the waste package and thus are considered to be success paths. Sequence 1 assumes the fire suppression system in the diesel storage tank area functions and eliminates the potential hazard related to the fire. Sequence 2 assumes the fire suppression system fails to operate adequately enough to prevent the fire from burning the fuel at the diesel storage tank. The fire could last for a long period of time and could cause considerable damage to anything in the surrounding area. Sequence 2 results in no adverse impact to the waste package because the heat from the burning fuel does not cause the waste container to overpressurize. This is because the design and administrative controls are assumed to not allow a transporter loaded with a waste package to be in the diesel refueling area. These sequences are considered success paths, since the waste package is not damaged in the accident.

5.2.4.2 Sequence 3. Sequence 3 is represented by the accident situation in which the fire results in an overpressurization of a waste package. This could only occur if a loaded transporter violates procedures and is in the diesel refueling area. The waste package is assumed to release radioactive particles, which could be carried by the confinement ventilation airflow to the confinement exhaust shafts. In this scenario, the confinement exhaust airflow is assumed to be properly aligned and filtered. The operation of the filters should be adequate to reduce the radionuclide release to the environment to acceptable levels. However, since the fire could potentially cause a large amount of volatile radionuclides to be driven out of the waste package, the adequacy of the confinement filtration function must be confirmed by a consequence analysis.

5.2.4.3 Sequence 4. Sequence 4 is similar to Sequence 3, with the exception that the ventilation filtration function fails to confine the radionuclide release from reaching the environment. The failure could occur in any of the three ventilation exhaust shafts (R5, R6, or R7), but would more than likely occur in R6 or R7. These filters are normally bypassed and the airflow must be diverted, by active components, to provide filtration. Since this scenario could result in a potentially unacceptable radionuclide release to the environment, the sequence must be shown to be of a low enough probability to be considered incredible. This will be done by a probability analysis.

5.2.5 Rock Failure Event Tree

The rock failure event tree is shown in figure 11 and is represented by individual success and failure responses to the three questions listed below.

- Does a rock failure event occur?
- Is the integrity of the waste package challenged?
- Is the confinement ventilation airflow filtered?

The rock failure event tree incorporates the potential impacts from a rock burst, rock fall, and drift collapse events in the repository. These events are grouped together, because they could all have a possible direct impact on a waste transporter carrying a waste package to an emplacement area.

5.2.5.1 Sequence 1. Sequence 1 depicts a rock failure incident that does not impact the waste package and thus is considered a success path. The reason a rock failure would not have an adverse impact on the waste package is because the failure did not occur where the transporter was traveling.

5.2.5.2 Sequence 2. Sequence 2 represents the scenario where the rock failure occurs near a loaded transporter and causes damage to the waste package. The damaged waste package could release radioactive particulate to the ventilation airflow, which could carry the particles to the confinement side ventilation exhaust shafts. In this sequence, the confinement filters are aligned and functioning properly. The filters should be adequate to limit the surface release of radionuclides to acceptable levels. This must be verified by a consequence analysis.

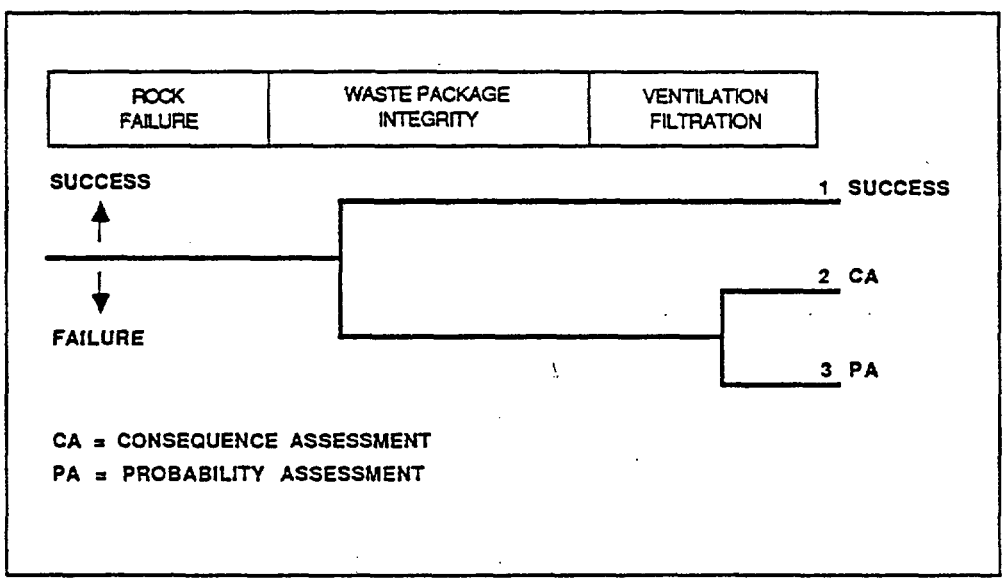


Figure 11. Rock Failure Event Tree.

5.2.5.3 Sequence 3. Sequence 3 depicts the rock failure scenario where the failure is assumed to damage the waste package (similar to Sequence 2). However, in this sequence the filtration on one of the confinement exhaust shafts is assumed to be bypassed. The filtration failure could result in a release of radionuclides to the environment. To demonstrate the acceptability of this sequence, the sequence must be shown to be incredible. This will be done by a probability analysis.

5.2.6 Seismic Event Tree

The seismic event tree is shown in figure 12 and is represented by individual success and failure responses to the six questions lsited below.

- Does an earthquake occur?
- Is a waste hoist cage drop induced?
- Does a drift collapse?
- Is the integrity of the waste package challenged?
- Is the dewatering pathway treated?
- Is the confinement side ventilation airflow filtered?

The seismic event tree represents the impacts of an earthquake on the repository. This event could result in damage to the ventilation and electrical power systems. In addition, the earthquake could potentially damage the waste hoist and cause the cage to drop down the shaft. In addition, an earthquake could cause portions of the drifts in the repository subsurface to collapse.

The seismic event tree is slightly different from the previous event trees in that the event could potentially degrade the performance of multiple systems and induce simultaneous common cause failures. Nonetheless, the repository should respond to this situation in essentially the same way as for other accidents considered. In actuality, the event tree is basically a combination of the rock failure and waste hoist cage drop event trees. Since the event has the potential for multiple and common failures, the probability of system failures may increase. In addition, many non-Q-Listed systems could fail and have a potential adverse impact on Q-Listed systems.

5.2.6.1 Sequences 1, 2, 5, and 10. Sequences 1, 2, 5, and 10 all result in no impact to the waste package and are thus considered to be success paths. The scenario represented by Sequence 1 is the occurrence of an earthquake that (1) does not damage the waste hoist's preventive features severely enough to cause a loaded cage to drop down the shaft, and (2) does not affect the stability of the subsurface drifts. Since neither of these situations occur, the waste package should not be damaged by the earthquake. Sequence 2 depicts the occurrence of an earthquake with the successful operation of the waste hoist preventive features. However, the subsurface

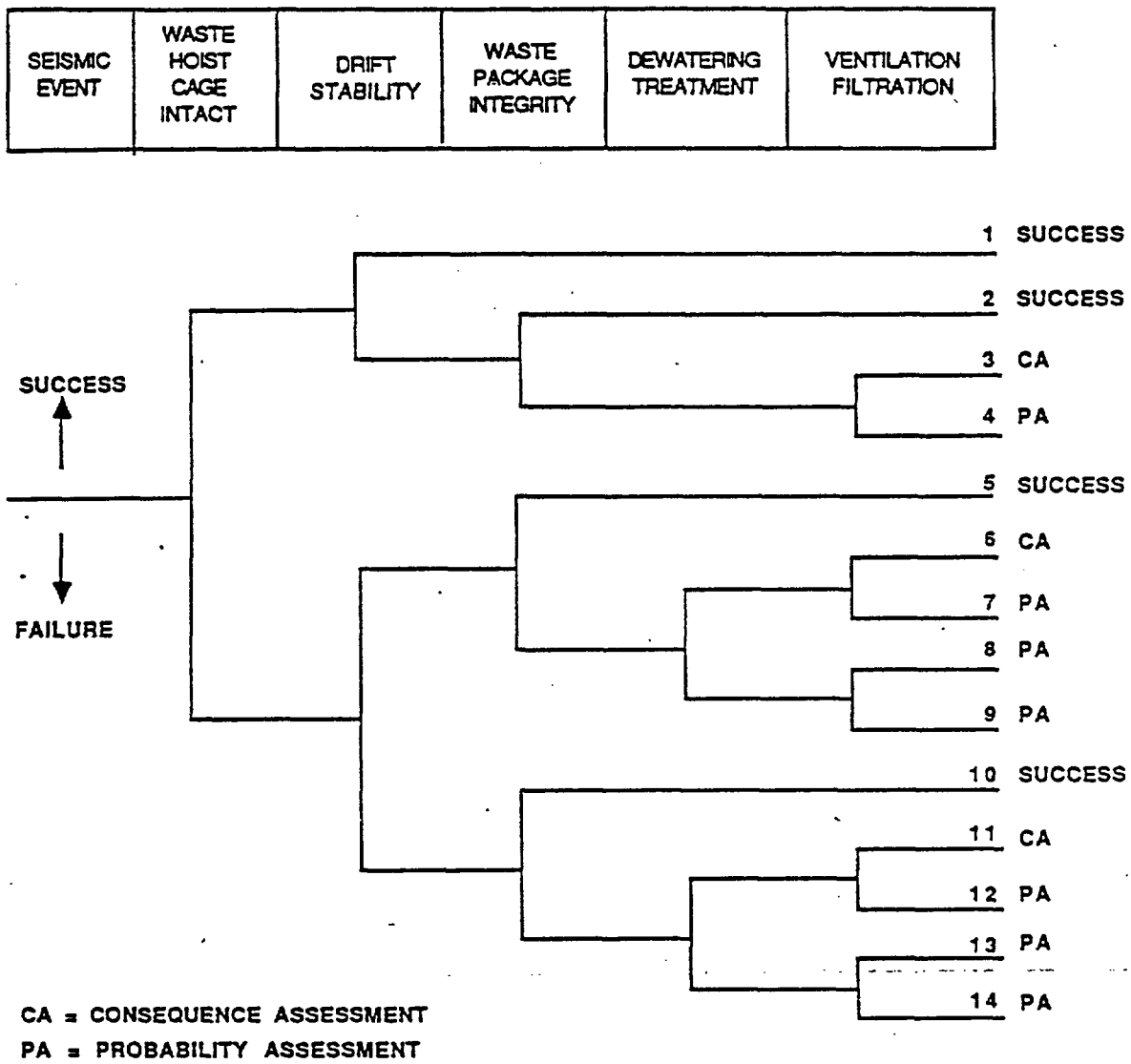


Figure 12. Seismic Event Tree.

drifts are assumed to be damaged, resulting in portions of the drifts to collapse. The waste package is assumed not to be impacted in this sequence, because a loaded transporter is not in the area where the collapsing occurs. Sequence 5 is the obverse of Sequence 2, in that the subsurface drifts are assumed not to be affected, but the waste hoist preventive features fail and the hoist cage falls down the shaft. However, the waste package is assumed not to be adversely impacted by the fall in this sequence. This would most likely occur if the waste package was not on the hoist when the earthquake and resulting drop occurred. Sequence 10 is the combination of Sequences 2 and 5. The waste hoist preventive systems are assumed to be failed and the subsurface drifts partially collapse. Again, the waste package is assumed not to be damaged in this sequence. This would occur if the waste package is not on the hoist, and if a loaded transporter is not in the area of the drift collapse.

5.2.6.2 Sequences 3 and 4. Sequences 3 and 4 depict a different progression for Sequence 2. Following the seismic event, the hoist system is assumed not to be damaged, but portions of the subsurface drift collapse. In both of these sequences, the waste package is assumed to be on a transporter in the area affected by the collapsing drifts. The damaged waste package could release radionuclides to the subsurface area, which could then be picked up by the confinement ventilation airflow and swept to the confinement ventilation exhaust shafts. Sequence 3 assumes that filtration is successful, which should be adequate to lower the radioactive release to the environment to acceptable levels. This will be verified by a consequence analysis. Sequence 4 assumes that filtration fails to confine the release. Should this occur, the release to the environment could be above acceptable levels. Therefore, this sequence must be shown by probability analysis to be incredible.

5.2.6.3 Sequences 6, 7, 8, and 9. Sequences 6, 7, 8, and 9 represent similar scenarios (a seismic-induced hoist cage drop, but no collapsing of drifts) up to the point of the operation of mitigative functions. At this point, the sequences vary, depending on the combination of success and failure of the dewatering and R5 ventilation filtration mitigative functions, which occur after the release of radionuclides from the damaged waste package.

As stated before, the R5 ventilation filtration is on-line during repository operations and is considered a passive system (i.e., active components are not required to operate for the filtration function to succeed). For dewatering treatment to be successful, it must detect the radiation and promptly respond to the incident by treating the contaminated water.

Sequence 6 represents the successful operation of both the dewatering treatment and ventilation filtration functions. This should be adequate to limit the release to the environment to acceptable levels. Sequences 7 and 8 represent the success of one of the mitigative functions and the failure of the other. Because this could result in a release to the environment by the failed mitigative pathway potentially above acceptable levels, the sequences must be shown to be incredible. Sequence 9 represents the failure of both mitigative functions. This would indicate the potential for a

release by both pathways and must be shown to be incredible. Sequences 6 through 9 will be verified by the appropriate consequence or probability analysis.

5.2.6.3 Sequences 11, 12, 13, and 14. Sequences 11, 12, 13, and 14 differ from the previously discussed sequences in that these sequences also assume that a loaded transporter is impacted by a seismic-induced collapse of portions of the subsurface drifts. Sequence 11 represents the scenario in which the dewatering treatment and ventilation filtration mitigative functions operate correctly. This should be sufficient to limit the releases to the environment to acceptable levels. This must be verified by a consequence analysis. Sequences 12 and 13 represent the successful operation of one of the mitigative functions and the failure of the other. This could allow a potentially significant release to the environment and thus must be shown by probability analysis to be incredible. Sequence 14 considers the failure of both mitigative functions, resulting in the potential for multiple, unmitigated pathways for the radionuclide release to the environment.

5.3 EVENT TREE PROBABILITIES

The event trees were developed by following the methodology outlined in section 5.2.1, and each event tree was described in the remaining sections of section 5.2. To determine if the the repository design is adequate and that all items that ought to be considered important to safety are Q-Listed, all event tree end states must be shown to be successes, be of acceptable consequence, or be incredible sequences (i.e., event sequences of acceptably low probability). Another way to say this is that the reliabilities of the required safety functions are acceptable. For the latter to be established, the probability of occurrence of the initiating events and the probability of failure for each of the required functions must be determined. The initiating event frequencies were provided in section 5.1. The probabilities for each resulting event will be discussed in this section for each event tree. The calculations used to develop the probabilities in this section are provided in appendix A.

To be determined as incredible, the probability of the sequence must be below an established probability threshold value. The probability threshold value, for this report, has been established to be a best estimate annual probability of 1×10^{-7} or an annual probability of 1×10^{-5} with 95% confidence.

5.3.1 Waste Hoist Cage Drop Event Tree Probabilities

The waste hoist cage drop event tree is shown in figure 13 with the associated intermediate event probabilities shown on the respective branches. The branch probabilities are only shown for the failure paths. To calculate the probability of success of the path, the failure probability is subtracted from unity (i.e., the probabilities of failure and success must add to unity). The probability of a waste hoist cage drop was provided in section 5.1 as being approximately 2×10^{-8}/yr. Each branch probability is discussed below.

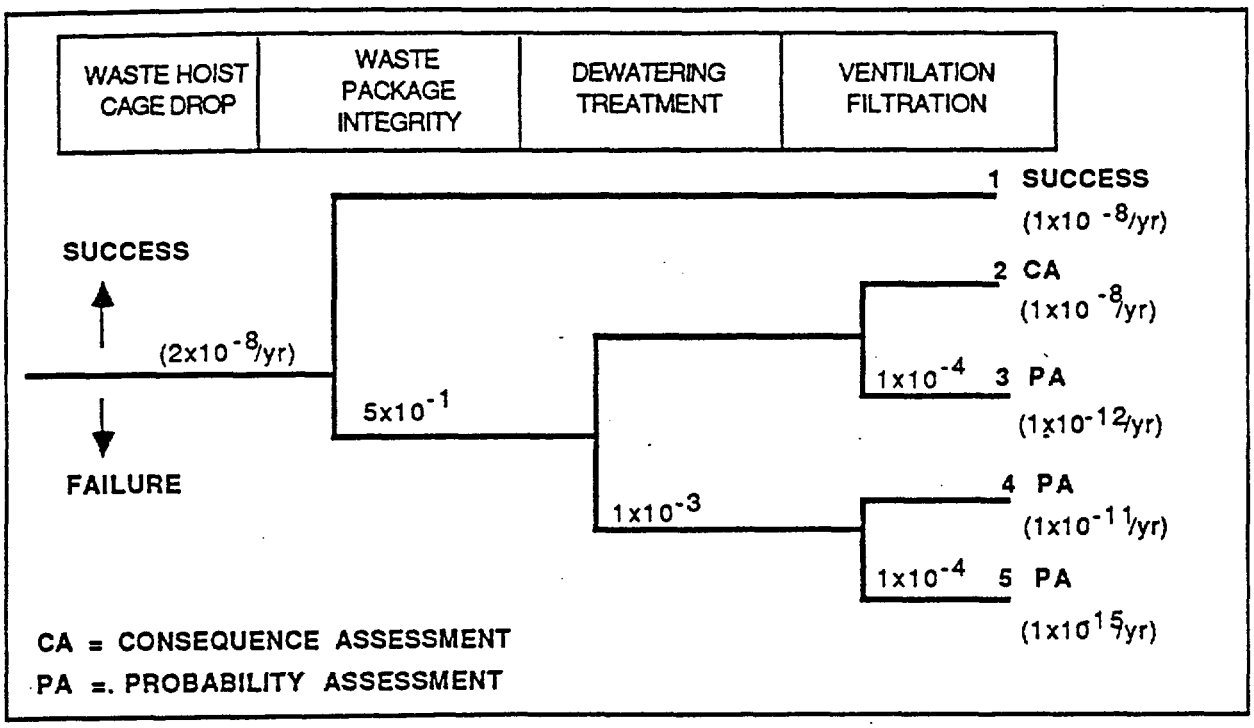


Figure 13. Waste Hoist Cage Drop Event Tree.

The waste package is assumed to fail if it is on the hoist cage when the drop occurs. Since the drop may occur anywhere along the hoisting cycle and the waste package is only on the hoist during descent (for emplacement operations), the conditional probability of the waste package being on the hoist cage when the drop occurs is 0.5.

If the dewatering system is operational after the drop, the water pumped to the retention ponds, which are on the surface, may need to be treated to prevent the radionuclides in the water from causing an unacceptable dose to the public. The chance that the treatment does not occur is primarily controlled by the ability of the dewatering system to detect the radiation in the water. Failure of treatment is considered very unlikely, on the order of 1×10^{-3} per demand.

The R5 shaft is an upcast shaft with a filter bank normally in the ventilation airflow. Filters are essentially passive components and are extremely unlikely to fail during accidents, unless the accident directly impacts and damages the filters. The frequency of filter failure is approximately 1×10^{-4} per demand for this event, since the waste hoist cage drop would not cause a direct impact on the filters.

Since the initiating event probability is extremely low (because of the high reliability of the waste hoist cage preventive systems), all of the sequences for the waste hoist cage drop accident are considered incredible. Sequence 2 will be addressed in section 5.4 to determine the adequacy of the dewatering treatment and R5 ventilation filtration mitigative functions even though the scenario is not credible.

5.3.2 Explosion Event Tree Probabilities

The explosion event tree is shown in figure 14, with the associated intermediate event probabilities depicted on the respective branches. As discussed in section 5.2, the explosion event tree is represented by the occurrence of an inadvertent detonation of explosives being transported to a development face. The probability of a this initiating event was given in section 5.1 as 3×10^{-7}/yr.

The explosion would not have the potential for an adverse impact on the waste package unless the explosives are transported through the confinement side and are in close proximaty to a loaded waste transporter. This accident would require the operator transporting the explosives to knowingly violate administrative procedures. The conditional probability of such a situation occurring is assumed not very likely and is therefore assumed to be approximately 1×10^{-2}.

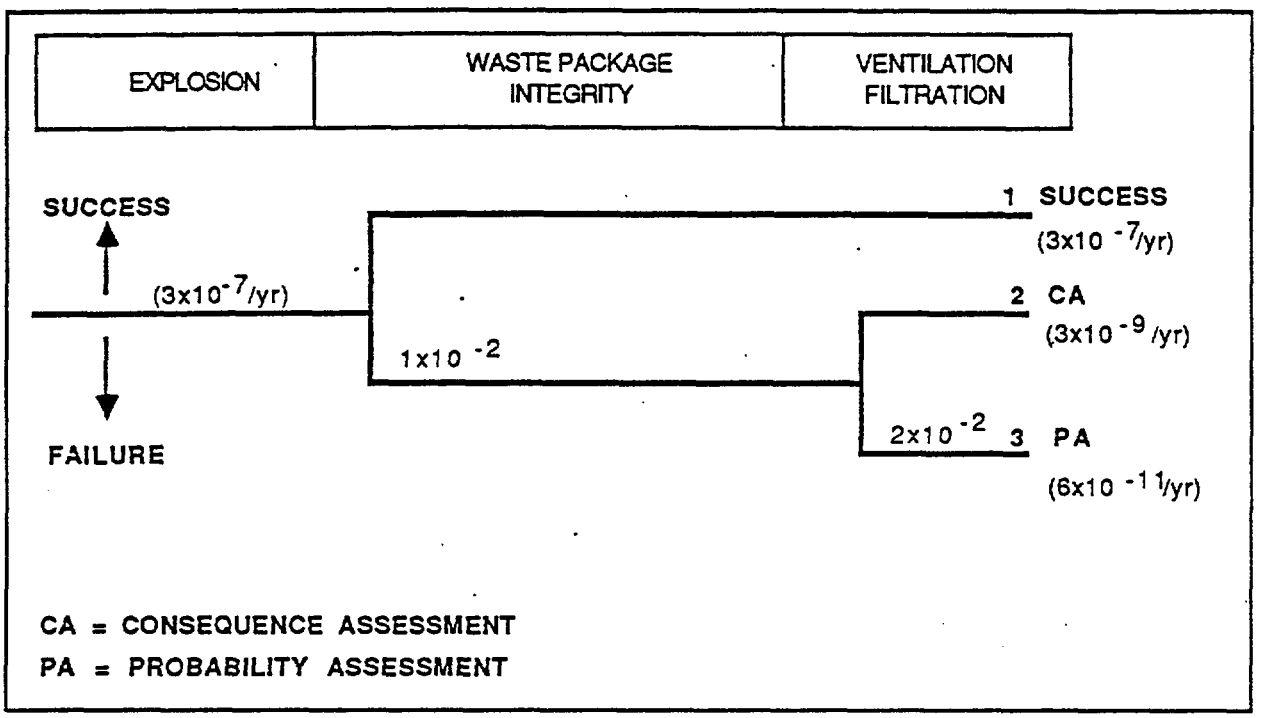


Figure 14. Explosion Event Tree.

Should a waste package be damaged and release radioactive particles, some particles would be picked up by the confinement ventilation airflow and be carried to the confinement exhaust shafts (R5, R6, and R7). The R5 shaft has filtration available during normal repository operations. The R6 and R7 filters are normally bypassed because of the requirements for normal ventilation airflow in these exhaust shafts (the R6 and R7 airflow velocity requirements are much greater than the R5 airflow). Upon detection of radiation, dampers are actuated to divert the airflow through these filters. The explosion should not have an effect on the filters, because the explosion would occur sufficiently far from the confinement exhaust shafts. The most probable failure mode is the failure of the R6 or R7 active components (R5 components are primarily passive). This failure could be the failure of radiation detectors to actuate the dampers or be the failure of the dampers to divert the airflow. Such a failure is estimated to have a probability of 1×10^{-2} per demand. Since there are two exhaust shaft airflows (R6 and R7) with this potential, the combined probability of an unfiltered path due to failure of active components is 2×10^{-2} per demand. The probability of passive filtration failure was stated in section 5.3.1 to be on the order of 1×10^{-4} per demand. In this event, there are three exhaust shaft airflows (R5, R6, and R7) with the potential for passive filtration failure, thus a combined probability of 3×10^{-4} per demand. This probability is significantly less than the probability of active filtration failure. Therefore, the probability of an unfiltered release due to filtration failure is approximately 2×10^{-2} per demand.

All of the sequences are either success paths, or their probabilities are below the probability threshold value and are thus considered incredible. Sequence 2 of the explosion event tree will be considered in section 5.4 to determine the adequacy of the confinement ventilation exhaust filtration function even though the event is not credible.

5.3.3 Fire Event Tree Probabilities

The fire event tree is shown in figure 15, with the associated intermedate event probabilities shown on the respective branches. As stated in section 5.2.4, the fire event tree depicts the occurrence of a fire in the diesel storage tank area. The probability of a fire in this area was calculated in section 5.1.6 to be 1.6×10^{-7}/yr.

The nearby fire suppression system should be automatically actuated to prevent the fire from increasing in intensity and potentially spreading. Most fire suppression systems initiate passively (i.e., their operation is based on natural phenomenon, such as the heat generated by the fire). The conditional probability that the fire suppression system fails to actuate and extinguish the fire is estimated to be 1×10^{-2} per event.

If the fire suppression system fails and a transporter loaded with a waste package is in the area, the waste container could be subjected to immense heat and could be overpressurized. The overpressurization is assumed to result in large amounts of volatile radionuclides being released from the waste container. The opportunity for a loaded waste transporter to

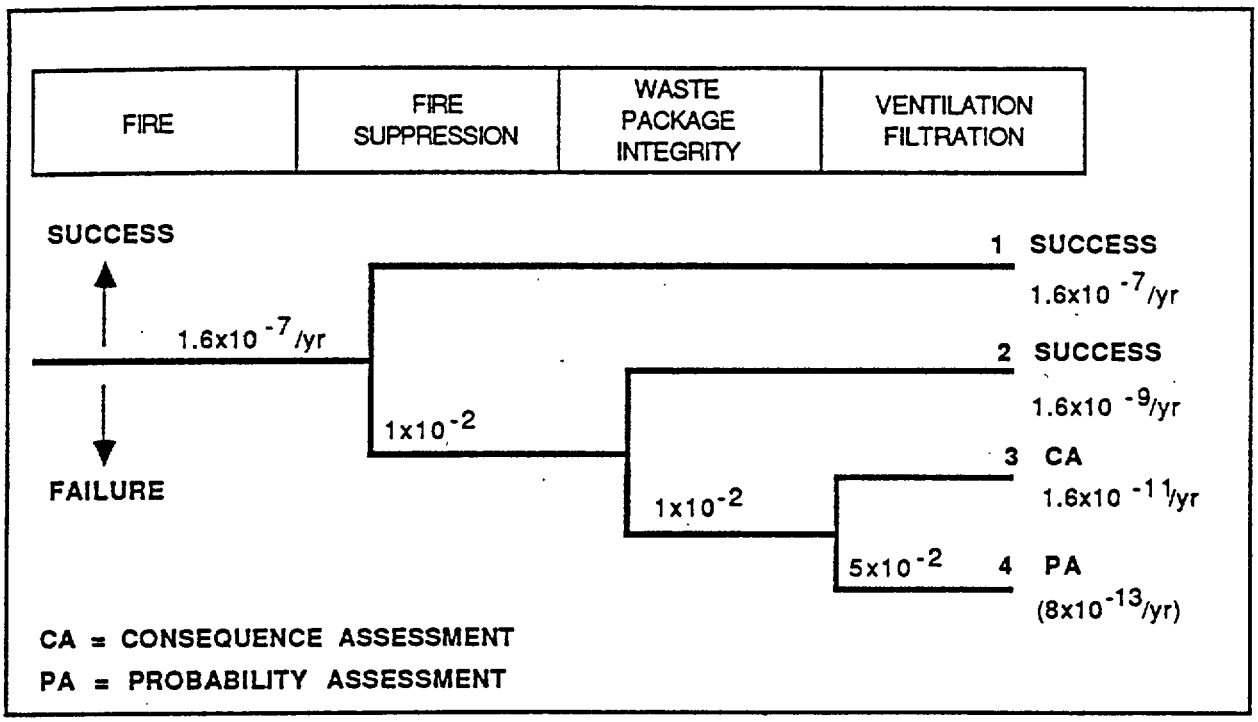


Figure 15. Fire Event Tree.

be in the refueling area is considered to be remote because of administrative controls and possibly design limitations. For a loaded transporter to be in the area, the operator would probably have to be in direct violation of an operating procedure. The conditional probability of knowingly violating such a procedure is estimated to be on the order of 1×10^{-2}.

If the waste container is overpressurized and releases volatile radionuclides to the subsurface ventilation airflow, the release will be transported to the confinement exhaust shafts. The ventilation filtration would operate as described in the prior sections, though there is the additional potential for the fire to clog or burn out the filters. The probability of passive filtration failure (i.e., failure of the filters, themselves) is expected to be two orders of magnitude more likely then given in section 5.3.2, for the explosion event tree, due to the potential for direct impacts on the filters from the fire. The active filtration failure probability is essentially unchanged by the fire's potential impacts. Therefore, the combined probability of failure for the filtration function is 5×10^{-2} per demand.

The fire event should result in the greatest environmental release due to the potential for a large release of volatile radionuclides that may not be confined by the filters. The frequency of having a fire in the diesel storage area is extremely low, and the low probability of having a loaded transporter in the area when the fire occurs reduces the probability of a radionuclide release even more. All of the sequences are either successes or are incredible. Sequence 3 will be addressed in section 5.4, even though it is not a credible event, to determine the adequacy of the confinement ventilation filtration function during a fire.

5.3.4 Rock Failure Event Tree Probabilities

The event tree depicting rock failure is shown in figure 16. The intermediate event failure probabilities are shown on their respective branches. As discussed in section 5.2.5, the rock failure event tree incorporates the potential impacts from rock burst, rock fall, and drift collapse events in the repository. However, it is expected that only the drift collapse event could cause a significant impact to and release from the waste package. Therefore, the initiating event probability is assumed to be the probability of a drift collapse, 1 event per year.

The occurrence of a rock failure has the potential for a direct impact on the waste package if the waste package is on a transporter near the area of the failure. For there to be a release from the waste package, three conditions must exist. The first condition is that the rock must fail over, or sufficiently near, the transporter (this length of drift is assumed to be 15 m (50 ft)). Because the transporter could be in any section of the confinement in which it travels (stated in section 5.3.2 as 193 km (120 mi)), the conditional probability of a rock failing sufficiently near a transporter is 8×10^{-5}. The second condition is that the transporter must be loaded with a waste package when the rock failure occurs. This could only occur when the transporter is traveling to an emplacement room. There are, on the average, 2,500 emplacements each year, and each emplacement is

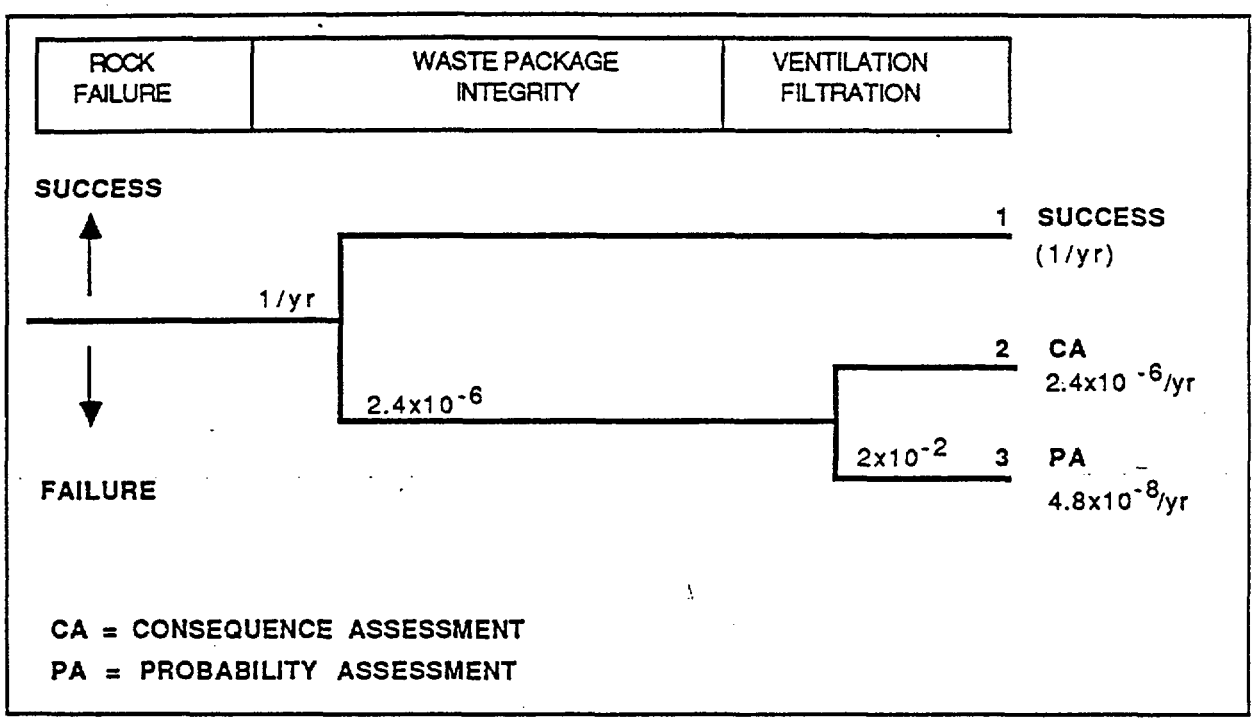


Figure 16. Rock Failure Event Tree.

conservatively expected to take 1 h. Therefore, the conditional probability of a rock failure during the transporter emplacement trip is approximately 3×10^{-1}. The third condition is that the impact on the waste package is significant enough to cause a radionuclide release. This conditional probability is assumed to be 0.1. The combined probability of a release due to a drift collapse onto a waste transporter is 2.4×10^{-6} (see appendix A for the complete calculation).

Following the rock failure and resulting release from the waste package, the ventilation airflow would need to be filtered. As discussed in section 5.3.2, the probability of filtration failure is approximately 2×10^{-2} per demand.

Sequence 1 of the rock failure event tree is a success path, and Sequence 3 has been shown to be incredible. Sequence 2 is a credible scenario in which confinement ventilation filtration operates. This sequence should be bounded by the Initiators B and C releases, which are addressed in section 5.4.

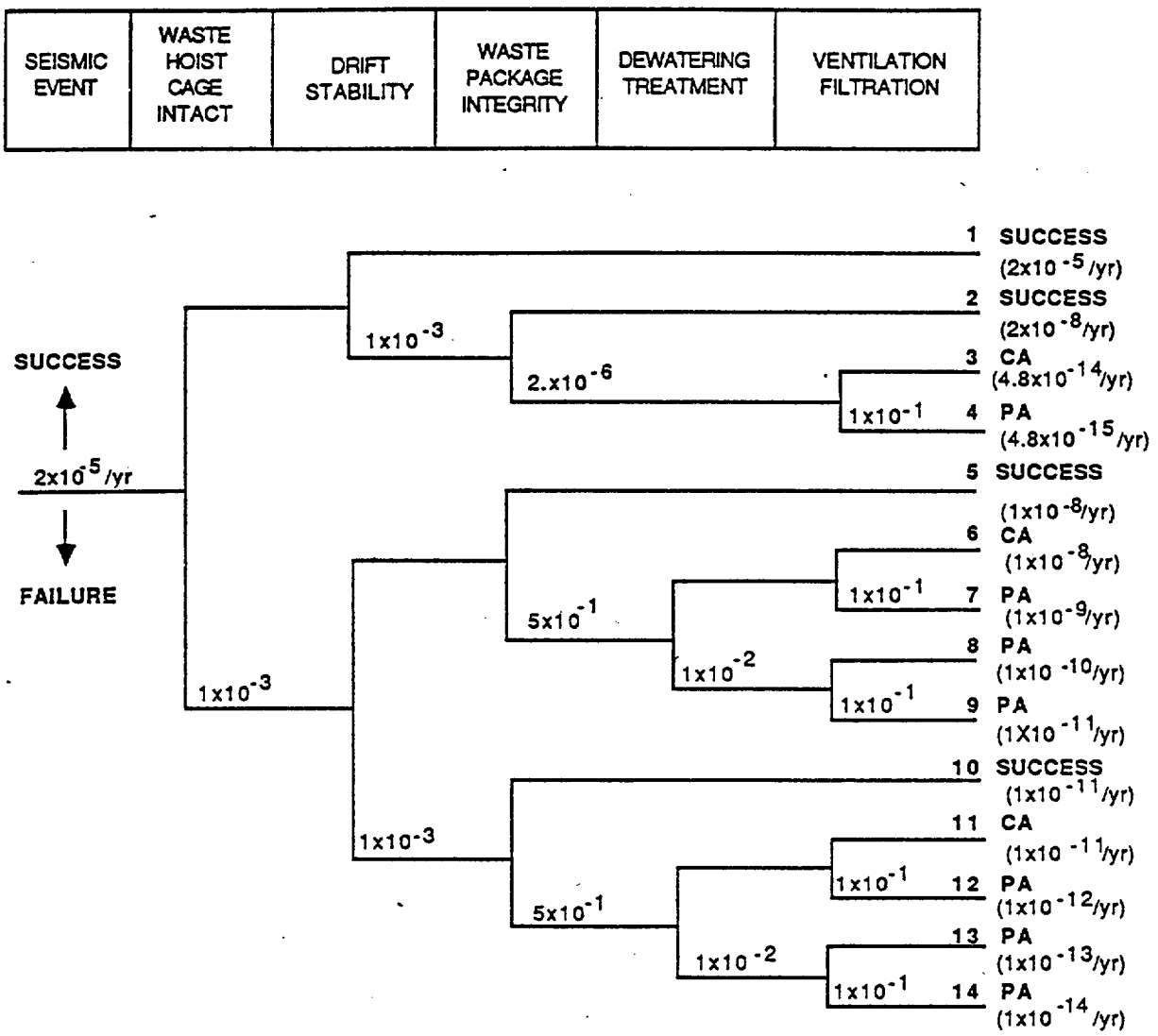
5.3.5 Seismic Event Tree Probabilities

The seismic event is shown in figure 17, with the intermediate event failure probabilities given on their respective branches. As discussed in section 5.2.6, the seismic event has the potential for inducing multiple failures. This is reflected in the increased failure probability of the intermediate events. The seismic event could cause damage to a waste package by causing a drift to collapse, or by causing the waste hoist cage to fall down the R5 shaft. Both of these potential impacts are modeled by the event tree. The probability of a seismic event was given in section 5.1.1 (for the design basis earthquake) as 2×10^{-5}/yr.

For the waste hoist cage to drop, the hoist control systems must fail. However, since these control systems are redundant and have already been identified as important preventive systems, they will be designed to withstand most events. Considering the waste hoist and the building it is in are designed to survive this event, the conditional probability of a drop being induced by the earthquake is on the order of 1×10^{-3}.

Sinigficant impact on the waste package due to collapse of a subsurface drift is not very likely. This conclusion is derived from qualitative mining experience in which large earthquakes occurred, and yet no subsurface effects were noticed. Based on this information, the probability of a seismic-induced drift collapse is estimated to be 1×10^{-3} per event. See section 5.1.1.1.1 for additional discussion of this issue.

The waste package could be impacted in two different areas. The first is during the lowering of the waste package to the subsurface level, and the second is during the transport to an emplacement room. The likelihood of impact during the transport operation is essentially the probability that the transporter is in the area in which the drift collapse occurs. This conditional probability was provided in section 5.3.4, the rock failure event tree, as 2.4×10^{-6}. The probability of a waste package damaged during



CA = CONSEQUENCE ASSESSMENT
PA = PROBABILITY ASSESSMENT

Figure 17. Seismic Event Tree.

the hoisting operation is much higher, if a hoist drop is assumed to occur. As discussed in section 5.3.1, the probability of damaging the waste package is basically the probability that the waste package is on the hoist when the drop occurs. This conditional probability was assumed to be 0.5.

If the dewatering system is operational after the hoist drop, the water pumped to the surface retention ponds may need to be treated to prevent the radionuclides in the water from causing unacceptable doses to the public. The treatment operation is primarily controlled by the ability of the dewatering system to detect the radiation in the water. This operation could be damaged by the earthquake. Therefore, the conditional probability of dewatering treatment failure, following an earthquake, is considered to be 1×10^{-2}.

The impact on the ventilation filtration function would also be increased by the seismic event. The passive components would still be more likely to survive the event than the active components. However, given that the normal power supply is probably lost, the active components in the R6 and R7 confinement exhaust airflows should be automatically actuated, and thus, the airflow should pass through their filters. Taking into account the increased probability of failure for the systems and the automatic actuation to divert flow, the passive filtration failure is assumed to be 1×10^{-3} per demand and the active filtration failure is assumed to be 5×10^{-2} per demand. The total probability of filtration failure would then be approximately 0.1 per demand.

All of the sequences for the seismic event tree are either success paths or are considered incredible events. This is primarily due to the reliability of the hoist preventive systems and the low chance of a drift collapse impacting a loaded transporter, following an earthquake. Though Sequences 3, 6, and 11 are noted as incredible, they will be considered in section 5.4 in the following manner. As discussed for the rock failure event tree, Sequence 3 is assumed to be bounded by the Initiator B and C release scenarios. Similarly, Sequence 6 is considered in the consequence analysis of the hoist cage drop. Sequence 11 is the combination of both of the above events and is considered covered by showing the adequacy of the filtration function for each individual scenario.

5.4 CONSEQUENCE ASSESSMENT OF MITIGATED ACCIDENTS

Radiological dose assessments for both the unmitigated accident initiators and the mitigated accident sequences are summarized in table 10. The only mitigation used for Initiators A through E is a particulate filtration system, assumed to consist of at least two HEPA filters in series, providing a particulate removal factor of 4.0×10^5 (or equivalently, particulate filtration efficiency of 0.9999975) and a volatile radionuclide removal factor of 1,000. For Initiator F, a dewatering system radwaste treatment subsystem with a decontamination factor of 1,000 is assumed. The assumptions regarding filtration efficiency are summarized in table 11. Section 5.4.1 discusses some of the issues surrounding the estimates of accident source terms and filtration efficiencies. Section 5.4.2.1 discusses the codes used for radiological dose assessment, and section 5.4.2.2 discusses the results of the assessment.

Table 10. Radiological Dose to Maximally Exposed Individual (50-yr Dose Commitment).

Accident initiator	Unfiltered dose (mrem)	Filtered dose (mrem)
Initiator A, cask drop	1.7 E + 06 bone	7.5
Initiator B, cask drop; flow up shaft R6	1.5 E + 06 bone	6
Initiator C, cask drop; flow up shaft R7	2.3 E + 06 bone	9.5
Initiator D/E, fire/explosion; flow up shaft R7	6.1 E + 05 bone	49
Initiator F, cask drop; unfiltered dewatering system flow to retention pond	5,300 + 5,300, dose from resuspended particulate + ingestion dose from strontium reaching river through unconfined aquifer	11

NOTE: See appendix B.

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Table 11. Filtration Efficiencies Assumed for the Mitigated Accident Analysis.

1. Based on two HEPA filters in series, the following radionuclide removal efficiencies were assumed for confinement exhaust filtration:

Radionuclides	Removal efficiency
Noble gases	0%
Halogens	50%
Volatiles	50%
Particulates	0.9995 x 0.995 = 1.0 - (2.5 E-06)

The particulate removal efficiencies result from two HEPA filters in series and apply to particulates larger than 0.3 um (1.8 x 10-5 in.). Removal of halogens and volatiles results from adsorption on particulate that is subsequently captured by the filters (Fullwood and Mendoza 1979, appendix A).

2. Initial radiological dose assessments of the mitigated accidents identified the need for additional removal of radiocesium and for some filtration of the dewatering flow from the repository. On this basis, it was assumed that an additional factor of 500 removal of cesium would be provided (see discussion of this point in section 5.4.1.2), and that a total decontamination factor of 1,000 would be provided for the dewatering flow. The radiological dose assessment for the unmitigated accidents reflects this additional filtration.

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5.4.1 Source Terms and Filtration Efficiencies

Determination of spent fuel accident source terms and filtration efficiencies is subject to significant uncertainties. Most of the experimental evidence on radionuclide releases from spent fuel involves high fuel temperatures more appropriate to reactor accidents. There is no experimental evidence truly representative of the extreme mechanical disruption of fuel and cladding that would accompany the hypothetical cask drop accident. Because of this, the Source Term Working Group arrived at its estimate of the subsurface cask drop accident source term using an extrapolation of experimental evidence on the amount of particulate material present in undisturbed spent fuel and the amount of new particulate created by mechanical disruption of spent fuel and other ceramic and vitrified materials. It should be emphasized that none of the experimental evidence involved the high burnups (60,000 MWd/MTU) that the DOE-HQ methodology requires or the high energy densities implied by the hypothetical cask drop accident. The Working Group estimated 0.5 wt% of existing suspendible particles, 1.5 wt% of cask drop generated new respirable particulate, and an additional 9.5 wt% of larger particulate generated by the cask drop. These estimates are roughly consistent with the estimates by Pepping et al. (1981) and the guidelines given by Harris et al. (1985).

Additional uncertainties surround the estimation of halogen and volatile radionuclide release. The only halogen remaining in 5-yr cooled spent fuel, iodine-129, is not present in sufficient quantity to produce a appreciable dose for an accident involving only one waste package. Of the volatile radionuclides, only cesium-134 and cesium-137 are present in sufficient quantity to produce an appreciable dose in a mitigated accident. The Working Group's estimates of volatile release were 20% of inventory released in the cask drop accident and 100% of inventory in the fire/explosion accidents. In all cases (except the cask drop into water), it was assumed that half of the volatiles are removed from the air flow on the way to the surface. In addition, it was assumed that the HEPA filters would remove another half of the volatiles (the physical mechanism for this was the assumption that that fraction of the volatiles would be tightly bound to particulate which was trapped by the filter). With these assumptions, dose to the maximally exposed individual during mitigated accidents (with the HEPA filters working) ranged from 3 to 24 rem (for the fire/explosion accident). Virtually all of this mitigated accident dose is due to cesium.

By the OCRWM methodology, when faced with unacceptably high consequences, the Q-List Preparation Team was required to assume the existence of additional mitigative systems capable of reducing the consequences below the threshold level. On this basis, we assumed the existence of a filtration system that actively removes cesium from the repository exhaust ventilation flow, providing an addition factor of 500 removal efficiency for cesium (so that the total filtration efficiency for cesium is assumed to be 0.999). In reality, the initial cesium source term assumptions were intentionally conservative, since the empirical basis for estimating cesium releases in repository (as opposed to reactor) accident was judged to be deficient. It is expected that, during the course of ACDs

either the cesium source term estimate will be refined (eliminating the need for an active cesium removal subsystem) or the cesium removal filtration will be added to the design or, perhaps, some combination of the two.

Although pure cesium is only a factor of five or so less volatile than iodine (Kelly et al. 1984), experimental evidence suggests that much of the cesium in spent fuel is chemically bound (e.g., cesium iodide, oxides, uranate) in forms that are significantly less volatile. If so, one might expect a smaller fraction of the cesium inventory to be released as cesium vapor and more of the cesium that is released to be tightly bound to the particulate mass, and thus susceptible to being trapped by the HEPA filter.

5.4.2 Discussion of the Radiological Dose Assessment

Table 10 provides the radiological dose assessment for both the unmitigated accidents for Initiators A through F and the mitigated accidents with the assumption of an additional factor of 500 removal of radiocesium.

5.4.2.1 Computer Codes Used. A set of computer programs has been developed at the Hanford Site to calculate the dose consequences from all significant exposure pathways. Each program accesses a common set of standardized libraries which, to the extent they are available, contain Hanford Site specific data. The program and data libraries are maintained by the Hanford Dose Overview Program, with all revisions or updates documented (McCormack et al. 1984). An overall dose model QA plan is in place and has been followed for all code development, revisions, and use. The computer programs have been documented separately, and only a brief description of their application is given here.

- DACRIN--This program (Houston et al. 1974; Strenge 1975) is used to analyze radiation doses from inhalation for Hanford Site operations. The program uses the model of the ICRP Task Group on Lung Dynamics (ICRP 1966) to predict both radionuclide movements through the respiratory system and lung doses. Once radionuclides reach the blood stream, the doses to other organs are calculated using exponential retention functions (ICRP 1959).

  Atmospheric concentrations can also be calculated by DACRIN using the bivariate normal distribution plume model. In addition, externally calculated dispersion factors may also be entered.

  Doses calculated in DACRIN are dependent upon the values of the release time and dose time used as input. Therefore, the doses that can be calculated for a maximally exposed individual include a 1-yr dose, 50-yr dose commitment, and cumulative dose.

  The code is documented (Houston et al. 1974; Strenge 1975) and is available from Pacific Northwest Laboratory (PNL), the Radiation Shielding Information Center (RSIC) at Oak Ridge National Laboratory, and the National Energy Software Center at Argonne National Laboratory.

- SUBDOSA--This program (Strenge et al. 1975) is used to calculate air submersion doses from accidental atmospheric releases of radionuclides. A space integration over the plume volume is performed. Dose results are reported for skin, male gonads, and whole body. Corresponding tissue depths are 0.007 cm (0.003 in.), 1.0 cm (0.4 in.), and 5.0 cm (2 in.), respectively. Doses are calculated for releases within each of several release time intervals. Up to six time intervals can be allowed and separate radionuclide inventories and atmospheric dispersion conditions can be considered for each time interval. Normally a 1-yr dose for the maximally exposed individual is calculated, which is equivalent to a dose commitment.

  The code is documented (Strenge 1975) and is available from PNL or RSIC.

- ALLDOS--This program is a report writer that takes input and output files from DACRIN and SUBDOSA and reformats them to produce a consistent and understandable report of the results.

5.4.2.2 Results of the Dose Assessment Runs. Radiological doses were calculated for six different initiators, with both filtered and unfiltered doses being calculated for five of those. The rationale for choosing the initiators is discussed in section 5.1.2, and the results of the dose assessment code runs are summarized in table 10.

Initiators A, B, C, and F are all based on the hypothetical cask drop accident. In all cases, it is assumed that the impact disrupts the waste transfer cask, the waste container, and the fuel rod cladding with sufficient force to create new particulate material amounting to 11 wt% of the spent fuel mass (in addition to the 0.5 wt% of existing particulate due to thermal cycling and spalling). Of the 11 wt%, 1.5 wt% consists of particles smaller than 10 um (3.94×10^{-4} in.) and 9.5 wt% of particles larger than 10 um (3.94×10^{-4} in.).

Initiator A is the cask drop with the cask hitting the bottom of the waste handling shaft, and the small particles being entrained in the slow air flow up the shaft. Initiator B is a cask drop source term assumed to be produced in repository drifts by some undefined accident (e.g., drift collapse, large earthquake) near the confinement exhaust shaft R6. It is assumed to occur close enough to R6 (in the Phase I emplacement drifts) that the large particulate is also entrained in the confinement exhaust airflow and carried to the surface. Initiator D is a similar accident occurring in drifts near confinement exhaust shaft R7 (but not so near that the large particulate is carried to the surface). Initiator F is a cask drop falling into water (so that no particulate is put into the air) with a large fraction of the spent fuel mass being carried as a sludge to the surface retention pond by the dewatering system. A failure of the associated radwaste treatment facility, evaporation of pond water, resuspension of a fraction of the radionuclides, and atmospheric transport to the site boundary was assumed. In addition, the analysis considered movement of radionuclides with groundwater to the Columbia River and the resulting

ingestion dose. Finally, Initiator D/E is the Fire/Explosion Source Term (these are combined because the analysis suggested little difference in the releases). It differs from the cask drop in involving less particulate (no new particulate) but a higher volatile radionuclide release than the cask drop.

The results of the dose assessment are similar for all but Initiator F. Unfiltered doses range from 650 to 2,300 rem; the corresponding filtered doses are in the range from 6 to 10 mrem, except for the fire/explosion initiator which is 49 mrem. In the dewatering system case (Initiator F), the resuspension dose is due to strontium and actinides, and the groundwater/ingestion dose is due almost entirely to strontium.

Two reasons exist for analyzing the consequences and preparing event trees for accident initiators that, under the rules of the methodolgy used, apparently could be eliminated on the basis of below-threshold probability of the accident initiator. One reason is a general impulse to conservatism in the consideration of such plausible and potentially significant accidents as a serious fire or explosion (particularly in view of the tentative nature of much of the quantitative data). The other reason is the developing NRC position on the repository Q-List (NRC 1986). The NRC has indicated that:

> "All structures, systems, and components of the geologic repository that could, irrespective of the probability of failure, initiate an accident which if unmitigated could cause an off-site radiation dose of 0.5 rem should be on the Q-List." (NRC 1986)

5.5 SUBSURFACE FACILITY Q-LIST

Based on the results of methodology A described in this chapter, the subsurface facility systems and components need to be Q-Listed are detailed below.

1. The confinement exhaust filtration subsystem of the ventilation system need to be Q-Listed. The function that needs to be maintained is the filtration of confinement exhaust upon demand. As such, failures that lead to temporary loss of confinement ventilation flow do not present a hazard to the public. On the other hand, failures that can lead to an unfiltered bypass flow path at a time when filtration is required because of a subsurface release of radionuclides cannot be tolerated.

   Thus, the radiation detectors and the repository utilities that support the detectors need to be Q-Listed. The actuation systems responsible for bringing the confinement exhaust filtration on-line when it receives an actuation signal are Q-Listed. The ductwork leading to the filters and the filter building and its airlocks are all Q-Listed. The confinement exhaust fans do not need to be Q-Listed unless their failure would lead to establishment of a unfiltered bypass flow.

2. Similarly, the radiation detection function and the radionuclide removal function on the dewatering system flow to the surface need to be Q-Listed.

3. Engineered systems whose failure could lead to a significant accident initiator are required by the OCRWM methodology and will apparently be required by the NRC Generic Technical Position (now out in draft form) to be Q-Listed. For the basalt repository, these include:

   - The waste handling hoist system--those functions that prevent a hoist cage drop
   - The subsurface explosives magazines
   - The subsurface diesel storage tank.

6.0 IMPORTANT-TO-WASTE ISOLATION Q-LIST (RETRIEVABILITY)

6.1 INTRODUCTION

A very coarse evaluation was performed in this area due to two factors: lack of definition of the importance of systems to retrievability and the symmetry of normal retrieval with normal emplacement. A BWIP Retrievability Task Force is developing information to help solve the first issue. The second observation leads to the conclusion that the understanding of what is important to safety for emplacement will provide most of what is needed for retrievability under a range of conditions up to and including several design-basis events. The Retrievability Task Force will provide information on design-basis events that, if they occur, may require special engineered features to preserve the retrievability option. When the Retrievability Task Force and the repository designers have adequately defined retrievability scenarios and the techniques that will be used for retrieval, the Q-List will be updated to fully implement the methodology described in section 3.3.4.

6.2 RETRIEVABILITY REQUIREMENTS

It is planned to evaluate for importance to retrievability using methods similar to those used in chapter 5. However, consequence screening will be based on technical judgement as to whether the event sequence renders retrievability impossible or impractical. A range of systems will be considered to deal with the aftermath of design-basis events to show that, given that such events have occurred, the emplaced waste can be safely retrieved while maintaining radiation releases to the environment and the public within acceptable levels. If the failure of facility equipment could render retrieval impossible or impractical, then the equipment would be placed on the Retrievability Q-List.

6.3 PRELIMINARY EVALUATION OF THE RETRIEVABILITY Q-LIST

Most of the systems, structures, and components listed for the subsurface important to safety Q-List are also important to retrievability in maintaining the safety of the waste packages, the configuration of the subsurface development and emplacement rooms, the workability of safety systems, and safe working conditions for facility operators. Some of these systems are also important in recovering from design-basis events in order to return the facility to conditions needed to begin recovery (e.g., dewatering, fire suppression, ventilation, and cooling systems).

There are three main classes of conditions that could seriously hamper retrievability operations should they occur: large-scale flooding, large-scale drift collapse, and significant radiation releases from multiple waste packages. As discussed in chapter 5, these events are of low probability. Large scale flooding poses some recovery problems, but much larger dewatering systems could be installed eventually and the emplacement rooms

could then be dried out. Large-scale drift collapse could in most cases be recovered by standard mucking techniques.

Significant radiation releases from multiple waste packages would be more difficult to deal with, but there are techniques available in the nuclear industry to recover from serious contamination, including the use of fixing coatings, shielding for workers, and shortened work shifts. Assuring that the waste package will not be breached in any design-basis event is one of the best protections to ensure retrieval. The only other apparent mechanism for significant radiation releases from multiple waste packages is systematic corrosion-caused failures of the waste package. Therefore, that aspect of waste package design that works to prevent early corrosion failure of the waste package (including its welds) should be Q-Listed on the basis of impact on both retrievability and long-term waste isolation.

6.4 CONCLUSIONS AND RECOMMENDATIONS

Most of the important-to-safety systems that maintain safety during emplacement are also important to safety during retrieval. Maintaining radioactive material containment in the waste package is also important to retrieval. The Q-List Preparation Team will be in a better position to develop the retrieval Q-List when the results of the Retrievability Study are available from the Retrievability Task Force.

7.0 IMPORTANT-TO-WASTE ISOLATION Q-LIST (POSTCLOSURE)

The waste isolation portion of the Q-List consists of items and activities that may have a significant influence on long-term performance of waste isolation in both the preclosure and postclosure phases of repository operation.

Items important to waste isolation are those natural and engineered barriers for which credit will be taken in determining that the repository meets the waste isolation requirements of 10 CFR 60.112.

Activities important to waste isolation are those site characterization program activities that meet the following criteria:

- Have the potential for deleterious impact on the waste isolation performance of the essential natural and engineered barriers
- Collect data for refining models of the natural barriers or data for better defining the design requirements for the engineered barriers
- Develop the methodologies, models, and computer programs used in prediction of waste isolation performance
- Comprise the performance assessment itself.

At the SCP stage of repository design, the Important-to-Waste Isolation Q-List is derived using technical judgment, because site characterization and design activities are in an early stage. The final Repository Q-List at licence application stage will have the benefit of formal performance assessments and will incorporate full consideration of site characterization, repository design, and the impacts of credible scenarios and events on long-term waste isolation.

7.1 ENGINEERED BARRIERS

The engineered barriers that are expected to be essential to repository waste isolation performance (and which are Q-Listed) are given below:

1. Waste container. 10 CFR 60.113(a)(1)(i)(A) requires substantially complete containment of radionuclides during the period when radiation and thermal output of the waste is dominated by fission product decay (required by 10 CFR 60.113(a)(1)(ii)(A) to be more than the first 300 yr but less than the first 1,000 yr). The waste container will be relied on to meet this requirement.

2. Waste package packing. 10 CFR 60.113(a)(1)(i)(B) requires that releases from the engineered barrier systems to the geologic setting be gradual over a long time period. The packing material between the waste container and the wall of the emplacement borehole will be relied on to meet this function.

3. Repository shaft and (vertical) borehole permanent seals. It may be necessary to rely on the performance of permanent shaft and borehole seals to ensure that the shafts and boreholes do not constitute a significant pathway for radionuclide transport to the accessible environment that bypasses the natural (geologic) barriers.

4. Damaged rock zone. The damaged rock around the emplacement boreholes and drifts is in a negative sense an engineered barrier. Drilling and blasting must be controlled so as to minimize the extent of the damaged rock zone and the degradation of the host rock permeability.

7.2 NATURAL BARRIERS

The portion of the geologic setting that is relied on to inhibit radionuclide transport to the accessible environment is the volume of rock and groundwater identified as the Repository Isolation Zone (RIZ). The horizontal extent of the RIZ coincides during the site characterization program with the Controlled Area Study Zone (CASZ), which consists of the sum of all likely controlled areas and represents the focus area for site characterization activities that will be used to determine the actual controlled area, as defined in 40 CFR 191.2 (EPA 1986). The CASZ boundary lies 5 km (3.1 mi) beyond the proposed outer edge of the underground facilities. The RIZ upper boundary is the top of the Priest Rapids Member of the Wanapum Basalt (the bottom of the Mabton interbed); the lower boundary is the contact between the N2 and R2 magnetostratigraphic units in the Grande Ronde Basalt, three to six flows below the Umtanum flow.

Future evaluation will result in the selection of the final controlled area (which will provide the horizontal boundaries of the RIZ) required by federal regulations. Also, the vertical extent of the RIZ may be decreased if assessments show that a smaller vertical section of the RIZ provides an adequate barrier to radionuclide transport.

7.3 ACTIVITIES RELATED TO WASTE ISOLATION

To assure that items on the Important-to-Waste Isolation Q-List will satisfactorily perform their intended function of isolating radionuclides from the accessible environment, site characterization activities and preclosure phase events must be controlled.

The conventional application of a nuclear quality assurance program to engineered items provides proper identification and control of those activities affecting the quality of the engineered items (e.g., as design, procurement, and fabrication). For the engineered barriers, it is necessary to ensure that the design fulfills its functional requirements and that procurement and fabrication are in conformance with the design requirements.

For the natural barriers, it is necessary to demonstrate that we understand and are able to predict the performance of the barriers, and that the overall performance will meet the regulatory requirements.

Activities to be carried out during the site characterization phase have not all been identified or planned in detail. In addition, it is expected that plans for activities later in the site characterization program will be adjusted and modified on the basis of evaluation of earlier activities. For these reasons, activities affecting quality are not listed individually in this report; rather, criteria are described that allow for evaluation of any intended activity for its impact on Q-Listed items and its importance to assessment of the waste isolation performance of the repository. Activities affecting quality fall into the four categories listed below.

1. Activities that produce a Q-Listed item (i.e., design and construction of engineered barriers), and activities that involve modification of a natural barrier. Examples of activities in this category include drilling boreholes and excavating underground openings in the RIZ and the design and construction of permanent seals.

2. Data collection activities that provide the basis for understanding and confirming the waste isolation performance characteristics of both engineered and natural barriers. These activities must be controlled to ensure the integrity and appropriateness of the data collected.

3. The activities that comprise the actual model-building that embodies the understanding of the long-term waste isolation performance of the natural and engineered barriers and that is the basis for the development of methods for predicting the waste isolation performance.

4. The performance assessment process of formally evaluating the waste isolation performance of the engineered barriers and the natural barriers against the performance requirements of 10 CFR 60.113, including consideration of credible processes and events to which the barriers may be subjected.

In addition to the criteria for classification of site characterization program activities, credible events occurring during the repository preclosure operational phase that may impact the long-term waste isolation performance of the repository require that preventive or mitigative systems be placed on the Important-to-Waste Isolation Q-List. The only such item that has been identified in this analysis is the subsurface explosives magazine. An inadvertent explosion of the magazine during development of the repository would involve damage to surrounding and overlying rock, which might compromise the waste isolation capability of the repository horizon flow interior.

7.4 Q-LISTING OF ACTIVITIES

Activities that fall within the categories identified above shall be controlled insofar as they affect the performance characteristics of a barrier, the understanding of those performance characteristics, or the prediction of barrier performance. For example, if the waste packing includes bentonite clay, and if the color of the bentonite does not affect its waste isolation performance, then the color does not need to be specified and controlled. On the other hand, if the composition and handling of drilling fluids affects the ability to properly interpret the analysis of a borehole sample, then the use of those fluids must be controlled so that it does not degrade the quality of the data collected from that borehole.

To better illustrate what is intended by Q-Listing site characterization program activities, using a not-so-hypothetical example, consider borehole X, which will be drilled near the repository, will penetrate to the level of the repository and will be used to collect groundwater pressure data and groundwater chemistry samples. By criterion 2 of Section 7.3 of this report, borehole X is a site characterization program activity that must be Q-Listed. The Q-Listing imposes on the activity's project manager the obligation to conduct an evaluation of the activity and its component subactivities to determine appropriate quality levels for each of the subactivities. This evaluation is to be conducted by the technical personnel involved in the overall activity with the approval of the Quality Assurance Department. For this example, the technical personnel should include project management, drillers, project engineers, and the hydrologists and geochemists who will be the consumers of the data generated in the borehole. The evaluation should consider each of the subactivities in the light of its importance to the ultimate goals of the borehole, the pressure and chemical data that will be used to refine the conceptual model of the site or used to provide input data for the computer models being developed to predict the long term performance of the repository.

Those subactivities important to the ultimate quality of the data collected would be assigned Quality Level 1; subactivities that cannot impact the quality of the data might be assigned Quality Level 3. Even among subactivities assigned Quality Level 1, there will be a wide variety of quality assurance programs used, depending, for instance, on whether the subactivities represent a first-time, one-of-a-kind, design and fabrication activity or merely the use of off-the-shelf equipment of known quality and good reliability. The Graded Quality Assurance Program requires an evaluation of the activity and a selection of those requirements (and only those requirements) from NQA-1 (ANSI/ASME 1986) and its supplements that are needed to adequately control the quality of the activity.

This process, carried out for the not-so-hypothetical borehole X, might produce the result that most aspects of the drilling of the borehole do not impact the quality of the data to be collected, and thus can be assigned Quality Level 3. On the other hand, geophysical logging and the handling of drilling fluid when the borehole approaches the target horizon might be judged to have significant impact on the quality of the data, with the result that they are assigned Quality Level 1.

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APPENDIX A

CALCULATIONS USED IN DETERMINATION OF THE PRELIMINARY BASALT REPOSITORY Q-LIST

This appendix details the calculations used by the Q-List Preparation Team to support estimation of probabilities and consequences associated with various accident scenarios. Each calculation will appear under a heading that keys it to the section of the text of the report in which it is used.

GENERAL INFORMATION

The rigorous mathematical distinction between the probability of occurrence of some event during a given time period and the frequency of occurrency of that event (which is the same as the expected number of occurrences of that event in the specified time period) is often suppressed throughout this document. The justification for this is that when the probability is less than 0.1/yr, the frequency (per year) becomes approximately equal to the probability. It is only as the probability increases above 0.1, that the probability and the frequency diverge significantly. Thus, a frequency of occurrence of 2/yr for a given event does not imply that the probability of that event occurring in a given year is 2.0, since probabilities by definition are always less than or equal to 1.0.

SECTION 5.1.1.1.2

An estimate for the probability of occurrence of earthquakes producing new faults in the reference repository location is developed in this section. The area of the repository plus a buffer region is taken to be 15 km^2 (5.8 mi^2). From a recurrence relation given in the final safety analysis report for the Washington Public Power Supply System nuclear powerplant WNP-2 (WPPSS 1981), a recurrence rate is obtained for earthquakes of local Richter magnitude M = 4.4 or greater occurring in the Central Columbia Plateau (an area of approximately 10,600 km^2 (4,100 mi^2) being instrumentally monitored) of 0.05 events per year.

Thus the probability, P1, of an earthquake greater than M = 4.4 ocurring within the 15 km^2 (5.8 mi^2) repository plus buffer is:

$$P1 = 0.05/yr \times 15\ km^2/10{,}600\ km^2$$

or

$$P1 = 7.1 \times 10^{-5}/yr.$$

Given that a strong earthquake occurs, there is some lesser probability that a new fault is created. The conditional probability, P2, that a new fault is created is estimated to be 0.01. The uncertainty in this estimate is represented explicitly by assuming that P2 is itself a random variable, log-normally distributed with the mean and the standard deviation of the log-transformed distribution being -2.0 and 0.5, respectively. The effect of this assumption is that an approximate 95% confidence interval for the value of P2 is given by:

$$0.001 < P2 < 0.1$$

Then the probability, P3, that a new fault is created within the repository plus buffer region is:

$$P3 = P1 \times P2,$$

which implies that a two-tailed 95% confidence interval for P3 is:

$$7.1 \times 10^{-8} < P3 < 7.1 \times 10^{-6}.$$

Both assignment of a point probability value to P1 and the assignment of an inputed distribution and confidence interval to P2 represent an extrapolation of the available data. The two largest events in the instrumental seismic record for the Central Columbia Plateau are magnitude 4.4 events north of the Saddle Mountain anticline. The data in the instrumental record correlate very well (linearly on a log-log graph) for coda magnitudes between 2.0 and 4.0. With respect to the conditional probability of a given earthquake larger than magnitude 4.4 creating a new fault (as opposed to causing movement along an existing fault), little quantitative information appears to be available. The explicit assignment of as log-normal distribution to the random variable P2 is in the tradition of risk analysts faced with uncertain data.

Since a decision to discard this initiating event was based on this extrapolated estimate of recurrence rates and an estimate of the balance between earthquakes that create new faults and those that cause movement along existing faults, conservatisms in the analysis that have not been explicitly quantified are addressed below.

- The map of instrumentally recorded seismic events shows almost all of the events occurring along known fault zones or around known earthquake swarm foci, and almost none occurring within the boundaries of the reference repository location. Thus, a probability estimate calculated using a simple spatial averaging of the data will probably significantly overestimate the occurrence rates of earthquakes with the reference repository location (Caggiano and Duncan 1983, chapter 6).

- Geometric considerations suggest that many earthquakes occurring within the reference repository location will not intersect any waste packages and hence will not subject packages to shearing forces.

- As indicated in the body of the report, seismic events provide little energy for effective dispersion of radionuclides from a failed waste container, hence the systems designed to mitigate the impact of the cask drop and fire/explosion events can be expected to mitigate the effects of a seismic event as well.

SECTION 5.1.1.5.1

Most rock bursts occur near the development face within a week of excavation of material. Data from a mine in the Coeur d' Alene mining district give a "serious" rock burst rate of 8.8×10^{-7}/t (8.0×10^{-7}/ton) of rock removed for deep mines. Quantitative data on rock burst frequencies are difficult to obtain since they tend to depend on rock type, internal stress, and mine geometry. The data used here are from a mine with fractured hard rock and higher internal stress and extraction ratio than for the basalt repository. Its use as a surrogate for the basalt repository is probably conservative. For the approximately 9.1×10^5 t (1.0×10^6 tons) of rock per year planned removal rate at the reference repository location, this would suggest a rate of serious rock bursts of 0.8/yr. Assuming that 90% of these occur within the development area before emplacement is attempted, the rate of rock bursts for the confinement areas would be 0.08/yr.

Normalizing this expected frequency of "serious" rock bursts (with "serious" defined as 45 t of rock or a mine fatality) to the total length of drift in the repository (approximately 40 km at the beginning of emplacement and 240 km by the end of the emplacement period) and per second we get:

f1 = frequency per meter of drift per year.

3.32×10^{-7} rock bursts/m/yr < f1 < 1.99×10^{-6} bursts/m/yr.

f2 = frequency per meter of drift per second.

1.05×10^{-14} rock bursts/m/s < f2 < 6.30×10^{-14} bursts/m/s.

Assuming that at any instant that a waste transporter is exposed to rock bursts, it will be exposed to bursts occurring along the surrounding 15.3 m of drift. Since there will be approximately 2,500 waste emplacements per year (using 51,000 total waste containers emplaced during 22 yr of emplacement), each requiring approximately 1 h of exposure of the loaded

waste transporter to rock bursts, the total probability of a loaded waste transporter being hit during the year by a rock burst can be calculated as:

p1 = probability of rock burst hitting loaded transporter.

$1.05 \times 10^{-14} \times 15.3$ m x 2,500 trips x 3,600 s = 1.44×10^{-6}/yr.

$6.30 \times 10^{-14} \times 15.3$ m x 2,500 trips x 3,600 s = 8.65×10^{-6}/yr

So, the probability, p1, is bounded by the two values calculated above:

$1.44 \times 10^{-6}/yr < p1 < 8.65 \times 10^{-6}/yr$

SECTION 5.1.1.6.1.

A fire involving the diesel fuel tank on the underground waste transporter would be expected to heat up the waste container if it is on board. Bounding calculations establish that even assuming that the total heat of combustion from 190 L (50 gal) of diesel fuel is put into the waste cask and waste container, pressure inside the waste container would only increase by a factor approximately equal to two and a half, not enough to fail the container.

The weight of cask and waste container is 40,800 kg (56 tons).

Of that weight, assume that 3/4th is lead shielding, and 1/4th is carbon steel.

Specific heat of carbon steel at 400 °C = 600 J/kg/°K.

Specific heat of lead at 25 °C = 130 J/kg/°K.

Average specific heat of lead and steel =
0.25 x 600 + 0.75 x 130 = 247 J/kg/°K.

Heat of fusion of lead (at 327.5 °C) = 23,000 J/kg.

The thermal properties are taken from Bolz and Tuve (1973).

Assume 188 L (50 gal) of diesel fuel available in the waste transporter fuel tank.

Specific weight of diesel fuel = 0.92 kg/L (Bolz and Tuve 1973).

Weight of diesel fuel in tank = 173 kg.

Heat of combustion of diesel fuel = 4.44×10^7 J/kg (Bolz and Tuve 1973).

Total heat generated from combustion of the diesel fuel in the waste transporter tank = 7.68×10^9 Joules.

Assume that the waste container and cask are initially at an average temperature of 150 °C, that all of the heat of combustion is put into the steel and lead, and that the lead begins melting at 327.5 °C.

Heat required to heat lead and steel to 327.5 °C = 247.5 J/kg/°K x 40,800 kg x (327.5 °K - 150 °K) = 1.79×10^9 J.

Heat required to melt 30,600 kg of lead = 30,600 kg x 23,000 J/kg = 7.0×10^8 J.

Remaining heat = 7.68×10^9 J - 1.79×10^9 J - 0.7×10^9 J = 5.19×10^9 J.

Temperature increase above 327.5 °C = 5.19×10^9 J/(40,800 kg x 247.5 J/kg/°K) = 514 °K.

Assuming ideal gas inside the waste container, a final pressure is obtained of

$$P2 = P1 \times (T2/T1) = P1 \times ((514 + 327.5 + 273)/(150 + 273)) = P1 \times 2.64,$$

where P1 was the initial pressure inside the waste container (at 150 °C).

It is reasonable to assume that the waste container will not fail during a waste transporter fuel tank fire because of the reasons listed below.

1. The pressure increase is modest, even accounting for the loss of strength in the steel due to the temperature increase.

2. The assumption that all of the heat from the fire is put into the steel and lead of the cask and waste container is highly conservative.

3. There is no accounting for radiative losses during the heating of the cask and waste container.

SECTION 5.1.1.6.2

An estimate of the probability of a diesel fire occurring while a waste transporter is being refueled was developed, using as a basis general United States truck transportation statistics. The intuitive rationale for this approach is that, even for gasoline, fires occurring during refueling are relatively rare, particularly when considering that there are on the order of 2.5×10^9 refuelings of gasoline vehicles each year. Moreover, diesel fuel is significantly less volatile than gasoline and hence less hazardous.

The previously mentioned 1969 to 1972 trucking industry accident data (Dennis et al. 1978) suggest that the number of fires resulting from these refuelings is less than 10/yr. This leads to an estimated probability of a

diesel fire occurring during a refueling of less than 1.6×10^{-9}/refueling. Assuming 32,000 waste transporter km/yr (20,000 mi/yr) and 100 refuelings (which implies a 78 to 156 L (20 to 40 gal) tank rather than a 190 L (50 gal) tank), the estimated probability of a waste transporter refueling fire is 1.6×10^{-7}/yr. As before, the trucking industry data involve no special design provisions, no special training, and include gasoline-engine trucks.

SECTION 5.1.1.7.2

The incidence of accidental explosions in underground development is on the order of 3.0×10^{-6}/yr at a 95% confidence level. This estimate is based on mining statistics that suggest there are on the order of 1 million trips per year transporting explosives between the magazines and the development faces together with assertions by Kaiser Engineers, Inc./Parsons Brinckerhoff Quade & Douglas, Inc. mining engineers that mine accidents involving inadvertent explosions during transportation to the development face "don't happen."

If we model the transportation of the explosives as a set of Bernoulli trials with 1 million trips with a probability, p, of an explosion during any one trip, then the probability of no explosions during those million trips would be:

$p1 = P(\text{no explosion}) = (1 - p)^{(1.0 \times 10^6)}$ {= (1-p) raised to the power one million}.

If $p = 3.0 \times 10^{-6}$, then p1 is approximately equal to 0.05, which implies the following.

If the true probability, p, of an inadvertent explosion during a trip from the magazine to the development were greater than 3.0×10^{-6}, then the there would be less than 1 chance in 20 of getting through a year with no such explosions occurring during the one million trips. In the standard notation, 3.0×10^{-6} is a 95% confidence upper bound for the "true" value of p.

For the basalt repository, we can assume one trip per working day between the magazines and each of the working development faces of the repository, or seven trips in all. This implies that the probability of an explosives transportation inadvertent explosion during any given year would be approximately

7×250 explosives transportation trips $\times 3.0 \times 10^{-6}$/trip
$= 5.25 \times 10^{-3}$/yr.

Since there is no particular reason for the explosives transporter to be violating the separation between the repository development side and the emplacement side, it is reasonable to conclude that the combined probability of an explosives transportation inadvertent explosion occurring next to a waste transporter loaded with a waste container should be well below the threshold probability of 1.0×10^{-5} (with 95% confidence).

SECTION 5.1.1.7.3

As with inadvertent transportation explosions, magazine design provisions and administrative controls are sufficient that magazine explosions "don't occur." The basis for this conclusion is the 25 yr of experience of the mining engineer who participated on the Q-List Preparation Team combined with a quick review of mine safety statistics. The accidental mine explosions occurring in United States metal mines during the last 20 years seem all to have occurred at the development face in the course of placing the charge or to have involved methane. Interestingly, there were several fires during that time in which stored explosives burned without exploding.

We can use the same method of modelling the occurrence of magazine explosions as a Bernoulli process to obtain a 95% confidence level upper bound for the "true" probability of a magazine explosion. With on the order of 1,500 magazines in underground metallic mines, the non-occurrence of magazines explosions leads to a 95% confidence level upper bound for the true probability of 2.0×10^{-3}/magazine/yr (Assuming only 1 yr of such non-occurrence. Twenty years of no explosions would divide this probability by twenty, giving a probability of 1.0×10^{-4}/yr).

SECTION 5.1.1.10.2

The probability of losing all repository AC power sources is the product of the unavailability of the main power source and the probability (per demand) of the emergency power systems failing to start and run. As indicated in section 5.1.1.10.1, the probability of loss of both offsite AC power sources (for more than an hour) is on the order of 0.015/yr. For nuclear power plants and 1-E emergency power systems, the latter probability is on the order of $0.04 \times 0.1 = 0.004$ (Harris 1985, table 7-1) where the second factor is the conditional probability that the second emergency generator fails to start and run, given that the first has failed. This conditional probability accounts for common-cause failures of the emergency generators.

Therefore, the probability of a combined failure of offsite power and both emergency systems is 0.015/yr x 0.004/demand = 6.0×10^{-5}/yr.

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APPENDIX B

LETTER FROM D. J. BRADLEY TO G. S. HUNT

Battelle
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April 14, 1987

G. S. Hunt, Manager
Management and Integration
Site Characterization Program
Basalt Waste Isolation Project
Rockwell Hanford Operations
P.O. Box 800
Richland, Washington 99352

Dear Mr. Hunt:

Subject: Final Report for the FY 1986 Task 3 of the Q-List Task Force Technical Support Project, Fiscal Year 1987 Statement of Work for Basalt Waste Isolation Project Preclosure Safety Assessment (L1E4C)

Reference: Letter to D. J. Bradley from G. S. Hunt dated March 16, 1987, RHO Letter No. R87-1075

Enclosed is the final report for the FY 1986 Task 3 of the Q-List Task Force Technical Support Project. The report describes potential radiation doses to offsite individuals from the various accident scenarios provided by BWIP. Please direct any questions on this report to B. A. Napier, 375-3896. This completes the FY 1986 Task 3 project requirements. The project files will be forwarded to Rockwell following final QA approval.

Also enclosed is the FY 1987 PNL approved Statement of Work "Development of Incident/Accident Statistics and Risk Assessment Planning in Support of Basalt Waste Isolation Project (BWIP) Preclosure Safety Assessment", L1E4C, Revision 0.

If you have any questions on this Statement of Work, please call me or B. A. Fecht on 376-4264.

Sincerely,

for

D. J. Bradley, Manager
Waste Package and Performance
Assessment Department

DJB:ksg

cc: JS Dukelow, RHO, with enclosure
BA Fecht, PNL, with enclosure
JV Livingston, RHO, with enclosure
MW Rosenberry, RHO, with enclosure

RECEIVED
APR 16 1987
G. S. HUNT

BWIP SITE CHARACTERIZATION PLAN Q-LIST METHODOLOGY IMPLEMENTATION: Q-LIST TASK FORCE TECHNICAL SUPPORT PROJECT

FINAL REPORT - TASK 3: POTENTIAL RADIATION DOSES TO OFFSITE INDIVIDUALS FROM POSTULATED REPOSITORY ACCIDENT SEQUENCES

As part of the preliminary performance assessment in support of BWIP site characterization, Rockwell Hanford Operations asked the Pacific Northwest Laboratory to provide estimates of the potential offsite radiological impacts associated with postulated operating accidents that might occur within the subsurface repository facility. Radionuclide releases were estimated by the Rockwell Q-List Task Force. Task 3 of the effort is to provide estimates of the potential radiation dose a maximally exposed offsite individual could receive as a result of these events, for various facility design parameters. These estimates are given in this report.

RADIONUCLIDE RELEASE

The BWIP Q-List Task Force provided estimates of the radiological releases that could be expected as a result of the various scenarios considered. These are summarized in Table 1.

Fuel elements irradiated to 60,000 MWd/MT with 5-year cooling were used as upper-bound source terms. Radionuclide inventories for this case was provided by Rockwell; these are reproduced in Table 2, normalized to one MT.

The fuel irradiated to 60,000 MWd/MT produces 3640 watts/MT after a 5-year cooling period. The current design limits for the Shielded Transport Container (STC) are 2200 watts (memo, S. FitzPatrick to B. Napier, 7/11/1985). Therefore, the inventories presented in Table 2 were modified by ratio to estimate the maximum allowable STC contents of each; i.e. a mass of 0.604 MT for the 60,000 MWd/MT inventory.

Table 1. Summary of Release Initiators, Release Fractions, and Distances to the Site Boundary for BWIP Q-List Postulated Accident Sequences

A. Waste cask drop down waste handling shaft, 6000 feet to Highway 240

Noble gases	100%
Halogens	10%
Volatiles (Cs,Te)	10%
Particles (<10 μm AMAD)	2%

B. Cask drop arbitrarily assumed to occur at a Phase I emplacement location 200 feet from confinement exhaust shaft R6, 7600 feet from Highway 240

Noble gases	100%
Halogens	10%
Volatiles (Cs,Te)	10%
Particles (<10 μm AMAD)	2%
Particles (10 to 150 μm AMAD)	9.5%

C. Cask drop release arbitrarily assumed to occur at a Phase II emplacement location 2000 feet from confinement exhaust shaft R7, 3400 feet from Highway 240

Noble gases	100%
Halogens	10%
Volatiles (Cs,Te)	10%
Particles (<10 μm AMAD)	2%

D./E. Diesel fuel fire causing over-pressure failure of the waste container, OR an explosion causing partial fragmentation of the waste container, 3400 feet from Highway 240

Noble gases	100%
Halogens	50%
Volatiles (Cs,Te)	50%
Particles (<10 μm AMAD)	0.5%

F. Cask drop into water pumped to outdoor retention pond without passing through radwaste treatment system (pond assumed to be 1 mile from Highway 240)

	To air	To water
Noble gases	100%	--
Halogens	--	20%
Volatiles (Cs,Te)	--	20%
Particles (<10 μm AMAD)	--	50%

TABLE 2. Radionuclide Inventories for Fuel Compositions Studied, Curies/MT

Radionuclide	60,000 MWd/MT 5-year Cooled
H3	9.38E+02
C14	2.44E+00
MN54	1.79E+01
FE55	1.80E+03
CO60	5.64E+03
NI59	6.40E+00
NI63	1.02E+03
ZN65	7.51E-01
KR85	1.03E+04
SR90	1.03E+05
Y90	1.03E+05
ZR93	2.48E-01
NB94	2.26E+00
TC99	2.11E+01
RU106	2.45E+04
RH106	2.45E+04
AG110M	6.46E+01
CD113M	1.03E+02
SN119M	3.79E+01
SN121M	9.47E+01
SB125	6.61E+03
TE125M	1.61E+03
CS134	6.83E+04
CS137	1.62E+05
BA137M	1.53E+05
CE144	1.22E+04
PR144	1.22E+04
PR144M	1.46E+02
PM147	3.26E+04
SM151	5.14E+02
EU154	1.69E+04
EU155	8.14E+03
U237	3.81E+00
NP239	7.21E+01
PU230	8.43E+03
PU239	3.67E+02
PU240	6.84E+02
PU241	1.55E+05
PU242	4.54E+00
AM241	1.67E+03
AM242	1.72E+01
AM242M	1.73E+01
AM243	7.21E+01
CM242	6.97E+01
CM243	8.29E+01
CM244	1.33E+04

Preliminary design concepts for the exhaust air filter system were provided by Rockwell. The conceptual design contains the following filtration efficiencies (memo. J. Dukelow to B. Napier, 9/18/1986):

Noble gases	0%
Halogens	50%
Volatiles (Cs,Te)	50%
Particles (<10 μm AMAD)	(1.0 - 2.5E-6)100%

Additional consideration must be given to the pond release postulated in Accident Sequence F. For this sequence, 100% of the noble gases are assumed to be released to the air, but the remaining radionuclides are first pumped with the water into a pond. To enable a first approximation of the dose to the individual at the site boundary, a release model from the pond to the air is needed. For simplicity, it is conservatively assumed that the halogens and volatiles will be dissolved in the water, and can evaporate congruently with it. The particles are assumed to deposit and be resuspended from the pond banks. It is assumed the pond covers an area of one hectare (10^4 m^2) and is one meter deep. The evaporation rate is assumed to be 1 cm/day, or 10^{-2} of the pond volume per day. The release fraction of the volatiles and halogens in the pond is then also 10^{-2}/day. It is further assumed that 4% of the particles settle on a one-meter wide shoreline around the pond, and are resuspended at a rate of 10^{-8} per second. Using these assumptions, the release fraction for particles in the pond is 3.5×10^{-5}/day.

RADIONUCLIDE TRANSPORT

The reference repository site defined in the Draft Environmental Assessment for the Hanford Site (U.S. DOE 1984) is located near the central portion of the site, approximately one mile to the northeast of State Highway 240. The maximally exposed individual is assumed to remain at this location--the closest point of approach for members of the general public--for the duration of the radionuclide release.

Radioactive material released to the atmosphere becomes diluted as it is carried by the wind away from the point of release. The degree of dilution and the resultant air concentrations are predicted through the use of the Gaussian plume model (NRC 1977) and onsite measurements of atmospheric conditions.

Atmospheric dispersion data (wind speed, wind direction, and atmospheric stability) for the 200 Areas are collected at the Hanford Meteorological Station (HMS), which has been in operation since 1945.

Because we cannot predict precisely when a hypothetical release would occur, we conservatively assume that the release coincides with adverse atmospheric conditions. This is accomplished by calculating dispersion based on the 95th percentile atmospheric conditions derived from the recorded hourly measurements of wind speed, wind direction, and atmospheric stability. These are the conditions that predict short-term (1-hour average) air concentration that would not be exceeded more than 5% of the time. For the individual, a plume centerline model is used, except for the releases from the pond, which occurs over a longer period so a sector-averaged value is used. For the combination of proposed repository location and Highway-240 individual, the value of time-integrated air concentration per unit release (E/Q), in units of sec/m^3 (Ci/m^3 per Ci released times time) is given in Table 3 for the distances indicated in Table 1 for the accident sequences. These values are interpolated from tabulated values in McCormack, Ramsdell and Napier (1984).

No reduction in air concentration is given for plume depletion by fallout or rainout. For gases and particles of the size postulated to be released, little to no deposition would be expected over the relatively short distances involved (Horst 1977).

Table 3. Time-integrated Air Concentration Parameters for Selected Distances, seconds/m^3

Distance	Ground-level Release E/Q	Elevated Release E/Q
3400 feet	5.0×10^{-4}	4.0×10^{-5}
5280 feet	1.0×10^{-4}	--
6000 feet	1.9×10^{-4}	3.0×10^{-5}
7600 feet	1.3×10^{-4}	2.6×10^{-5}

RADIATION DOSIMETRY MODELS

A set of computer programs has been developed at Hanford to calculate the dose consequences from all significant exposure pathways. The evaluation

of potential environmental radiation impacts is facilitated through the use of these computerized dose calculation programs. Each program assesses a common set of standardized libraries which, to the extent they are available, contain Hanford-specific data. The programs and data libraries are maintained by the Hanford Dose Overview Program, with all revisions or updates documented (McCormack, Ramsdell and Napier 1984). An overall dose model QA plan is in place and followed for all code development, revisions, and use. The computer programs have been documented separately, and only a brief description of their application is given here.

DACRIN - This program (Houston, Strenge and Watson 1974; Strenge 1975) is used to analyze radiation doses from inhalation for Hanford operations. The program uses the model of the ICRP Task Group on Lung Dynamics (ICRP 1966) to predict radionuclide movements through the respiratory system and lung doses. Once radionuclides reach the blood stream, the doses to organs other than the lung are calculated using exponential retention functions (ICRP 1959).

DACRIN can also calculate atmospheric concentrations using the Gaussian, bivariate, normal distribution plume model. However, externally calculated dispersion factors may also be entered.

Doses calculated in DACRIN are dependent upon the values of the release time and dose time used as input. Therefore, the doses that can be calculated for a maximally exposed individual(MI) include a one-year dose, dose commitment and cumulative dose.

DACRIN is written in FORTRAN and typically uses about 80K of computer memory during an average 3-minute run. The code is documented (Houston 1974, Strenge 1975) and is available from PNL, the Radiation Shielding Information Center (RSIC) at Oak Ridge, and the National Energy Software Center (NESC) at Argonne.

SUBDOSA - This program (Strenge, Watson and Houston 1975) is used to calculate air submersion doses from accidental atmospheric releases of radionuclides. A space integration over the plume volume is performed. Dose results are reported for skin, male gonads, and total body. Corresponding tissue depths are 0.007, 1.0, and 5.0 cm, respectively. Doses are calculated for releases within each of several release time intervals. Up to six time intervals can be allowed, and separate radionuclide inventories and atmospheric

dispersion conditions can be considered for each time interval. Normally, a one-year dose for the MI is calculated, which is equivalent to a dose commitment.

SUBDOSA is written in FORTRAN and typically uses about 50K of computer memory during an average 1-minute run. The code is documented (Strenge 1975) and is available from PNL or RSIC at Oak Ridge.

RADIATION DOSES

Radiation doses estimated for the individual assumed to remain along Highway 240 downwind during the duration of the accident are summarized in Table 4. Doses are presented for accidents involving 0.604 MT of 60,000 MWd/MT fuel cooled 5 years, for both ground-level and stack releases, with and without filtration on the exhaust gases for the elevated releases. The doses given are 50-year dose commitments from the single exposure.

The doses presented are dominated by the inhalation pathway. For no result shown does the air submersion dose (calculated with SUBDOSA) ever contribute more than one or two percent to the total. This is because the energetic, short-lived radionuclides have essentially all decayed from the hypothetical fuel mixtures.

The doses from the unfiltered releases are dominated by plutonium, curium, americium, and strontium. The doses from the filtered releases are controlled exclusively by cesium. This cesium domination is a direct result of considering the volatile radionuclides to have only a 50% removal through the HEPA filters. This result is probably conservative.

TABLE 4. Calculated Radiation Dose Commitments to an Individual on Highway 240 from Specified Releases, rem

Release Description	Organ			
	Total Body	Bone	Lung	GI Tract
A. Waste cask drop down waste handling shaft				
1. Unfiltered Releases				
a. Ground-Level	5.3E+2	1.1E+4	1.0E+4	1.2E+1
b. Elevated	8.3E+1	1.7E+3	1.6E+3	1.9E+0
2. Filtered Releases				
a. Elevated	3.6E+0	3.1E+0	8.9E-1	5.0E-2
B. Cask drop at Phase I location 200 feet from shaft R6				
1. Unfiltered Releases				
a. Ground-Level	3.6E+2	7.4E+3	7.1E+3	9.3E+0
b. Elevated	7.2E+1	1.5E+3	1.3E+3	1.6E+0
2. Filtered Releases				
a. Elevated	3.2E+0	2.8E+0	8.7E-1	1.6E-1
C. Cask drop at Phase II location 2000 feet from shaft R7				
1. Unfiltered Releases				
a. Ground-Level	1.4E+3	2.8E+4	2.7E+4	3.1E+1
b. Elevated	1.1E+2	2.3E+3	2.0E+3	2.5E+0
2. Filtered Releases				
a. Elevated	4.8E+0	4.2E+0	1.2E+0	6.7E-2

D/E. Diesel fuel fire OR explosion

1. Unfiltered Releases

a. Ground-Level	9.1E+2	7.6E+3	6.9E+3	1.6E+1
b. Elevated	7.3E+1	6.1E+2	5.2E+2	1.3E+0

2. Filtered Releases

a. Elevated	2.4E+1	2.1E+1	5.8E+0	1.8E-1

F. Cask drop into water, pond release

1. Unfiltered Releases from Pond

a. Ground-Level	6.9E-1	5.3E+0	4.5E+0	1.2E-2

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