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**Civilian Radioactive Waste Management System**
**Management & Operating Contractor**

# Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation

**TDR-MGR-SE-000009 REV 00 ICN 01**

**August 2000**

Prepared for:

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ENCLOSURE

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**August 2000**

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## CHANGE HISTORY

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# EXECUTIVE SUMMARY

## BACKGROUND

The U.S. Department of Energy has been investigating a repository site for spent nuclear fuel and high-level waste since 1983 at Yucca Mountain, in Nevada. In December 1998, the Department of Energy submitted a Viability Assessment to Congress and the President to allow an informed decision to be made concerning program direction and funding.

Based on the scientific study of Yucca Mountain, the Department of Energy believes that the site remains promising for development as a geologic repository. The Secretary of Energy has not yet decided whether or not to recommend the site to the President of the United States. That decision is scheduled for 2001, after issuance of a final environmental impact statement and an evaluation of the suitability of the site for development as a geologic repository.

This preliminary preclosure safety assessment supports the site recommendation sufficiency arguments. The results show that the Monitored Geologic Repository can operate in the preclosure period with minimal impact to the health and safety of the public and workers.

## FACILITY OVERVIEW

The mission of the Monitored Geologic Repository at Yucca Mountain is to safely dispose of the nation's spent nuclear fuel and high-level radioactive waste in such a way that it protects the health and safety of the facility worker, the public, and the environment. The Monitored Geologic Repository will receive spent nuclear fuel and vitrified high-level waste and prepare the waste for emplacement in the underground repository. The prepared waste will then be transported underground and deposited in excavated emplacement drifts. The waste will be monitored until such time that a decision is made to close the repository. Until the decision is made to close the repository, the option to retrieve the waste will remain open.

The site for the proposed repository is located in Nye County in Southern Nevada, approximately 161 kilometers (100 miles) northwest of Las Vegas, on land controlled by the U. S. Government. There are no permanent residents within 20 kilometers (12.5 miles) of the proposed facility. The closest permanent population concentration is in Amargosa Valley, a primarily agricultural-based community on the south edge of the Nevada Test Site. Little of the area surrounding Yucca Mountain is privately owned, and there is very little built-up or urban land. Close to the Yucca Mountain Site it is likely that a large percentage of the land will remain federally owned and controlled. In addition, the Nevada Test Site is withdrawn from public use entirely.

A surface complex of waste handling facilities that will include waste receipt and preparation processes will support the proposed repository. These waste handling facilities will be located at the North Portal to the subsurface facility and comprise a radiological controlled area. The major structures will include a carrier preparation building and a waste handling building with an attached structure for the management of site-generated low-level radioactive waste. Administrative and support facilities will be located at the North Portal outside the radiological controlled area. The subsurface facility, which is also within the radiological control area, will consist of development and emplacement areas separated by isolation barriers. The moveable

isolation barriers will allow emplacement operations while repository construction is underway. Separate access mains will provide access to the subsurface emplacement drifts where the waste will be placed and to construction areas on the development side of the repository. The Monitored Geologic Repository preclosure safety strategy provides general guidance for the establishment of system design requirements. Specific design bases and operating limits for the facilities evaluated in this assessment are described in the applicable System Description Documents.

Spent nuclear fuel and vitrified high-level waste will be transported to the repository north portal security station in certified casks by licensed cask transporters. Facility personnel will then verify the shipping manifests and inspect the cask and carrier. An onsite prime mover will then move the cask and carrier to the Carrier Preparation Building, where the cask is prepared for receipt in the Waste Handling Building.

In the Waste Handling Building, the cask enters one of two waste handling systems: either the assembly transfer or the canister transfer system. The assembly transfer system receives casks containing bare spent fuel assemblies or nondisposable canisters containing spent fuel assemblies. The assemblies are removed from either the casks or canisters in a pool environment, after which they will be transferred to, and dried in, a fuel assembly transfer cell prior to being loaded into a disposal container. The pools provide radiation shielding and cooling for the bare commercial fuel assemblies.

The canister transfer system receives Department Of Energy generated and naval spent nuclear fuel, Department Of Energy spent nuclear fuel of commercial origin, vitrified high-level waste, and special defense waste forms that have been sealed in canisters prior to shipment to the repository. The canisters are transferred from the casks directly into disposal containers by an overhead crane.

The disposal container handling system receives loaded disposal containers from both the assembly transfer system and the canister transfer system and welds on a permanent lid. After the disposal container has been loaded, sealed, tested and decontaminated, it is thereafter referred to as a waste package.

A pair of locomotives conveys the waste package carried on a transporter through the North Portal and down the North Ramp to the subsurface access main drift (tunnel) for placement into a specified emplacement drift. A remote-controlled emplacement gantry engages and lifts the waste package and transports it to a specified position within an emplacement drift.

Solid and liquid low-level radioactive wastes generated by the Monitored Geologic Repository facilities are accumulated at the point of origin, sent to the waste treatment building, and treated as appropriate. Hazardous waste and sanitary waste are collected for proper disposition. Low-level radioactive waste and hazardous waste are shipped offsite to a licensed disposal facility. Mixed waste is not expected to be produced during normal waste handling operations; however, provisions are made for temporarily staging a small quantity of this waste prior to shipping it offsite.

## SAFETY ASSESSMENT OVERVIEW

Proposed rule 10 CFR Part 963 provides the requirements for a preclosure suitability evaluation of the Yucca Mountain site. Specifically, this proposed rulemaking requires a preliminary description of potential hazards, event sequences, and their consequences.

Hazards analyses were performed to identify hazards and their potential for initiating event sequences associated with the Monitored Geologic Repository preclosure operations. Internal hazards are those hazards presented by facility operation and processes, while external hazards at the proposed site involve natural phenomena and external man-made hazards such as those posed by aircraft and nearby military/industrial facilities. The potential initiating events were documented and input into the Monitored Geologic Repository design basis event selection process. Design basis event sequences beginning with an initiating event and ending with a potential radiological release were identified and analyzed. The design basis event sequences were categorized by event sequence frequency according to Department of Energy interim guidance as follows:

- Category 1 design basis event describes "Those natural and human-induced event sequences that are expected to occur one or more times before permanent closure of the geologic repository operations area."

- Category 2 design basis event consists of "(a) Other human-induced event sequences that have at least one chance in 10,000 of occurring before permanent closure of the geologic repository, and (b) appropriate consideration of natural events (phenomena) that have been historically reported for the site and the geologic setting."

## SAFETY ASSESSMENT CONCLUSIONS

The public and worker radiological dose limits resulting from normal operations and design basis events are specified by Department of Energy interim guidance. This guidance requires compliance with applicable requirements for public and occupational dose limits and as low as is reasonably achievable requirements. The Monitored Geologic Repository dose limits are exclusive of the dose contributions from natural background radiation.

The preliminary preclosure safety assessment shows that the Monitored Geologic Repository can operate in the preclosure period within public and worker dose limits. Through a preliminary identification of potential hazards, event sequences, and their consequences at the proposed MGR site, design basis events have been identified and their doses compared to limits. Based on this assessment, important to safety structures, systems, and components relied upon to protect the public and workers have been identified. In this assessment, no reliance on operator actions is assumed in the prevention or mitigation of design basis events.

### Public Dose

The most limiting of the proposed public dose limits, for normal operations and Category 1 design basis events, is a 10 mrem per year constraint to implement as low as is reasonably achievable requirements. For comparison, the calculated annual Total Effective Dose Equivalent for all Category 1 design basis events and normal operational releases is 6E-2 mrem per year.

The Category 2 design basis event dose limits for any individual located on or beyond any point on the boundary of the site include:

- The more limiting of a Total Effective Dose Equivalent of 5 rem, or
- The sum of the Deep Dose Equivalent and the maximum Committed Dose Equivalent to any individual organ or tissue (other than the lens of the eye) of 50 rem; and
- A Lens Dose Equivalent of 15 rem; and
- A Shallow Dose Equivalent to Skin of 50 rem.

Analysis of the maximum radiological consequence Category 2 design basis event at the Monitored Geologic Repository resulted in the following doses:

- Total Effective Dose Equivalent = 2E-2 rem.
- Sum of Deep Dose Equivalent and Maximum Committed Dose Equivalent = 1E-1 rem (maximum Committed Dose Equivalent is to the Lung).
- No Lens Dose Equivalent was calculated. Krypton-83m is the only nuclide having a submersion exposure-to-dose conversion factor and the Krypton-83m source term is negligible because of its short half-life (1.8 hours) and the minimum decay time (5 years) for the spent nuclear fuel handled at the Monitored Geologic Repository.
- Shallow Dose Equivalent to Skin = 4E-2 rem.

As indicated above, the bounding dose results for Category 2 design basis events are well below the proposed dose limits for the public.

**Worker Dose**

The occupational dose limits for adults include:

1. An annual limit of either (whichever is more limiting):
   - Total Effective Dose Equivalent of 5 rem, or
   - The sum of the Deep Dose Equivalent and the Committed Dose Equivalent to any individual organ or tissue, other than the lens of the eye, of 50 rem; and
2. Annual limits to the lens of the eye, to the skin, and to the extremities of:
   - A lens dose equivalent of 15 rem, and
   - A shallow-dose equivalent of 50 rem to the skin or to any extremity.

The dose limits for workers apply to Category 1 event sequences only, which are expected to occur during the preclosure lifetime of the Monitored Geologic Repository facilities, and normal

operational exposures. A dose assessment of Category 1 event sequences was performed to estimate the worker dose from inhalation and submersion pathways at an assumed distance of 100 meters. This distance is typically used in nuclear facility dose calculations of noninvolved or colocated workers. The current worker dose assessment does not include contributions from direct radiation exposures to workers during normal operations. However, direct radiation exposures will be minimized by use of facility design controls and administrative controls. The results of the worker dose calculations are provided below:

- Total Effective Dose Equivalent = 1E-2 rem/year.
- Sum of Deep Dose Equivalent and Maximum Committed Dose Equivalent = 1E-1 rem/year.
- No lens dose equivalent was calculated.
- Shallow Dose Equivalent to Skin = 1E-1 rem/year.

As indicated above, the worker dose results for Category 1 design basis events and normal operational exposures are well below the applicable occupational dose limits. In addition, as low as is reasonably achievable requirements are satisfied for workers by incorporating facility design controls and administrative controls that limit occupational exposures.

In summary, the results of the preclosure safety evaluation indicate that the Monitored Geologic Repository is able to comply with all applicable radiation protection standards for site workers and individual members of the public.

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# ACRONYMS AND ABBREVIATIONS

## Acronyms

| | |
|---|---|
| ALARA | as low as is reasonably achievable |
| ARF | airborne release fraction |
| ATS | assembly transfer system |
| BDBE | beyond design basis event |
| BOP | balance of plant |
| BWR | boiling water reactor |
| CDE | committed dose equivalent |
| CFR | Code of Federal Regulations |
| CPB | carrier preparation building |
| CQ | conventional quality |
| CRF | cladding release fraction |
| CTS | canister transfer system |
| DBE | design basis event |
| DCF | dose conversion factor |
| DCHS | disposal container handling system |
| DDE | deep-dose equivalent |
| DEP | deposition factor |
| DF | damage fraction |
| DISP | disposable canister |
| DOE | U.S. Department of Energy |
| DPC | dual-purpose canister |
| EHA | External Events Hazards Analysis |
| FC | frequency category |
| FHA | fire hazards analysis |
| FPP | fire protection program |
| HEPA | high-efficiency particulate air |
| HLW | high-level radioactive waste |
| HVAC | heating, ventilation, and air conditioning |
| IHA | Internal Hazards Analysis |
| LDE | lens dose equivalent |
| LLW | low-level radioactive waste |
| MGR | Monitored Geologic Repository |
| MTU | metric tons of uranium |

## ACRONYMS AND ABBREVIATIONS (Continued)

| | |
|---|---|
| NRC | Nuclear Regulatory Commission |
| NTS | Nevada Test Site |
| QA | Quality Assurance |
| QL | quality level |
| PMF | probable maximum flood |
| PSS | preclosure safety strategy |
| PWR | pressurized water reactor |
| RCA | radiological controlled area |
| RF | respirable fraction |
| SFA | spent fuel assembly |
| SNF | spent nuclear fuel |
| SSC | structure, system, and component |
| TEDE | total effective dose equivalent |
| TR | topical report |
| UC | uncanistered |
| WHB | Waste Handling Building |
| WRS | waste retrieval system |
| WTB | Waste Treatment Building |
| YMP | Yucca Mountain Site Characterization Project |

**Abbreviations**

| | |
|---|---|
| ≤ | less than or equal to |
| < | less than |
| ≥ | greater than or equal to |
| 1E-6 | 0.000001 |
| % | percent |
| °C | degrees Celsius |
| °F | degrees Fahrenheit |
| Ci/yr | curies per year |
| cm | centimeter |
| E | east |
| GWd | gigawatt-day/metric tons of uranium |

## ACRONYMS AND ABBREVIATIONS (Continued)

| | |
|---|---|
| $kg/cm^2$ | kilograms per square centimeter |
| lb | pound |
| $lb/in^2$ | pounds per square inch |
| mrem | millirem |
| mrem/yr | millirem per year |
| N | north |
| rem | roentgen equivalent man |
| rem/yr | rem per year |
| s | second |
| Sv/Bq | sievert per becquerel |
| $\chi/Q$ | atmospheric dispersion factor (seconds/cubic meters) |
| $\lambda$ | event frequency |

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# 1. PURPOSE

The purpose of this report is to document the preliminary safety assessment of Monitored Geologic Repository (MGR) operations in the preclosure period. The report is based on the preclosure safety assessment work performed throughout Fiscal Year 1999 and includes updates as required due to additional repository design work performed in Fiscal Year 2000.

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# 2. OBJECTIVE AND SCOPE

## 2.1 INTRODUCTION

The objective of this report is to document the safety assessment work performed by the Preclosure Safety Analysis team. This safety assessment work includes the identification of facility hazards and their potential for initiating events, identification of MGR design basis events (DBEs), evaluation of DBE occurrence frequencies and consequences, and the identification of those structures, systems, and components (SSCs) important to safety. Important to safety, with reference to SSCs, is defined in *Revised Interim Guidance Pending Issuance of New U.S. Nuclear Regulatory Commission (NRC) Regulations (Revision 01, July 22, 1999), for Yucca Mountain, Nevada* (Dyer 1999). Important to safety SSCs are those engineered features of the geologic repository operations area whose function is: (1) to provide reasonable assurance that high-level waste can be received, handled, packaged, stored, emplaced, and retrieved without exceeding regulatory limits, or (2) to prevent or mitigate DBEs that could result in doses equal to or greater than regulatory limits.

This report also provides the MGR strategies for criticality safety, radiation protection, and fire protection, along with a description of the provisions for the control and management of low-level radioactive waste. Descriptions of the MGR site characteristics and facility design are provided to support the identification of hazards and the evaluation of DBEs. The safety assessment documented in this report can be used to support Site Recommendation sufficiency arguments.

## 2.2 REQUIREMENTS

Proposed rule 10 CFR Part 963 (64 FR 67086) provides the methods and criteria that the U.S. Department of Energy (DOE) will use for determining the suitability of the Yucca Mountain site for the location of a geologic repository. The proposed rule provides guidelines for preclosure and postclosure site suitability determination, methods, and criteria. Only the preclosure period is addressed in this report.

The proposed guidelines for the preclosure safety evaluation method (10 CFR 963.13(b)) require an assessment of the adequacy "...of the repository facilities to perform their intended functions and prevent or mitigate the effects of postulated design basis events that are deemed sufficiently credible to warrant consideration" using the criteria in 10 CFR 963.14.

The results of this preclosure safety evaluation will establish that the MGR is likely to comply with all applicable radiation protection standards.

## 2.3 QUALITY ASSURANCE

This report was prepared in accordance with the development plan for *Preclosure Safety Assessment* (CRWMS M&O 1999). The preparation of this report was evaluated in accordance with QAP-2-0, *Conduct of Activities*, and determined to be not subject to the requirements of the *Quality Assurance Requirements and Description* (DOE 2000). This determination is documented in an activity evaluation (Gwyn 1999).

This report is prepared in accordance with the applicable portions of AP-3.11Q, *Technical Reports*, as required for a non-quality-affecting report. Tracking of To Be Verified/To Be Determined information will not be performed in this non-quality-affecting report.

This document may be affected by technical product input information that requires confirmation. Any changes to the document that may occur as a result of completing the confirmation activities will be reflected in subsequent revisions. The status of the input information quality may be confirmed by review of the Document Input Reference System database.

## 2.4 REFERENCES

### 2.4.1 Documents Cited

CRWMS M&O (Civilian Radioactive Waste Management System Management and Operating Contractor) 1999. *Preclosure Safety Assessment*. Development Plan TDP-MGR-SE-000005 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19991029.0156.

DOE (U.S. Department of Energy) 2000. *Quality Assurance Requirements and Description*. DOE/RW-0333P, Rev. 10. Washington D.C.: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ACC: MOL.20000427.0422.

Dyer, J.R. 1999. "Revised Interim Guidance Pending Issuance of New U.S. Nuclear Regulatory Commission (NRC) Regulations (Revision 01, July 22, 1999), for Yucca Mountain, Nevada." Letter from J.R. Dyer (DOE/YMSCO) to D.R. Wilkins (CRWMS M&O), September 3, 1999, OL&RC:SB-1714, with enclosure, "Interim Guidance Pending Issuance of New NRC Regulations for Yucca Mountain (Revision 01)." ACC: MOL.19990910.0079.

Gwyn, D.W. 1999. "QAP-2-0 Evaluations." Interoffice Correspondence from D.W. Gwyn (CRWMS M&O) to R.A. Morgan, October 18, 1999, LV.SA.DWG.10/99-093, with attachment. ACC: MOL.19991105.0076.

### 2.4.2 Codes, Standards, Regulations, and Procedures

64 FR (Federal Register) 67086. Part 963-Yucca Mountain Site Suitability Guidelines. Readily available.

AP-3.11Q, Rev. 1, ICN 1. *Technical Reports*. Washington, D.C.: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ACC: MOL.20000714.0549

QAP-2-0, Rev. 5. *Conduct of Activities*. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19980826.0209.

# 3. SITE CHARACTERISTICS

Proposed rule 10 CFR Part 963 (64 FR 67086) provides the requirements for a preclosure suitability evaluation for the Yucca Mountain site. Specifically, 10 CFR 963.13(b)(1) requires that the preclosure safety evaluation consider a preliminary description of the site characteristics.

This section provides a description of the site characteristics necessary for understanding the MGR site environment important to the hazards and design basis events analyses presented in Section 5. A discussion of applicable natural phenomena and external man-made hazards and nearby facilities that could affect MGR operations is also included. Site characteristics applicable to repository postclosure safety are not discussed in this section.

## 3.1 SITE DESCRIPTION

This section provides an overview of the general geography and demography of the region encompassing the potential repository at the Yucca Mountain Site. The discussion includes the general physiography and topography of the region and a more detailed description and identification of the Yucca Mountain Site, facilities, and boundaries. The section also identifies the three-county area which will receive most of the socioeconomic impacts of the repository. This section discusses the population distribution and density and provides a brief socioeconomic overview of the region, including a focus on the population within 84 kilometers (52 miles) of the potential repository. The site description provided in this section is based upon *Yucca Mountain Site Description* (CRWMS M&O 1998a).

### 3.1.1 Geography

The Yucca Mountain Site is located in Nye County in Southern Nevada, approximately 160 kilometers (100 miles) northwest of Las Vegas, on land controlled by the U.S. Air Force (Nellis Air Force Range), the DOE, Nevada Test Site (NTS), and the U.S. Bureau of Land Management (CRWMS M&O 1998a, p. 1.1-1).

The Yucca Mountain Site and surrounding areas are in the southern part of the Great Basin, the northern-most subprovince of the Basin and Range Physiographic Province. The topography of the Yucca Mountain Site and surrounding region is typical of the Great Basin and the larger Basin and Range Province which are generally characterized by more or less regularly spaced, generally north-south trending mountain ranges and intervening alluvial basins that were formed by faulting. The Great Basin subprovince is an internally draining basin; i.e., precipitation that falls over the basin has no outlet to the Pacific Ocean (CRWMS M&O 1998a, p. 1.1-1).

Elevation changes and variations in topographic relief are considerable within the area of the Yucca Mountain Site. On the NTS, elevation varies from approximately 1,000 meters (3,280 feet) above sea level in Frenchman Flat and Jackass Flat to about 2,339 meters (7,675 feet) on Rainier Mesa and about 2,199 meters (7,216 feet) on Pahute Mesa. Within 50 miles south of the Yucca Mountain Site, Death Valley in California presents the lowest point in the Western Hemisphere, 86 meters (282 feet) below sea level at Badwater (CRWMS M&O 1998a, p. 1.1-1).

Yucca Mountain is an irregularly shaped volcanic upland which reaches an elevation ranging from 1,500 to 1,930 meters (4,922 to 6,332 feet) at the crest and has about 650 meters (2,132 feet) of relief. The Yucca Mountain climate is arid and the mountain historically receives less than 25 centimeters (10 inches) of rain per year (CRWMS M&O 1998a, p. 1.1-1).

There are no perennial streams in the general vicinity of Yucca Mountain (CRWMS M&O 1998a, p. 1.1-1). Streams in the vicinity of Yucca Mountain are ephemeral, fed by runoff from snowmelt and from precipitation during storms that are most common in winter, although they occur occasionally in spring and fall with localized thunderstorms during the summer. Surface water runoff in the Yucca Mountain area is through Fortymile Canyon and south through Fortymile Wash. Jackass Flats, east of Yucca Mountain and one of the three primary valleys on the NTS, is topographically open with drainage via the Fortymile Wash. The Fortymile drainage, in turn, intersects the Amargosa River in the Amargosa Desert about 32 kilometers (20 miles) southwest of the NTS. The Amargosa River ends at Death Valley. For more information on surface hydrology, see Section 3.2.2.

The Yucca Mountain Site exists in proximity to a number of natural hazards including faults/seismic activity and volcanic activity, and man-made hazards including weapons testing.

Faults have been identified and there has been historic seismic activity in the Southern Great Basin. The Southern Great Basin has also been the location of volcanic activity as recently as the Pleistocene. The NTS has been the location of nuclear tests, and is currently used to test conventional weapons and to conduct toxic waste disposal and scientific experiments. Section 3.3 provides additional discussion of the NTS and other activities in the vicinity of the potential repository (CRWMS M&O 1998a, p. 1.1-2).

The Yucca Mountain Site Characterization Project (YMP) has identified an area surrounding the potential repository known as the Preclosure Controlled Area. As the future administrative areas of the YMP have yet to be determined, the Preclosure Controlled Area is currently used as a boundary for determining infrastructure and activities that are "onsite" versus "offsite." As site characterization activities are completed, the YMP will identify the boundaries for the areas defined in Section 2 of the Revised Interim Guidance (Dyer 1999). The establishment of these regulatory-required boundaries that will replace the Preclosure Controlled Area will be coordinated with the identification and analysis of design basis events. The site boundaries used in the calculation of public radiological dose due to MGR release are discussed in Section 5.3.5.3.

Public access to the Yucca Mountain Site and the NTS is restricted and guard stations are located at all entrances to the NTS, as well as throughout the NTS. Access to the Yucca Mountain Site is through the NTS, which is accessed through four main, paved points. Other existing, unpaved roads can provide entrance or exit routes in case of emergency. The primary entrance to the NTS is through Gate 100 on the Mercury Highway, which originates at U.S. Highway 95, 105 kilometers (65 miles) northwest of Las Vegas. A second entrance, a turnoff from Highway 95 to Jackass Flats Road, is 8 kilometers (5 miles) west of Mercury. This entrance is presently barricaded. A third entrance from Highway 95 is through Gate 510 at Lathrop Wells Road, approximately 32 kilometers (20 miles) west of Mercury. A fourth entrance to NTS is via State Road 375 through Guard Station 700 in the northeast corner of the NTS. Transportation to

the Yucca Mountain Site through the NTS is primarily by Lathrop Wells Road, Jackass Flats Road, Cane Springs Road, and H-Road; and is further augmented by a network of graded gravel roads and jeep trails (CRWMS M&O 1998a, p. 1.1-3).

### 3.1.2 Demography

The demographic study area surrounding the Yucca Mountain Site includes three counties: Clark, Lincoln, and Nye, which cover approximately 95,000 square kilometers (37,000 square miles) and have an estimated population of 1,224,000 (CRWMS M&O 1998a, p. 1.2-1).

Population and related economic activity in Southern Nevada are concentrated in Clark County in the incorporated cities and in the unincorporated areas of the Las Vegas Valley. The incorporated cities include Boulder City, Henderson, Las Vegas, Mesquite, and North Las Vegas, which contain about 680,000 of Clark County's approximately 1,192,000 persons. Most of the remainder of the Clark County population resides in the unincorporated areas near Las Vegas, including East Las Vegas, Paradise, Spring Valley, and Sunrise Manor, which together total approximately 430,000 persons (CRWMS M&O 1998a, p. 1.2-1). Lincoln County has a total population of only approximately 4,000 persons, about 2,600 (64 percent) of whom live in the incorporated town of Caliente or the unincorporated towns of Alamo, Panaca, or Pioche. The overall population density of Lincoln County is only 0.15 persons per square kilometer (CRWMS M&O 1998a, p. 1.2-1).

Nye County, where the Yucca Mountain Site is located, has approximately 28,000 persons, 0.59 persons per square kilometers. Of this population, approximately 23,000 persons (84.6 percent) live in the incorporated town of Gabbs and the unincorporated towns of Amargosa, Beatty, Manhattan, Pahrump, Round Mountain, and Tonopah. The largest population concentration is in Pahrump, with approximately 18,970 persons, 69 percent of the total county (CRWMS M&O 1998a, p. 1.2-1).

The population in the vicinity of the potential repository is important in assessing the potential risk to the public health and safety. Accordingly, an area of population analysis centered on the site (Longitude 116°25'33.32" E and Latitude 36°51'11.61" N) has been established in accordance with Appendix D of NRC Regulatory Guide 1.109. The area is 84 kilometers (52 miles) in radius and is designated the Radiological Monitoring Grid. The circle at the center has a diameter of approximately 4 kilometers (2.5 miles). Each succeeding circle has a radius 8 kilometers (5 miles) greater than the previous circle. Much of the Grid is located in the southernmost portion of Nye County with smaller outer portions of the Grid in the Nevada counties of Clark, Lincoln, and Esmeralda, and in Inyo County in California (CRWMS M&O 1998a, p. 1.2-1).

The population concentrations within Nye County are important for this safety assessment. In particular, there are no permanent residents within 20 kilometers (12.5 miles) of the center of the Grid. The only residents in this area are transient populations at Mercury on temporary duty at the NTS who are under the control of the NTS and are subject to being moved as needed. The closest permanent population concentration is in Amargosa Valley, a primarily agricultural-based community on the south edge of the NTS. The population densities in this region are between 2 and 4 persons per square kilometers in the inhabited sectors. Several of the

sectors have zero populations. Similarly, in the Beatty area, population densities in the most populated sectors are approximately 6 and 11 persons per square kilometers (CRWMS M&O 1998a, p. 1.2-1).

Pahrump is at the edge of the Radiological Monitoring Grid and has a population of approximately 19,000 (it is partially within the Radiological Monitoring Grid and is the only town within the 84 kilometers Radiological Monitoring Grid to have a population greater than 2,500) (CRWMS M&O 1998a, p.1.2-2). Rapid growth in Pahrump has been the result of increased immigration of retirees and increases in population of persons who live in Pahrump and commute to Las Vegas for employment.

Within the 84-kilometer Grid, population concentrations are primarily a result of agricultural, mining, tourism, and service activities. Agricultural development is concentrated in Amargosa Valley and Pahrump. Mining operations, tourism, general services, and employment on the NTS and the Nellis Air Force Range help support these two places and other towns located within the Radiological Monitoring Grid, including Mercury, Beatty, Johnnie, Furnace Creek Ranch, and Death Valley Junction.

Current land use patterns and economic drivers will drive future population changes within the 84-kilometers area. Little of the area surrounding Yucca Mountain is privately owned, and there is very little built-up or urban land. The effects of Native American uses of the land for cultural purposes are discussed in the *Draft Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada* (DOE 1999). Close to the Yucca Mountain Site it is likely that a large percentage of the land will remain federally owned and controlled. In addition, the NTS is withdrawn from public use entirely. Considering the substantial disturbance of the environment on the NTS, it is unlikely that it will be available for unrestricted public use or habitation in the near future. Consequently, it is assumed that there will be a lack of economic impetus and resulting infrastructure on the limited private land near the site sufficient to support large populations.

## 3.2 ENVIRONMENTAL DESCRIPTION

This section provides summary descriptions of the meteorology, hydrology, and geology associated with the MGR site and is based upon *Yucca Mountain Site Description* (CRWMS M&O 1998b, CRWMS M&O 1998c and CRWMS M&O 1998d). This section supports the selection and evaluation of natural phenomena discussed in Section 5.

### 3.2.1 Meteorology

Present-day climate in southern Nevada is semi-arid, with hot summers and mild winters. The regional weather is influenced by complex topography and weather system circulation patterns. Local and regional monitoring stations provide weather data for the vicinity of Yucca Mountain. The annual average precipitation in the Yucca Mountain area is approximately 100-250 millimeters (4-10 inches) per year, depending on topographic elevation and exposure. About 30 years of monitoring at Amargosa Farms southwest of Yucca Mountain indicates an average of about 100 millimeters (4 inches) per year; at a station 10 kilometers (6.2 miles) east of

Yucca Mountain average precipitation is 133 millimeters (5.2 inches) per year; geostatistical studies suggest an average of 250 millimeters (10 inches) per year at higher elevations along the north of Yucca Mountain (CRWMS M&O 1998b, Section 4.1.3.2 and CRWMS M&O 1998c, Section 5.3.4.1.2.2). The estimated annual potential evapotranspiration (maximum surface moisture loss to the atmosphere) is 1,680 millimeters (66 inches) per year (Houghton et al. 1975). Snowfall is infrequent, light, and short-lived below about 1,070 meters (3,510 feet) above mean sea level. The estimated maximum daily rainfall is bounded by a value of 125 millimeters (5 inches). Lightning can accompany summer thunderstorms, but very few tornadoes have been reported in the region. Other potentially severe weather events, such as blizzards and hurricanes, do not occur in the region (CRWMS M&O 1997, p. xi).

The annual average temperature in the Yucca Mountain area ranges from about 15 to 18°C (59 to 65°F), depending on elevation. Summer temperature can exceed 40°C (104°F), and winter temperatures occasionally fall below 0°C (32°F) (CRWMS M&O 1997, p. xi). Low relative humidity throughout most of the year is indicative of the semi-arid climate. The annual average relative humidity ranges from an early morning value of 38 percent to a late afternoon value of 21 percent. Afternoon summer humidity values less than 5 to 10 percent are not uncommon (CRWMS M&O 1997, p. 4-17). Regional weather systems and the mountain and valley topography cause a regular wind pattern of well-mixed airflow toward the north during the daytime and stable (low mixing) airflow toward the south into Amargosa Valley at night.

The aridity and warm temperatures result from a combination of large-scale atmospheric circulation patterns and the large mountain ranges, such as the Sierra Nevada, on the moisture pathway from the primary source, the Pacific Ocean. The Yucca Mountain area is affected by typical mid-latitude global circulation patterns, with weather systems moving from west to east. Storms moving into the area from the southwest during winter tend to have the greatest potential for high precipitation as rain or snow. Significant late summer, southwest monsoon precipitation events occur with moist airflow from the south originating either in the Gulf of Mexico or the Pacific Ocean. Naturally recurring short-term changes in typical circulation patterns alter storm paths and precipitation patterns. One example is the El Niño pattern, which tends to increase winter precipitation in Southern Nevada by approximately 50 percent.

The meteorological description of this section focuses on the data obtained at monitoring Site 1 (NTS-60). Site 1 is located at an elevation of 1,143 meters (3,750 feet) mean sea level in western Midway Valley, an area bounded on the west by Yucca Mountain; on the east by Fran Ridge; and on the south by a saddle between Fran Ridge and Yucca Mountain. This site has a multi-level meteorological tower that makes continuous wind and temperature measurements at 60 m and 10 m heights typically required by the NRC for dispersion analyses. The atmospheric profile information helps define both the vertical temperature and wind layer structures. Of the network sites, this site is the most representative of the Exploratory Studies Facility and proposed surface facilities area. Thus, data from this site are likely to be the focal point of atmospheric dispersion modeling activities. A detailed description of the meteorological monitoring network is provided in Section 4.1.4 of CRWMS M&O 1998b.

### 3.2.2 Hydrology

This section discusses surface water flow at Yucca Mountain as applicable to repository design. Surface facility design must consider surface water flow and flood potential to ensure that storm events do not affect safe operation of the repository during the preclosure period. Yucca Mountain hydrology as applicable to the postclosure performance of the repository is not discussed in this report.

Presently, there are no permanent streams or other surface water bodies near Yucca Mountain. Springs sustain year round flow for short distances near Beatty in Oasis Valley, about 25 kilometers (15 miles) west of the site, and also about 100 kilometers (62 miles) to the southwest in Death Valley. The steep western slope of the Yucca Mountain site drains to Solitario Canyon, which joins other channels that drain southward to the Amargosa River channel. Fortymile Wash, which drains southward from Pahute Mesa between Timber Mountain and Shoshone Mountain to Jackass Flats, collects runoff from several washes that drain the gentler, eastern slope of the potential repository site. Fortymile Wash also continues southward into the Amargosa Desert where it joins the Amargosa River (CRWMS M&O 1998c, Section 5.1).

Although flow in these channels is rare, the area is subject to flash flooding from intense summer thunderstorms or occasional sustained winter precipitation. In July 1984, for example, a peak discharge of 40.5 cubic meters (1,430 cubic feet) per second was measured in Fortymile Wash near Amargosa Valley just south of Yucca Mountain (Pabst et al. 1993, p. 23). A peak discharge of 34.0 cubic meters (1,200 cubic feet) per second was measured at the same site in March 1995 (CRWMS M&O 1998c, Table 5.1-2, Map Designation number 38).

Bullard (1986) developed probable maximum flood discharge hydrographs (water flow measurements) for the small basins of interest at Yucca Mountain. Blanton (1992) developed water-surface profiles for probable maximum flood discharges in the operations area. The topography and predicted extent of flooding have been used to design the proposed operations area to limit flooding effects.

### 3.2.3 Geology

This section describes the geology of the Yucca Mountain site and region, with an emphasis on those site aspects that are most relevant to preclosure safety. This discussion of MGR site geology is based upon the *Yucca Mountain Site Description* (CRWMS M&O 1998d).

#### 3.2.3.1 Regional Geology

Yucca Mountain lies in the Basin and Range province of the western United States within the region known as the Great Basin. The Great Basin encompasses nearly all of Nevada and parts of Utah, Idaho, Oregon, and California. The mountain ranges of the Great Basin are mostly north-south aligned, tilted, fault-bounded blocks that may extend more than 80 kilometers (50 miles) in length and are generally 8–24 kilometers (5–15 miles) wide. Relief between valley floors and mountain ridges is typically 300–1,500 meters (984–4,921 feet), and valleys occupy approximately 50–60 percent of the total land area. The valleys are filled with thick deposits of alluvium derived from erosion of the adjacent ranges. The ranges are separated north and south

by roughly 25–30 kilometers (15–19 miles), but many ranges arc toward each other and merge together. This pattern is the result of generally east-west-directed crustal extension that began in the Tertiary period and continues at present (Hamilton 1988).

The highest rate of modern tectonic activity in the southwestern Great Basin (i.e., active faulting and volcanism) is found in the Inyo-Mono domain to the west and south of Yucca Mountain, which includes the Furnace Creek-Death Valley fault zone, the Sierra Nevada front, and the area north of the Garlock fault in the Mojave Desert. This domain includes basins and ranges with great structural relief, including Death Valley Basin and the Panamint Range. The modern faulting and volcanic activity is caused by the continuation of the same tectonic extension that resulted in the formation of the entire Basin and Range. The crust on the western edge of the Great Basin (the Sierra Nevada) is gradually moving to the west relative to the eastern edge of the basin (the Wasatch Front in Utah).

#### 3.2.3.2 Regional Stratigraphy

The lithology (rock type) and stratigraphy (the sequence in which rocks were deposited) of the regional geologic setting provide the basis for understanding the geologic history and evolution of the area, which is fundamental to analyzing geologic hazards such as those associated with earthquakes and volcanoes. The stratigraphy and lithology provide the framework for understanding more local aspects of the Yucca Mountain site, including its structural geology and tectonics, geoengineering properties, mineral resource potential, hydrology, and geochemistry. The following summary briefly describes rock units important to the regional geologic setting of Yucca Mountain:

- **Precambrian Era**. Precambrian rocks (greater than approximately 570 million years in age) include two major rock types: an older, metamorphosed basement assemblage and a younger, metasedimentary assemblage. Both groups tend to retard the flow of groundwater except where extensive faulting or fracturing is present.

- **Paleozoic Era**. Paleozoic rocks (rocks ranging from approximately 570 million to 240 million years in age) in the Yucca Mountain region include lower (older) carbonate strata (limestone and dolomite); a middle, fine-grained shale, siltstone and sandstone unit; and an upper (younger) carbonate unit (limestone).

- **Mesozoic Era**. Strata from this era (rocks ranging from approximately 240 million to 66 million years in age) are generally absent near Yucca Mountain. Regionally, these strata are dominantly continental and shallow marine sediments (sandstones and siltstones) with minor Cretaceous granitic plutonic rocks. Structurally, the Mesozoic was a period of active tectonic activity characterized by regional compression.

- **Cenozoic Era**. Cenozoic rocks (66 million years in age to the present) near Yucca Mountain fall into three groups. The last two are of major importance to the Yucca Mountain site: Pre-middle Miocene (greater than about 16 million years old) sedimentary rocks, Mid-to-late Miocene (15 million to 7.5 million years old) volcanic rocks that constitute the southwestern Nevada volcanic field, including Yucca Mountain, and Plio-Pleistocene (3.7 million years old to modern) basalts and basin sediments.

The volcanism that culminated in the formation of the southwestern Nevada volcanic field is the most significant depositional event of the Cenozoic era near Yucca Mountain. This event formed six major calderas (volcanic centers) between 15 million and 7.5 million years ago (Sawyer et al. 1994). This event also created the rocks of Yucca Mountain, and brought to a close the major regional tectonic activity that created the present Yucca Mountain geologic setting.

The most recent deposits in the region consist of alluvial sediments, formed during highland erosion, and infrequently erupted basaltic volcanic rocks. The basaltic eruptions represent a continuation of the activity during the mid- to late-Miocene epoch (Crowe et al. 1995). Following an episode 3.7 million years ago, a subsequent basaltic eruption occurred between 1.7 million and 0.7 million years ago consisting of four cinder cones (Little Cone, Red Cone, Black Cone, and Makani Cone) aligned north-northeast along the Crater Flat axis. The final episode of basaltic volcanism created the Lathrop Wells Cone, which includes fissure eruptions, spatter and scoria cones, and basaltic lava flows. Satellite spatter cones at the east base of the main cone have a northwest alignment. The Lathrop Wells Cone complex is approximately 75,000 years old (CRWMS M&O 1998d).

#### 3.2.3.3 Regional Tectonic Models

Several alternative models have been proposed to explain the known structural, volcanic, and seismic characteristics of the site (Whitney 1996, Chapter 8). The models provide a means for integrating and understanding data such as the history of volcanism, deposition of sediment, and fault movement in the site vicinity. In assessing volcanic and earthquake hazards, scientists considered a range of models in evaluating the likelihood of future events.

In resolving potential MGR seismic licensing issues, the DOE and NRC agreed upon the use of the Topical Report (TR) approach. The DOE then developed a plan to address seismic issues in three separate TRs. The first TR (TR-1) addresses the proposed DOE methodology to assess seismic hazards. TR-2 addresses the proposed DOE seismic design methodology and TR-3 addresses vibratory ground motion and fault displacement inputs that will be used in repository design and performance assessments. TR-1, *Methodology to Assess Fault Displacement and Vibratory Ground Motion Hazards at Yucca Mountain* (YMP 1997a) and TR-2, *Preclosure Seismic Design Methodology for a Geologic Repository at Yucca Mountain (August 1997)* (YMP 1997b) have been issued. TR-3 is currently being developed.

## 3.3 NEARBY FACILITIES

This section identifies present and projected industrial, transportation, and military facilities and operations that occur in the vicinity of the Yucca Mountain Site that may have a potential effect on the MGR. This information is based on the *Yucca Mountain Site Description* (CRWMS M&O 1998e) and supports the identification of external hazards as discussed in Section 5.

The identification of nearby facilities is based upon NRC guidance established for nuclear power plants, specifically NUREG 0800, Section 2.2.1 – 2.2.2 (NRC 1981) and NRC Regulatory Guide 1.70, which direct the identification of all facilities and activities within 8 kilometers (5 miles) of the plant. Both documents also direct that facilities and activities at greater distances should be analyzed if they have the potential for affecting safety-related features.

The term "plant" is interpreted for this section to represent the surface facilities at the Yucca Mountain Site that will be active if the potential repository at Yucca Mountain is authorized for waste reception. The area within an 8-kilometer (5-mile) radius of the potential repository includes parts of the Nellis Air Force Range, Area 25 of the NTS, and public lands managed by the U.S. Bureau of Land Management.

### 3.3.1 Nearby Facilities and Activities within 8 Kilometers (5 Miles)

#### 3.3.1.1 Airspace

The area within 8 kilometers (5 miles) of the potential repository is located beneath or adjacent to restricted airspace areas, control over which has been delegated to the U.S. Air Force and DOE by the Federal Aviation Administration. This restricted airspace was established because of the classified and/or hazardous nature of the activities conducted within these airspaces or in the areas beneath these airspaces. Restricted area R-4807 extends north of the potential repository site over Nellis Air Force Range withdrawn lands and overlies ground support facilities for military air-to-ground weapons training including convoys, simulated airfields, and electronic combat threat emitters. Electronic Combat South is the closest subrange of R-4807 to the potential repository. Electronic Combat South is primarily used as an entry/exit corridor for the R-4807 subranges and contains manned electronic threat emitters (USAF 1994). No ordnance is used in this area. The potential repository underlies the western portion of R-4808 (R-4808W), a DOE-restricted area associated with NTS activities. By agreement with the DOE, military aircraft may use flight routes within R-4808 for entering/exiting R-4807. Nellis aircraft using R-4808 to enter and exit the Nellis Air Force Range are randomly dispersed. There are currently no set entry and exit routes. Some aircraft do fly within three miles of the proposed repository surface facilities, however, flight procedures generally keep aircraft east of the proposed repository at 4.9 kilometers (16,000 feet) above mean sea level while transiting this area (USAF 1994). DOE Nevada Operations Office and Nellis Air Force Base have a classified Memorandum of Understanding regarding use of R-4808 for entering and exiting the Nellis Air Force Range Complex.

Numerous military training routes that traverse the state are used by U.S. Air Force and U.S. Navy aircraft for low level, high speed flight training. Most of these military training routes are located outside of the Nellis Air Force Range and may or may not be used in conjunction with other training taking place within the Nellis Air Force Range. One of these routes, VR-222, lies south and west of the potential repository and outside the Nellis Air Force Range. This military training route has a width of five nautical miles on either side of its centerline. The centerline is approximately eight nautical miles from the proposed repository; therefore the military training route is three miles, at the closest, from the proposed repository (DOD 1997)

#### 3.3.1.2 Nevada Test Site Area 25

Area 25, the largest area on the NTS, occupies 223 square miles and is divided into four land use zones, the Yucca Mountain Site Characterization Zone; the Research, Test and Experiment Zone; the Reserved Zone; and the Solar Enterprise Zone. The Yucca Mountain Site Characterization Zone has been reserved by the DOE for Yucca Mountain Site characterization activities. The Research, Test, and Experiment Zone is used by the U.S. Army Ballistic Research Laboratory for depleted uranium testing and other activities. Reserved Zones at NTS are used to provide areas

and facilities that allow flexible support for diverse short-term testing and experimentation. The Reserved Zone in Area 25 is used for military land navigation and training exercises. Research sites within the Area 25 Reserved Zone include the Treatability Test Facility and Bare Reactor Experiment Nevada Tower. The Treatability Test Facility was established for bench-scale testing of physical processes for separating plutonium and uranium from contaminated soils. The 465-meter (1526-foot) Bare Reactor Experiment Nevada Tower has been used by a number of organizations to conduct sonic-boom research, meteorological studies, and free-fall/gravity drop tests. The Solar Enterprise Zone is designated for the development of a solar energy power-generation facility and associated light industrial equipment and commercial manufacturing capability. In the 1980s, Area 25 was used for missile siting studies and canister ejection certification tests (DOE 1996).

#### 3.3.1.3 Bureau of Land Management Land

There are no known formal industrial/commercial land uses or infrastructure on Bureau of Land Management land (exclusive of dirt roads) within 8 kilometers (5 miles) of the potential repository (CRWMS M&O 1998e, Section 2.2).

### 3.3.2 Nearby Facilities And Activities Greater Than 8 Kilometers (5 Miles)

Outside of the 8-kilometer (5-mile) radius from the potential repository, there are military, transportation, and industrial/commercial facilities and activities on the Nellis Air Force Range, the NTS, and Bureau of Land Management land which could potentially affect daily operations and performance or be used as design basis events for the potential repository.

#### 3.3.2.1 Nellis Air Force Range

The Nellis Air Force Range "north range" extends north of Electronic Combat South and is used extensively for weapons training and testing. Large amounts of live and inert ordnance are used on the northern portions of this range that are approved for ordnance use. There are substantial numbers of aircraft flights within the north range where training missions, exercises, and weapons testing take place daily. Although Yucca Mountain is not directly beneath any military routes or in close proximity to live ordnance use on the Nellis Air Force Range, the existence of a high density of flights in the Nellis Air Force Range and the possibility of an aircraft accident could present a potential threat to daily operations and performance of the potential repository.

#### 3.3.2.2 Nevada Test Site

The NTS was the primary location of United States continental nuclear weapons testing from 1945 to 1992, and during that period more than 900 above- and below-ground nuclear weapons tests were performed. Nuclear weapons tests were banned by treaty in 1992; however, the DOE is still directed by the Executive Office to maintain a state of preparedness to test nuclear weapons in the future. Potential areas for future tests include Pahute Mesa and Yucca Flat (DOE 1996), both of which lie within approximately 60 kilometers (37 miles) of the potential repository. Nuclear weapons tests may affect seismicity in the region and administrative policies enforced during such weapons tests could affect the daily operations of the potential repository (DOE 1988). In addition to maintaining preparedness for possible nuclear weapons testing, NTS operations include destroying damaged nuclear weapons and conducting dynamic experiments under the Stockpile Stewardship Program, including impact, passive, and chemical tests

(DOE 1996). Another activity includes rocket launches by Sandia National Laboratory from Wahmonie in Area 26 to the Tonopah Test Range, approximately 113 kilometers (70 miles) to the northwest (Rogers 1997). While these activities take place outside of the 8-kilometer (5-mile) boundary, they could potentially pose a health and/or safety hazard and affect daily operations or performance of the potential repository. Other current and potential uses of the NTS are found in the NTS Environmental Impact Statement (DOE 1996).

A part of the NTS is under development for private use, and in 1997 a 10-year use permit was signed by NTS Development Corporation and the DOE, enabling Kistler Aerospace Corporation to begin development of launch operations for a fully reusable orbital launch vehicle. Kistler Aerospace is expected to conduct testing in Area 18 of the NTS. Kistler Aerospace activities are considered here because launch and re-entry activities could potentially pose a health and/or safety hazard to the potential repository if operations continue past 2010 (CRWMS M&O 1998e, Section 2.3).

As MGR development continues, other nearby facilities and activities may pose special public health and safety, or radiological health and safety hazards, to the development, operation, or closure of the potential repository. Potential NTS activities are the development of new transportation corridors or the promotion of mineral resource exploration and development in Area 25. Should such NTS activities be initiated, additional safety analyses will need to address such issues.

#### 3.3.2.3 Other Areas

This report also considered commercial, industrial, and transportation operations more than 8 kilometers (5 miles) from the Yucca Mountain Site to see if they would pose a health or safety hazard or would affect daily activities at the potential repository. Three such activities were identified on land outside the NTS and Nellis Air Force Range. The first is the Razorback Grazing Allotment, which borders the southwestern corner of the Nellis Air Force Range and is located just outside the 8-kilometer (5-mile) buffer of the potential repository. The grazing allotment, covering 72,880 acres of public land, is scheduled to expire in 2005 (BLM 1995) and does not appear to pose a threat to preclosure activities at the potential repository.

The second activity is gold mining near Beatty and associated water usage from volcanic boreholes that support these mining activities. Currently, VH-2 is the only known volcanic borehole in the area that provides water for mining activities. The area also contains numerous other boreholes that have been drilled as part of the site characterization activities. Maps showing these boreholes are available in the *Yucca Mountain Site Characterization Project Site Atlas* (DOE 1997). These activities do not present a threat to the potential repository-related activities at Yucca Mountain.

The third activity is commercial aircraft activities in proximity to the NTS and the Yucca Mountain Site. The *MGR Aircraft Crash Frequency Analysis* (CRWMS M&O 1999) determined that there are no commercial, private, or DOE aircraft activities that present a credible threat to activities related to the potential repository at Yucca Mountain.

There are no other known commercial, industrial, or transportation operations outside 8 kilometers (5 miles) from Yucca Mountain that could pose a health or safety hazard or could affect daily operations at the potential repository.

### 3.3.3 Nearby Transportation Routes

Transportation routes of potential concern to health, safety, and normal operations at the potential repository are potential railroads or heavy haul truck routes that may be built or upgraded to transport high-level radioactive waste (HLW) to the potential repository, highways in the vicinity of the Yucca Mountain Site, and commercial and military flight zones. There are no streams or rivers in the vicinity of the Yucca Mountain Site that are capable of supporting water-based forms of transportation.

#### 3.3.3.1 Railroads and Heavy Haul Truck Routes

If the potential repository is built and operated, at least some HLW is expected to be transported to Nevada by rail. When the HLW reaches Nevada, it will be transported from the national rail lines to the potential repository by either rail or heavy haul trucks. Because there are currently no rail lines to the potential repository at Yucca Mountain, a line will have to be constructed if it is decided to deliver by rail directly to the repository. Similarly, if the heavy haul implementing alternative is selected, heavy haul routes will have to be either constructed or existing routes will have to be upgraded. There are currently five potential heavy-haul truck routes and five potential rail corridors in Nevada being considered for transportation of HLW to Yucca Mountain. These potential routes have been planned with consideration of necessary rights-of-way, land withdrawals, use restrictions and land-use conflicts, and are described in the *Draft Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada* (DOE 1999). There is only a small portion of each of these routes sufficiently close enough to the potential repository to potentially affect daily operations or performance. Scenarios for onsite HLW transportation hazards are being evaluated.

#### 3.3.3.2 Flight Corridors and Highways

Military and commercial air transportation corridors and activities were described in Sections 3.3.1 and 3.3.2. Aside from commercial aircraft traffic, U.S. Highway 95 is the only primary transportation route near the Yucca Mountain Site. U.S. Highway 95 lies in a northwest/southeast orientation and passes approximately 19 kilometers (12 miles) to the southwest of the potential repository at Yucca Mountain. A traffic event on U.S. Highway 95 substantial enough to pose a direct hazard to the plant, and that would be considered a design basis event, is not considered credible and need not be evaluated. However, U.S. Highway 95 is the primary land-based route to the potential repository at Yucca Mountain and is heavily relied upon for the transportation of workers and materials to the NTS and the Yucca Mountain Site. A traffic event on U.S. Highway 95, particularly between Las Vegas and the entrances to the NTS, could potentially disrupt the delivery of HLW, materials, and employees to the potential repository, and hence affect daily operations or performance, although such an event would likely be of short duration.

## 3.4 NATURAL PHENOMENA AND EXTERNAL MAN-MADE HAZARDS

This section provides a listing of natural phenomena and external man-made hazards at the Yucca Mountain site and region that have been identified as potential, credible initiators of radiological accidents during the preclosure operating period of the MGR.

The site and region have been examined for natural phenomena and man-made hazards that are potential initiators of event sequences that could result in the release of radioactivity. The *MGR External Events Hazards Analysis* (CRWMS M&O 2000a) provides a comprehensive and structured identification and screening of such natural phenomena and man-made hazards to determine those that must be addressed in the preclosure safety analysis of the MGR. The rationale and details of the screening analyses are presented in the analysis.

Initially, the external hazards analysis applied a generic list of 53 natural and man-made hazards to the Yucca Mountain site and region. Four levels of screening were applied that eliminated 33 from the initial list, leaving 20 external hazards as candidates for inclusion in the preclosure safety analysis. For example, a detailed analysis screened out aircraft crashes as a credible initiating event (CRWMS M&O 1999).

Upon examination of the list of 20 candidate hazards, eight were eliminated from the list of hazards because they are covered by other analyses that support the design bases, or their effects are included within another hazard category. In particular, inadvertent, and intentional intrusions will be addressed in the MGR safeguards and security analyses, and range fire will be addressed in the fire hazards analyses. Hazards that were combined with other hazards are: rainstorm (effects are covered under flooding), debris, and landslide hazards; sandstorm (effects are covered under extreme wind/tornado wind); subsidence (effects are covered under seismic activity-surface/subsurface fault displacement); and dissolution and static fracturing (effects are covered by the analyses of rockfall/keyblock hazard). Rockfall/keyblock hazards are addressed in the *Monitored Geologic Repository Internal Hazards Analysis* (CRWMS M&O 2000b).

As a result of the processes of screening and combining hazards, the list of 53 potential hazards was reduced to 12 categories of natural phenomena and man-made hazards present at the Yucca Mountain site that are addressed in the preclosure safety analysis:

1. Debris Avalanching
2. Extreme Wind (including sandstorms)
3. Flooding (including rainstorm and river diversion)
4. Industrial-Activity-Induced Accident
5. Landslide
6. Lightning
7. Loss of Offsite/Onsite Power
8. Military-Activity-Induced Accident
9. Seismic Activity, Earthquake
10. Seismic Activity, Surface Fault Displacement
11. Seismic Activity, Subsurface Fault Displacement (including subsidence)
12. Tornado.

These hazards are discussed in Section 5 as part of the preclosure safety analysis.

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# 4. FACILITY DESCRIPTION

Proposed rule 10 CFR Part 963 (64 FR 67086) provides the requirements for a preclosure suitability evaluation for the Yucca Mountain site. Specifically, 10 CFR 963.13 requires that the preclosure safety evaluation consider a preliminary description of: (a) the surface and underground operating facilities (10 CFR 963.13(b)(1)); the design bases for the operating facilities and of any associated limits on operations (10 CFR 963.13(b)(2)) and; (c) the structures, systems, components, equipment, and operator actions intended to mitigate or prevent accidents (10 CFR 963.13(b)(4)).

This section discusses the above criterion and provides a description of the facility SSCs and processes along with preliminary design bases. The identification of important to safety SSCs is discussed, as well as the measures in place to ensure that important to safety SSCs perform associated preventive and mitigative functions as designed. The scope of this section focuses on those facility features necessary to support the hazards and DBE analyses. Utility and auxiliary systems not required to prevent or mitigate a design basis event (DBE) that could exceed radiological dose limits are discussed briefly to describe the facility as it pertains to the hazards and DBE analyses. Operating limits that are credited to prevent or mitigate a DBE, or to implement the preclosure safety strategy (e.g., limit canister lift heights to below design basis), are documented in the applicable System Description Documents.

## 4.1 FACILITY OVERVIEW

The primary mission of the MGR during the preclosure period is to receive spent nuclear fuel (SNF) and HLW shipments, prepare, and package the wastes for underground emplacement. The packaged waste is then transported to the subsurface facility and placed in underground emplacement drifts.

This section describes the major processes required to package and emplace the waste and provides a summary description of surface waste handling activities during the emplacement phase. The North Portal to the underground repository will be the primary location for the MGR surface facilities.

Commercial SNF and vitrified HLW are transported to the repository in certified casks by cask transporters. The waste is transported by rail or truck carriers from the point of origin to the North Portal security station, where personnel verify the shipping manifests and inspect the cask and carrier. After the cask and carrier enter the repository, they are stationed in staging areas designated for either truck carriers or rail carriers. When the waste in a cask is scheduled for processing, an onsite prime mover transports the cask and carrier to the Carrier Preparation Building (CPB).

Inside the CPB, workers retract or remove personnel barriers; survey the cask surface for radiation; decontaminate cask surfaces, if necessary; measure the cask surface temperature; retract or remove impact limiters, and remove the cask tie-downs, if necessary. After personnel prepare the shipments for processing, the shipments are taken to a staging area. Shipments are moved to the Waste Handling Building (WHB) carrier bay according to operations scheduling requirements. In the WHB carrier bay, the cask is removed from the carrier and placed on a cask

transfer cart. The carrier remains in the WHB carrier bay until the cask is emptied and reloaded onto the carrier for shipment back to a waste generator. The cask is transferred to one of two waste handling systems for unloading: the assembly transfer or the canister transfer system

The assembly transfer system receives casks containing individual fuel assemblies that have either been loaded into the cask directly or are contained in a nondisposable canister that must be removed from the cask and opened before the assemblies can be removed. Some nondisposable canisters are welded closed and must be cut open. The assemblies are removed from the casks or canisters in a pool environment, after which they will be transferred to, and dried in, a fuel assembly handling cell prior to being loaded into a disposal container. Fuel storage pools are provided primarily for biological shielding.

The canister transfer system receives SNF, vitrified defense HLW, and special defense waste forms, including immobilized plutonium, in canisters. The canisters are transferred from the casks directly into disposal containers by overhead crane.

The disposal container handling system receives loaded containers from both the assembly transfer system and the canister transfer system. The inner and outer lids are welded and inspected to ensure that they meet specifications for disposal. After the disposal container has been loaded, sealed, and tested, it is thereafter referred to as a waste package. The waste package is then placed in a horizontal orientation and loaded into a waste package transporter for transportation to the subsurface facility.

A pair of locomotives convey the waste package transporter through the North Portal and down the North Ramp to the subsurface access main. One of the locomotives is then disengaged and the transporter is moved by remote control to a specified emplacement drift entrance. The drift doors are opened and the transporter is moved into position for offloading the waste package. The waste package is withdrawn from the transporter and a remotely controlled emplacement gantry engages and lifts the waste package and transports it to a specified location within the drift. The emplacement gantry is returned to the drift entry and the waste package transporter and the locomotives then return to the surface to repeat the emplacement cycle.

Empty casks are loaded onto carriers in the carrier bay and returned to the CPB, where impact limiters and personnel barriers are reinstalled. The cask/carrier is then sent to a staging area or to the security station for offsite shipment. Empty nondisposable canisters are placed into a protective overpack and prepared for shipment to an offsite facility for disposal or recycling. Arriving empty disposal containers are delivered to the security station and from there to a staging area or empty disposal container preparation area in the WHB.

Low-level radioactive waste (LLW) and hazardous waste is generated in the surface waste handling facilities and operating areas. Solid and liquid LLW is accumulated at the point of origin, and then sent to the waste treatment facility, where they are treated as appropriate and packaged in drums. Hazardous waste is collected and packaged in drums and sanitary waste is collected for proper disposition. LLW and hazardous waste is shipped offsite to a licensed disposal facility. Mixed waste exhibits the characteristics of LLW and hazardous waste. Mixed waste is not normally produced during normal waste handling operations, however, provisions are made for temporarily staging a small quantity of this waste prior to shipping it offsite.

## 4.2 FACILITY STRUCTURES

Receipt, handling, packaging, and emplacement of SNF and HLW is performed at the MGR North Portal. The major structures involved in these processes are the CPB, the WHB, and the Waste Treatment Building (WTB). Support facilities at the North Portal include an administration building, fire station, medical center, central warehouse, central shops and motor pool, mockup building and a utility building. These support facilities are not directly involved in the handling of SNF or HLW. Facilities located at the MGR South Portal support repository construction and are not involved in the handling of SNF or HLW. The following descriptions are based on *Engineering Files for Site Recommendation* (CRWMS M&O 2000a).

### 4.2.1 Carrier Preparation Building

The CPB, to be located at the North Portal pad, will support preparation of the waste transportation casks before they enter the WHB. The building will be an on-grade, one-story, high-bay, steel-framed structure, enclosed with an insulated steel roof and wall panels. The interior framing will be of light-gauge steel and wall panels that are easily decontaminated. The foundations will consist of reinforced concrete spread footings, to support the building's columns, and continuous reinforced concrete mat foundations, to support the railroad tracks. The building will be approximately 60 meters (190 feet) long, 37 meters (121 feet) wide, and 14 meters (46 feet) high. The operations area, divided into two identical carrier-staging bays, will accommodate four parallel rail tracks/roadways for passage of both rail and truck carriers. Each staging bay will have two truck/rail lines separated by a dual-function work platform and equipment lay-down area, a bridge crane, and a bridge-mounted manipulator. The transportation carriers will enter and exit the building through one of eight remotely operated roll-up doors (CRWMS M&O 2000a, Attachment II, Section 1.3.1).

### 4.2.2 Waste Handling Building

The WHB will provide the structures, controlled areas, and accesses required to house and operate the waste handling systems, protect operating personnel, and maintain radiological confinement. The WHB will be divided into primary, secondary, and tertiary confinement areas. Integral to the facility structure will be the essential waste handling systems, including the carrier/cask handling system, assembly transfer system, canister transfer system, disposal container handling system, and waste package remediation system.

The WHB will be located close to the North Portal, within the controlled area. The structure will establish the operating and equipment areas; the boundaries required for safe handling of shipping casks, waste forms, facility waste, and disposal containers; and facility office and support operations.

The building will be a multi-level, concrete and steel structure made of noncombustible materials. The exterior walls will be mainly concrete; the outer walls of areas that do not require radiation protection will be constructed of metal siding panels with insulation. The roof will consist of a concrete slab supported by steel beams and concrete walls. Exterior doors will be made of insulated steel. The building will be approximately 180 meters (600 feet) wide by 210 meters (700 feet) long.

The building's foundation will be a reinforced concrete material (CRWMS M&O 2000a, Attachment II, Section 1.1.6). Before constructing the foundation, the native soil will be replaced with engineered compacted soil to minimize settlement. The building will be designed to withstand winds of up to 302 kilometers/hour (189 miles/hour), a pressure drop of 0.1 kg/cm$^2$ (1.5 lb/in$^2$), tornado-generated missiles, and the design basis earthquake.

The design of the WHB will include features to limit worker radiation exposure to levels that are As Low As is Reasonably Achievable (ALARA) (CRWMS M&O 2000a, Attachment II, Section 1.1.6). Radiological areas will have 1.5-meters (5-feet) thick concrete floors that can support loads of up to 126 metric tons (140 tons) to accommodate the heavy equipment that will handle casks and waste packages (CRWMS M&O 2000a, Attachment 1.1.6). The walls will also be concrete, measuring 1.5 meters (5 feet) thick at the base and 1 meter (3 feet) at the ceiling. The roof will be a concrete slab 20 to 25 centimeters (7.9 to 9.8 inches) thick, supported by steel beams and concrete walls (CRWMS M&O 2000a, Attachment 1.1.6). The roof will shield radiation and withstand tornado-force winds.

The WHB complex will contain several nonradiological facilities that will be physically isolated from the main building to avoid possible interactions during an earthquake (CRWMS M&O 2000a, Attachment 1.1.6). The facility support area will include security and administrative offices and laboratories. It will be a two-story, steel-framed structure. The structure will consist of steel beams, columns, and bracing with metal-clad insulated siding and insulated roofing (CRWMS M&O 2000a, Attachment II, Section 1.1.6). The first floor will be concrete slab on grade; the second floor will be concrete slab on metal decking.

The other nonradiological facilities will be used for cleaning transport equipment and storing and preparing new waste packages. The structure for these facilities will be light steel framing with sheet metal siding. The floors will support the loads from rail and truck transporters bringing waste packages into the buildings and from storing prepared, empty waste packages. The WHB will house the electrical switchgear, diesel generator, and associated equipment.

### 4.2.3 Waste Treatment Building

The WTB will be one story, approximately 60 meters (200 feet) wide and 80 meters (260 feet) long. It will be an open, high-bay industrial structure. The main operating floor will consist of a reinforced concrete slab at grade level. The superstructure will be braced frames of structural steel, with insulated metal siding and an insulated metal roof. The WTB will not contain overhead cranes, and the major equipment will be anchored to the ground floor slab. The foundation will consist of individual spread footings of reinforced concrete, which will support the building columns (CRWMS M&O 2000a, Attachment II, Section 1.2.1). The WTB will be separated from the WHB by a seismic joint to prevent structural interaction during an earthquake (CRWMS M&O 2000a, Attachment II, Section 1.2.1).

## 4.3 PROCESS DESCRIPTION

For convenience, the description of MGR processes is divided into the following functional areas:

- Waste Receipt and Carrier/Cask Transport

- Carrier/Cask Preparation
- Waste Handling - Carrier Bay
- Waste Handling - Canister Transfer
- Waste Handling - Assembly Transfer
- Waste Handling - Disposal Container Handling and Waste Package Remediation
- Subsurface Transport and Emplacement
- Site-Generated Waste Treatment

These functional areas are also used in the identification and evaluation of MGR hazards as discussed in Section 5, Hazard and Design Basis Event Analysis. The following process descriptions provide the information required to support the safety assessment of Section 5. The descriptions are based on *Engineering Files for Site Recommendation* (CRWMS M&O 2000a).

### 4.3.1 Waste Receipt and Carrier/Cask Transport

Transportation casks containing SNF and vitrified HLW and associated carriers are received at the repository waste entry point or security gate. The SNF and HLW is contained in casks equipped with impact limiters and personnel barriers. At the security gate, the cask carrier and offsite prime mover are inspected for contraband, sabotage and radioactive contamination. Following inspection, the offsite prime mover is decoupled and an onsite diesel-driven prime mover is used to transport the carrier and cask to the CPB. The carrier/cask transport system also transports empty transportation casks and associated carriers from the WHB to the CPB for preparation and on to the repository security gate for dispatch from the site.

Transportation casks for commercial SNF may contain (1) uncanistered fuel assemblies, (2) single element, screened end disposable canisters, or (3) a canister loaded with assemblies. Defense HLW forms and most DOE-owned SNF are expected to be received in disposable canisters. A small amount of DOE fuel, of commercial origin, will be uncanistered.

### 4.3.2 Carrier/Cask Preparation

The carrier/cask transport system moves rail and truck cask configurations to and from the CPB, the security gate, and the WHB. Carrier preparation operations include moving a loaded carrier/cask into an available preparation bay with a site prime mover, performing radiation surveys, removing the personnel barrier, inspecting for contamination, measuring the cask temperature, and removing the impact limiters. The prepared carrier/cask can be transported to the WHB or transferred to a staging area. Truck or rail carriers with empty casks are prepared for offsite shipment by installing impact limiters, performing cask surveys, and installing personnel barriers. Remote handling equipment will be available for operator safety, to minimize radiation exposure and facilitate maintenance.

### 4.3.3 Waste Handling–Carrier Bay

Loaded transportation casks and associated carriers are transported from the CPB to the WHB carrier bay by the onsite diesel-driven prime mover (rail or tired vehicle). Incoming casks are prepared for waste removal by upending the cask on the carrier, lifting the cask

transportation casks and emptied dual-purpose canisters in overpacks onto carriers for shipment from the repository. The system performs these functions utilizing remotely operated cranes and manipulators, however, some direct contact operations may be required. The carrier/cask handling system will be located in the WHB.

### 4.3.4 Waste Handling–Canister Transfer

The canister transfer system will receive transportation casks on cask transfer carts through the cask transfer air lock into the cask preparation area. The cask preparation area will include a preparation station and a decontamination station. Remote handling equipment will consist of a cask transfer cart, cask preparation manipulator, and tools required to perform cask unbolting, venting, lid removal, and decontamination. Workers preparing a cask will sample the cask's vent ports, vent the cask and purge its gases to a monitored exhaust system, loosen the outer lid bolts, and secure a lifting fixture to the outer lid. The outer lid will then be removed and staged in the cask preparation area. Workers will repeat this process before removing the cask's inner lid. The cask transfer cart will then move the cask to the canister transfer cell (CRWMS M&O 2000a, Attachment II, Section 1.1.2).

The canister transfer system will unload the canisters from a cask, stage them (as required), load them into a new disposal container, and prepare the empty cask for shipment offsite. Canister transfer operations will be performed remotely in shielded canister transfer or off-normal canister handling cells. The canister transfer cell will consist of upper and lower transfer rooms, a cask unloading port, a cask loading port where canisters will be loaded into disposal containers, an off-normal canister transfer port, a small canister staging area, and a crane maintenance area. Small canisters will either be loaded directly into a disposal container or staged in the canister transfer cell until enough canisters are available to fill a disposal container. The canister transfer system will then deliver the loaded disposal containers to the disposal container handling system. Any canisters that are damaged, contaminated, or received in a condition that does not meet acceptance criteria will be considered off-normal. Off-normal canisters will be transferred to the off-normal canister handling cell for corrective action. Emptied transportation casks and associated handling fixtures will be delivered to the cask preparation area, decontaminated as required, closed, and transferred to the carrier/cask handling system (CRWMS M&O 2000a, Attachment II, Section 1.1.2).

The canisters will be removed from a transportation cask one at a time by remote equipment and placed in a disposal container, taken to the staging area, or moved through the port for off-normal canisters to the off-normal canister handling cell. Remote handling equipment in the transfer cell will include a 65-ton overhead bridge crane, an electromechanical manipulator, and a suite of small canister-lifting fixtures. The remote equipment will be designed to facilitate in-cell operations, maintenance, and recovery from off-normal events. A maintenance bay inside the cell will facilitate in-cell maintenance. (CRWMS M&O 2000a, Attachment II, Section 1.1.2).

An off-normal canister handling cell will be located next to the canister transfer cell, connected by the off-normal canister transfer tunnel. Special equipment will receive, handle and, if necessary, repackage off-normal canisters before final disposal in the repository. The cell's equipment will include a small overhead crane, a bridge-mounted electromechanical

manipulator, and two overpack loading and welding stations (for canisters with different diameters and heights).

### 4.3.5 Waste Handling–Assembly Transfer

The assembly transfer system will include the equipment, facilities, workers, and processes for preparing SNF assemblies for disposal in the repository.

Two nearly identical assembly transfer system lines will be housed in the WHB. Each will operate independently to handle waste throughput and support maintenance operations. Each will include a cask unloading area and a transfer cell area. The cask unloading area will contain an air lock, a cask preparation and decontamination area, and a pool area. The pool area will contain a cask unloading pool and an assembly staging pool. A single transfer canal will connect the two pools. An incline transfer canal will be used for moving the waste from the staging pool to the assembly handling cell. The transfer cell area will include an assembly handling cell, a disposal container loading cell, and a disposal container decontamination cell. The assembly transfer system will also include fuel basket storage pools and a special pool for nonstandard fuel, which will be located in an annex to the WHB. The physical arrangement of the assembly transfer system is documented in the *WHB/WTB Space Program Analysis for Site Recommendation* (CRWMS M&O 2000b, Section 6.2.1).

A transportation cask enters a cask preparation area through an air lock on a cask transfer cart. The cask preparation and decontamination area will include two cask preparation and decontamination rooms. Each room will contain a station for unloading and loading transportation casks from the cask transfer cart to a cask preparation pit. These stations will also be used to transfer empty transportation casks and dual-purpose canister overpacks on transfer carts to the decontamination area. The pit will also include a cask preparation manipulator and hoist that will be operated remotely. The system will contain a variety of remotely operated tools and accessories for preparing and decontaminating casks using the cask-preparation manipulator and hoist. Each assembly transfer system line will include a large overhead bridge crane.

The cask preparation and handling area equipment will include a cask transfer cart, a bridge crane that serves the cask unloading area, and two cask preparation manipulators with hoists mounted on gantries. The equipment will also include yokes for lifting casks and dual-purpose canister overpacks, handling fixtures, and remotely operated tools and accessories. The cask unloading and staging pools will be equipped with remotely operated assembly transfer machines mounted on the pool deck, grapples for lifting fuel assemblies, and cutting tools for removing lids from dual-purpose canisters. The cask unloading and assembly staging pools will contain dual-purpose canister overpacks, assembly baskets, basket staging racks, and transfer carts.

Remote or manual cask preparation operations consist of gas sampling, venting, lid unbolting and removal, gas and water cool-down, shield plug unbolting, and attachment of the shield-plug lifting fixture. If the cask contains individual spent fuel assemblies with no dual-purpose canister, it will be filled with water in the preparation pit and then transferred to the cask unloading pool.

If the cask contains a dual-purpose canister, workers will remove the cask outer lid while the cask is in the preparation pit. Using remotely operated and manual tools, workers will then open the vent valves on the dual-purpose canister; sample, vent, and cool the interior cavity; attach a lifting fixture to the canister; and fill the canister with water. The bridge crane and lifting yoke will transfer the cask, containing the dual-purpose canister, to the cask unloading pool.

If a cask contains individual spent fuel assemblies, the bridge crane, cask shield plug fixture, and lifting yoke will be used to remove its shield plug underwater in the cask unloading pool. If the cask contains a dual-purpose canister, the bridge crane, canister lifting fixture, and lifting yoke will be used to lift the canister from the cask and place it in a dual-purpose canister overpack. Using remote cutting tools, the operators will then sever and remove the dual-purpose canister lid. These activities will take place underwater in the cask unloading pool.

The cask unloading pool is connected by a transfer canal to the assembly staging pool. Another inclined transfer canal will connect the assembly staging pool to a dry cell handling area. Transfer canals that contain transfer carts for fuel baskets will connect both staging pools to fuel basket storage pools. Another transfer canal will connect the cask unloading pool and the nonstandard fuel pool.

A wet assembly transfer machine will remove the individual fuel assemblies from the opened shipping casks and dual-purpose canisters and load them into assembly baskets in the staging pool. The fuel will remain in these baskets until it is dried and placed in repository-qualified disposal containers. The fuel baskets will contain either four fuel assemblies from pressurized-water reactors or eight fuel assemblies from boiling-water reactors. The staging pool can hold a maximum of sixteen fuel baskets at any one time. When the assembly baskets in the staging pool are full, the wet assembly transfer machine will move the baskets to a transfer cart, which, in turn, will move the loaded fuel baskets to a fuel inventory pool or the assembly handling cell for disposal container loading.

The fuel inventory area, located in an annex to the WHB, will contain four fuel basket storage pools for SNF and one pool for nonstandard fuel. Each inventory pool will have the capacity to store a maximum of 750 fuel baskets loaded with SNF. Transfer canals that also connect to the assembly staging pool in each assembly transfer line will connect the fuel basket storage pools. The pools will have isolation gates so that, if necessary, one pool can be isolated from the other pools. The fuel inventory area will also have a separate pool for handling off-normal and damaged fuel assemblies. Spent fuel and basket-handing operations will be conducted under at least 3.35 meters (11 feet) of water for worker shielding.

The fuel assemblies will stay in the fuel basket storage pools until they are selected, according to their heat output, for placement in a disposal container. The maximum heat generation requirement for a disposal container loaded with SNF is 11.8 kW/hour (CRWMS M&O 2000a, Attachment II, Section 1.1.1). Any hot SNF loaded into the disposal container must be thermally blended with cold SNF to meet this limit. This procedure is called "fuel blending." Some fuel assemblies will remain in the inventory pool until they generate less heat from radioactive decay or until cooler fuel assemblies become available for blending. Approximately 12,000 spent fuel assemblies in 2,800 assembly baskets will accumulate in the fuel basket storage pools during the emplacement period to satisfy the blending requirement. The fuel basket storage pools will be

large enough to accommodate 5,000 metric tons of heavy metal (i.e., SNF); each pool will have a capacity of 1,250 metric tons of heavy metal, or 750 fuel baskets (CRWMS M&O 2000b, Section 6.2.1).

A fuel assembly is selected for placement in a disposal container according to the heat generation of the assemblies in the disposal container. Six assembly baskets and the fuel will be transferred from the fuel basket storage pools. The fuel inventory pool basket transfer machine will lift and place the fuel basket on a transfer cart, which will take the basket back to the assembly staging pool. The wet assembly transfer machine will move the assembly basket to another transfer cart for the inclined canal. This cart will transport the assembly basket up the inclined canal, out of the pool water, and into the dry assembly handling cell.

The dry assembly handling cell will contain a disposal container loading port, an assembly transfer machine, an in-cell manipulator, an in-cell service crane, and a maintenance bay. A dry assembly transfer machine will move the assembly basket into one of two drying vessels. It will be necessary to dry the fuel assemblies to meet repository waste package performance criteria. After drying the assemblies, the machine will remove them from the drying vessel and load them into a disposal container. The disposal container will be joined to the disposal container loading port below the assembly handling cell. The dry assembly transfer machine will reinstall the sealing device and the disposal container's inner lid. The transfer cart will then transfer the disposal container to the decontamination cell, where the top lid area and the inner-lid sealing device will be decontaminated. The system will then evacuate the disposal container internal cavity and fill it with nitrogen gas. Finally, the transfer cart will transfer the disposal container to the disposal container handling system for lid welding and inspection.

All assembly transfer system remote operations will be controlled from operating galleries next to each assembly handling cell. Strategically located closed-circuit television systems and shield windows will be used to monitor remote operations. Transfer cell area equipment will be designed to facilitate remote operation and removal for contact decontamination and maintenance. Interchangeable components will be provided where appropriate. The assembly transfer system will also be designed to provide safe and efficient recovery from equipment failures and malfunctions.

### 4.3.6 Waste Handling–Disposal Container Handling and Waste Package Remediation

#### 4.3.6.1 Disposal Container Handling

The disposal container handling cell will be a large, shielded structure containing areas for several welding and inspection stations, staging of loaded containers, transfer cart operations, tilting the container to a horizontal position, and maintenance of the overhead cranes. Handling operations for disposal containers will involve two remotely operated bridge cranes and hoists, as well as peripheral equipment. An empty disposal container will be lifted by one of the cranes. The container will either be staged or directly transferred to a transfer cart servicing one of the two assembly transfer system or canister transfer system lines. The empty container will be taken to the assembly transfer system or the canister transfer system for loading. When-loaded, the disposal container will be returned to the staging area or to one of eight welding stations (CRWMS M&O 2000b, Section 6.2.1). Each welding station will be equipped with a robotic

gantry, a turntable, and multiple sealing tools. The outer lids for the disposal container will be staged near the welding stations for sealing after the container is loaded.

A transfer cart will transfer disposal containers between the disposal container handling cell, the decontamination cell, and the loading cell. An isolation door will separate the loading cell and the decontamination cell, and a shield door will separate the decontamination cell and the handling cell. A loading port mating device in the loading cell will provide a contamination barrier between the assembly handling cell, the disposal container loading port, and the disposal container during transfer of SNF. The decontamination cell will be equipped with a bridge-mounted inerting manipulator, a bridge-mounted decontamination manipulator, a decontamination tool, and a contamination sample pass-through glove box. Contamination survey samples will be transferred using the pass-through glove box into an adjacent operating gallery for counting.

The disposal container handling system will receive a loaded and temporarily sealed disposal container from the assembly transfer system or the canister transfer system, then transfer it to a staging area or a welding station. Welding and sealing will include a number of steps and remote equipment operations. Additional steps and remote equipment will also be required to conduct weld inspections and post-weld heat treatment operations. Following weld inspection and weld certification, the container will either be staged or prepared for transfer to the repository. A loaded, closed, welded, inspected, and certified disposal container is called a waste package.

The cranes in the disposal container handling cell will be used to lift and transfer a loaded container to one of the eight independent lid-welding stations. A remotely controlled robotic gantry will set up, prepare, weld, and backfill the container with inert gas. The gantry will also serve as the remote handling platform to inspect the sealing operations, which will include securing the disposal container to the welding station's turntable, removing temporary sealing devices, purging the lid with inert gases for welding, backfilling the container with helium prior to closure, turning the container, welding the inner lid, installing the outer lids, and welding the outer lids. Welding will be performed using automatic welders deployed from the robotic gantry platform such that they can be removed from the cell for retooling, testing, adjustments, and maintenance. This feature eliminates the need for personnel to enter the radiation environment in the handling cell. The robotic gantry may be withdrawn into a welder maintenance bay through a welder service room, where a number of contact change-out, service, and repair operations can be performed. The welder maintenance bay will be located next to the disposal container handling cell.

One welder room will be provided for each of the eight welders. The welder room will provide access to the robotic gantry, welder, non-destructive examining equipment, and post-weld heat-treating equipment (CRWMS M&O 2000a, Attachment II, Section 1.1.5). Access and service work on the equipment will be possible in these rooms without exposing the workers to the atmosphere and radiation sources in the disposal container handling cell.

The staging area for loaded disposal containers will be used to stow loaded disposal containers or waste packages awaiting transfer to the waste package transporter loading cell. Waste handling simulations have shown that staging 20 loaded disposal containers and/or waste packages in the

disposal container handling cell can accommodate a two-week interruption in repository emplacement operations (CRWMS M&O 2000a, Attachment II, Section 1.1.5).

To reduce radiation levels in the crane maintenance bay, loaded disposal containers will be staged in a separate area inside the disposal container handling cell. This area will have partial walls and an access door to facilitate transfers of disposal containers to and from staging locations. The partial walls will provide shadow shielding for the main portion of the cell and the maintenance bay. The design configuration incorporates both distance and shielding by isolating radiation sources to one area of the handling cell and adding a wall separating the staged disposal containers from the welding, handling, and crane maintenance areas. This will significantly reduce radiation doses to equipment during normal operations, while also reducing radiation levels during manned entry into the cell for periodic maintenance and test operations.

The final handling sequence for the surface facilities involves repositioning the waste package to a horizontal position, transferring the sealed waste package to a decontamination and transporter loading cell, and loading the waste package onto the waste emplacement system transporter. These operations include lifting, transferring, final decontamination, final inspection, certification, and data recording. The operations will be performed using a remotely operated horizontal transfer cart, a waste package horizontal lifting system, decontamination and inspection manipulators, and the waste package transporter.

Only one transporter loading line will be available for the final decontamination, inspection, transfer, and loading of waste packages onto a transporter. The waste package, once it is moved into the transporter loading cell from the disposal container handling cell, will be lifted off the horizontal transfer cart using the lifting collar, the base collar, and the horizontal lifting machine. While suspended, the waste package will be decontaminated, inspected, and certified. Important data needed for repository record keeping will be recorded. The emplacement pallet of the transporter will then move into the cell, and the waste package will be lowered onto the pallet. The handling collars will then be remotely removed and taken out of the waste package transporter loading cell for reuse. Any contamination picked up during disposal container sealing will be manually removed in contaminated equipment rooms before the collars are transferred to the empty disposal container preparation area for reuse.

A transporter air lock will be provided at the exit of the transporter loading line so that the waste package transporter vehicle may enter and be docked for loading. The air lock will prevent movement of air between the transporter loading cell and the outside atmosphere. In the final surface waste handling steps, the waste package pallet will be pulled into the shielded waste package transporter, the transporter shield doors will be closed, and the waste package transporter will be disengaged from the loading cell dock. Then the waste package and pallet will be hauled into the repository.

#### 4.3.6.2 Waste Package Remediation

When a waste package is found to be abnormal or damaged, workers will transfer it from the disposal container handling system to the waste package remediation system. This system will be housed in a multipurpose cell inside the WHB.

The waste package remediation system will receive disposal containers and waste packages that have failed the weld inspection processes and that are defective or abnormal. Repairs to abnormal or damaged waste packages (or disposal containers, if they are not correctly sealed and inspected) will be performed remotely. After examination, repair, or, if necessary, unsealing the damaged disposal containers or waste packages, the remediation system will deliver them back to the disposal container handling system (CRWMS M&O 2000a, Attachment II, Section 1.1.4).

If inspections of the closure weld reveal an unacceptable but repairable welding defect, the disposal container will be examined, prepared for rewelding, and unsealed if necessary. Correction of rejected closure welds will require removal of the weld material in such a way that the disposal container can be returned to the disposal container handling system to complete the closure welding process. If examination of the closure weld shows the defect or damage to be irreparable, the container will be opened. If a waste package is retrieved from the repository for any reason—suspected damage or known failure—it will be opened in the waste package remediation system (CRWMS M&O 2000a, Attachment II, Section 1.1.4).

The processes for opening a waste package or disposal container will include remotely cutting the closure weld, collecting and processing the cutting fines, removing and disposing of the cutting waste, and installing a temporary seal to contain contamination to the inside of the container.

All remediation operations on radioactive waste packages or disposal containers will be performed remotely in a dedicated, shielded cell accessible directly from the large handling cell inside the disposal container handling system. The remediation cell will accommodate one waste package or disposal container at a time. A shield door will open to allow the transfer cart to enter. After the transfer cart enters the remediation cell, the damaged container will be positioned at one of two work stations in the remediation cell, and will exit the cell without being removed from the cart. The two remotely operated work stations will accommodate different repair tasks. One will facilitate cutting the lids of the containers, removing them, and staging them. The other will allow remote inspection, examination, and purging of the container, as well as backfilling it with inert gas, temporarily sealing it, and decontaminating it (CRWMS M&O 2000a, Attachment II, Section 1.1.4).

The remediation system will use a variety of remotely operated equipment, including an overhead bridge crane, an in-cell multipurpose manipulator, a lid-cutting machine, and closed-circuit television viewing systems. System operations will be performed remotely using equipment designed to facilitate decontamination, maintenance, and replacement of interchangeable components, as required (CRWMS M&O 2000a, Attachment II, Section 1.1.4).

For closure welds rejected because of minor damage or abnormality, the remediation system will accommodate the removal of weld material in such a way that the welding station of the disposal container handling cell will be able to correct the abnormality. If examination of the closure weld indicates an irreparable welding defect, or if a waste package has been retrieved because of suspected failure or damage, the package will be opened. Opening waste packages and disposal containers should be infrequent, but it will require the capability to unseal the container and vent it. Opening a sealed container will require remotely cutting the closure welds of the inner and outer lids, removing and staging the lids, collecting and processing cutting fines, removing and

disposing of cutting waste, and installing a temporary seal to contain contamination to the inside of the container. Following remediation, the container will be inspected for contamination and remotely decontaminated, as required. The container will then be returned to the disposal container handling system for rewelding, transferred to the assembly transfer system for unloading of fuel assemblies, or transferred to the canister transfer system for unloading of canisters (CRWMS M&O 2000a, Attachment II, Section 1.1.4).

### 4.3.7 Subsurface Transport and Emplacement

A typical route and sequence of events in the process of subsurface transport and emplacement would be:

- The waste package will be loaded onto the transporter at the WHB.
- The locomotives will move the loaded transporter from the WHB to the North Portal, down the North Ramp to the North Ramp Extension, and from there to the preselected emplacement drift turnout. The train will generally follow the shortest route to the emplacement drift; however, orientation of the transporter is important depending on whether the emplacement drift is going to be reached from the east or west mains. The reason is that the open deck of the transporter has to face the drift docking area for transfer of the waste package. The transporter and locomotives, once they enter the subsurface area in the early emplacement years, cannot rotate or change the orientation of the transporter. Therefore, the transporter must be oriented in the proper direction at the surface facilities using railroad turnouts.
- If an emplacement drift will be reached from the west main, the train will get there by going around the north end.
- The return trip to the surface will follow the same route in the reverse sequence. The transporter's open deck must face the WHB to accept another waste package. Therefore, the transporter must be properly oriented before docking at the WHB.

The waste package handling sequence will not change throughout the waste emplacement period. Waste package transportation for emplacement is always by rail. The sequence can be summarized as follows (CRWMS M&O 2000c, Section 6; CRWMS M&O 2000d, Section 6):

1. The waste package, loaded on an emplacement pallet, is placed on the deck of the transporter waiting at the receiving dock of the WHB.
2. A semi-rigid chain mechanism pulls the pallet and waste package into the shielded enclosure of the transporter.
3. The shielded enclosure doors are closed to protect the operators, and a primary locomotive pulls the loaded transporter away from the WHB docking area.
4. During a stop at a track turnout outside the WHB, a secondary locomotive joins the train by coupling itself to the transporter. Both locomotives are driven by operators.

5. Both locomotives, one in front and one behind, move the loaded transporter into the subsurface facilities.

6. The train stops near the predetermined emplacement drift turnout.

7. The locomotive controls are turned over to remote control operators in a control center at the surface, and the locomotive operators leave the locomotives and move to a designated location to protect themselves from radiation while the emplacement drift doors are open.

8. The locomotive at the rear of the transporter is decoupled from the transporter by remote control operators. The locomotive in front of the transporter moves the transporter into the turnout and stops before reaching the emplacement drift docking area.

9. The transporter doors are fully opened, and the emplacement drift isolation doors are also fully opened from the surface control center.

10. The locomotive docks the transporter, pushing the open deck section of the transporter completely inside the emplacement drift.

11. The semi-rigid chain mechanism pushes the pallet and waste package from inside the shielded transporter enclosure to the open deck area of the transporter.

12. The waste package emplacement gantry, also riding on rails and remotely operated, moves from inside the emplacement drift completely over the waste package and pallet, straddling the transporter's open deck. The gantry lifts the waste package by its pallet, and moves back into the drift to the waste package emplacement location.

13. The locomotive moves the transporter away from the emplacement drift docking area and stops. The transporter doors and the drift doors are completely closed.

14. The locomotive moves the transporter from the turnout to the main drift and stops to allow the second locomotive to couple to the train.

15. The operators board the locomotives, the controls are turned back to manual operation, and the train proceeds to the surface for another cycle.

16. Before docking at the WHB, the train stops, the locomotive in front of the transporter is decoupled and moved away to a standby location, and the locomotive behind the transporter moves the transporter to the building dock.

### 4.3.8 Site-Generated Waste Treatment

Liquid and solid LLW will be generated by the waste handling facilities within the controlled area. The site-generated waste treatment process will include collecting solid and liquid waste, processing liquid waste to enable reuse of water, placing the waste in containers so that it will be

suitable for disposal, and staging waste to wait for offsite disposal. The WHB and WTB will house the systems that process the waste streams described in this section.

In the surface waste preparation and handling operations, solid and liquid LLW will be produced, most of which will be from the WHB. The operations will include decontamination of shipping casks and equipment, pool water cleanup, and air cleanup. Some of these operations will be performed with chemicals and wipes, which will also become waste. Waste will also be collected from the WHB pool skimming and filtration equipment; cuttings from the process of opening dual-purpose canisters; contaminated tooling and clothing; facility ventilation filters; chemical sumps; and carrier and transporter washdowns. Used (opened and unloaded) dual-purpose canisters will be considered LLW; they will be placed in an overpack suitable for shipping. The used canisters will be packaged for offsite shipment at the WHB and will not be processed by the system described in this section. Other waste will arrive in sealed containers on the site transportation system. Recyclable liquid waste will be treated and made available for users. Non-recyclable liquid will be packaged for disposal. Compactible solid waste will be sorted, compacted, and packaged for offsite disposal. Noncompactible solid waste will be cut into pieces, or otherwise physically reduced in size, and packaged for offsite disposal. Non-recyclable waste will be processed and compacted in 55- or 85-gallon drums and readied for shipment to a LLW disposal facility.

It is not anticipated that mixed waste will be produced as a result of normal operations. The generation of mixed waste will be minimized by controlling the use of hazardous materials. If mixed waste is generated, it will be isolated and packaged for disposal. No permanent equipment is provided that will process mixed waste, however, an area will be provided to stage packaged mixed waste awaiting disposal.

## 4.4 IMPORTANT TO SAFETY SYSTEMS

Proposed rule 10 CFR Part 963 (64 FR 67086) provides the requirements for a preclosure suitability evaluation for the Yucca Mountain site. Specifically, 10 CFR 963.13(b)(4) requires a preliminary description of the site SSCs and operator actions intended to mitigate or prevent accidents.

*Revised Interim Guidance Pending Issuance of New U.S. Nuclear Regulatory Commission (NRC) Regulations* (Revision 01, July 22, 1999), for Yucca Mountain, Nevada (Dyer 1999) also requires the identification of MGR important to safety SSCs and provides specific guidelines for their identification. These guidelines were used in the preliminary identification of MGR important to safety SSCs.

This section provides a description of the important to safety SSCs relied upon to protect the health and safety of the public and the onsite worker as well as the process for identifying these important to safety SSCs. The measures taken to ensure important to safety SSCs function as designed are also described.

### 4.4.1 Identification of Important to Safety Structures, Systems, and Components

Important to safety SSCs are identified through a classification process performed in accordance with MGR procedure QAP-2-3. Guidance provided by procedure YAP-2.7Q is also used in this

classification process. The process consists of establishing the configuration and function of MGR SSCs and the effects of the SSCs on MGR safety. This information is then evaluated against criteria provided in QAP-2-3 to determine the Quality Assurance classification of the particular item. The classification criteria are provided in the form of checklists in procedure QAP-2-3. The following classification categories are specified by QAP-2-3 to meet the requirements of Section 2 of the *Quality Assurance Requirements and Description* (DOE 2000).

Quality Level 1 (QL-1): Those SSCs whose failure could *directly* result in a condition adversely affecting public safety. These items have a high safety or waste isolation significance.

Quality Level 2 (QL-2): Those SSCs whose failure or malfunction could *indirectly* result in a condition adversely affecting public safety, or whose *direct* failure would result in consequences in excess of normal operational limits. These items have a low safety or waste isolation significance.

Quality Level 3 (QL-3): Those SSCs whose failure or malfunction would not significantly impact public or worker safety, including those defense-in-depth design features intended to keep doses ALARA. These items have a minor impact on public and worker safety and waste isolation.

Conventional Quality: Those SSCs not meeting any of the criteria for Quality Levels 1, 2, or 3. Conventional quality items are not subject to the requirements of the *Quality Assurance Requirements and Description* (DOE 2000).

All MGR SSCs classified QL-1, QL-2, and QL-3 are important to safety and subject to the requirements of the Quality *Assurance Requirements and Description* (DOE 2000).

This classification method is based on an iterative design-classification process where each analysis iteration is considered a final product for that phase of design. MGR SSC classifications therefore, will be reevaluated as necessary using a methodology appropriate to the level of DBE analysis and system design detail. The approach is consistent with NUREG-1318, *Technical Position on Items and Activities in the High-Level Waste Geologic Repository Program Subject to Quality Assurance Requirements* (Duncan et al. 1988, Section 4.2(a)), which allows engineering judgement and conservative bounding assumptions to be used in cases where data are limited.

The following sections describe the input information utilized in the Quality Assurance classification of MGR SSCs.

#### 4.4.1.1 MGR Design Configuration and Architecture

Prior to the Quality Assurance classification of MGR SSCs, the system architecture to successive level of detail and the system design configuration as well as the function of SSCs are established. The classification analyses are based upon the system design and functions as established by the System Description Documents and the *Monitored Geologic Repository Concept of Operations* (CRWMS M&O 1999ah). The *Monitored Geologic Repository Project*

*Description Document* provides general safety requirements and architecture to the System Description Documents (CRWMS M&O 2000e).

#### 4.4.1.2 MGR Safety Strategy

The purpose of this section is to present a description of the strategy to prevent or mitigate preclosure offsite exposure for the MGR. This preclosure safety strategy (PSS) has been used to establish site recommendation design requirements. The strategy allows the engineering functions of requirement development, design and verification to focus on the features that are important to preclosure safety. The strategy focuses on the offsite exposure constraints mandated by Dyer (1999). In addition, this strategy considers preclosure activities that occur prior to and including the activities required to begin the postclosure period. This strategy is a general plan and does not provide requirements for the potential repository. The approach advocated in this strategy will be considered in developing those requirements as well as criteria for data acceptance. The requirements for SSCs are provided in the System Description Documents.

##### 4.4.1.2.1 Evolution of the Preclosure Safety Strategy

The PSS will be reviewed as the MGR design matures with the expectation that PSS will be expressed at a greater level of detail as the design matures. The evolution of the PSS for the potential repository including surface and subsurface SSCs is made because of increasing site information and design development. In developing the PSS at the site recommendation stage, consideration has been given to the safety standards of Dyer (1999), constraints related to the postclosure safety case, and the preclosure operational functions associated with receiving, handling, packaging, storing, emplacing, and retrieving the waste. One of the goals in developing the PSS is to minimize the complexity of the design while ensuring that the optimum safety design features are in place to provide public and worker safety.

The PSS provides an allocation of event prevention and consequence mitigation features to SSCs to address concurrent construction and waste handling operations and equipment based upon the operational functions being performed. Defense-in-depth features are included in the PSS as determined by the use of a risk-informed process that results in identification of DBEs.

Importance to safety is defined in Dyer (1999) as follows:

> "Importance to safety, with reference to structures, systems, and components, means those engineered features of the geologic repository operations area whose function is:
>
> (1) To provide reasonable assurance that high-level waste can be received, handled, packaged, stored, emplaced, and retrieved without exceeding the requirements of Sec. 111 for Category 1 design basis events; or
>
> (2) To prevent or mitigate Category 2 design basis events that could result in doses equal to or greater than the values specified in Sec. 111(b)(2) to any individual located on or beyond any point on the boundary of the preclosure controlled area."

Consideration of this definition is important to developing an acceptable PSS. Two DBE categories are defined in Dyer (1999) as follows:

- DBE Category 1 describes "Those natural and human-induced event sequences that are expected to occur one or more times before permanent closure of the geologic repository operations area."

- DBE Category 2 consists of "(a) Other human-induced event sequences that have at least one chance in 10,000 of occurring before permanent closure of the geologic repository, and (b) appropriate consideration of natural events (phenomena) that have been historically reported for the site and the geologic setting."

Because different dose limits apply to each DBE category, it is necessary to assess the frequencies of the DBEs and then group them into the appropriate category. The threshold frequency for identification of credible DBEs is interpreted to be greater than 1E-6 per year. Based on the DBE definitions from Dyer (1999), DBE categories are defined as follows based on a preclosure operational period of 100 years:

- Category 1: $\lambda \geq$ 1E-2 per year
- Category 2: 1E-6 $\leq \lambda <$ 1E-2 per year
- Not Credible: $\lambda <$ 1E-6 per year.

where $\lambda$ is defined as event frequency.

Conversion of the category frequency thresholds from a basis of "expected occurrences per year" assumes that the preclosure period will be 100 years and that event frequencies will be average annual frequencies (averaged over the preclosure period). Conservatism (or nonconservatism) can be introduced based on the validity of these assumptions. Surface facility handling and subsurface emplacement of waste packages is expected to last approximately 30 years. Using this more realistic period would result in a Category 1 cutoff at 3.3E-2 per year and potentially allow more DBEs to be compared with the less restrictive Category 2 dose limits. Assuming a preclosure period of 300 years would lower the Category 1 cutoff to 3.3E-3 per year and lower the Category 2 cutoff to 3.3E-7 per year. However, since the waste handling and emplacement operations will be complete after approximately 30 years, no additional surface events would be identified. For the subsurface, it is not expected that any new events would be identified for a longer preclosure time. The MGR will be designed to meet the applicable regulatory DBE dose limits for the preclosure period selected. The preclosure safety assessment provided herein shows that the MGR is likely to meet the regulatory dose limits for a reasonably conservative 100-year preclosure period. Also, because DBEs have been categorized based on maximum annual throughput, a considerable measure of conservatism is built into the annual frequency cutoffs. These frequency cutoffs are interpreted to apply to DBE scenarios that include an initiating event and any subsequent failures that result in a radionuclide release.

Dyer (1999, Section 112) requires safety assessments for criticality control. Systems for receiving, handling, emplacing, and isolating the waste are to be designed with means to control nuclear criticality during normal operations and assuming occurrence of DBEs.

Hazard assessment methods are being used to identify events with potential radiological consequences that are applicable to the MGR during the preclosure operational period. The methodology used provides a systematic method to identify and group DBEs (see Section 5, Hazard and Design Basis Event Analysis). The process of DBE identification is an iterative design process, coupled with requirements and design development. This iterative process continues until the design phases are completed.

#### 4.4.1.2.2 Preventing/Mitigating Preclosure Offsite Exposure

The PSS is based on the functions of MGR operations: (1) Receipt of Waste, (2) Transfer of waste to Lag Storage (as required), (3) Packaging/Sealing of the Disposal Container, (4) Transfer of Waste Package to an Emplacement Drift, and (5) Waste Package Emplacement in Drift. The safety strategy for each of the basic functions is either; containment/confinement augmented by prevention, or prevention augmented by containment/confinement. Containment/confinement is the utilization of features to ensure that offsite exposures are less than 10 mrem/year for Category 1 DBEs (ALARA constraint), and 5 rem/yr for Category 2 DBEs. Prevention is the utilization of features that ensure the frequency of occurrence is either less than 1E-2/yr to prevent accidents from occurring during the MGR preclosure lifetime or less than 1E-6/yr to eliminate the event from the design basis. No operator actions are assumed in the prevention or mitigation of MGR DBEs.

As the design matures PSS concepts chosen for each of the operational functions will be expressed in terms of the MGR location, the specific SSCs relied upon for safety at this location, the MGR facility functional safety requirements, MGR facility defense-in-depth SSCs, and defense-in-depth functional description. The defense-in-depth features will be determined, as required, when showing compliance with the risk-informed guidance provided in Dyer (1999). Requirements of 10 CFR Part 20 and ALARA will also be addressed.

The PSS is discussed in the following sections for each of the general MGR operational functions.

##### 4.4.1.2.2.1 Receipt of Waste

This function covers the period from the time the waste arrives on site until the transportation cask is opened. The safety strategy is to handle transportation casks when in the Cask Preparation Building and the Carrier Bay area of the WHB within the cask design basis, such that events that result in a breach of a cask are not credible. Containment is provided by the transportation cask. Prevention is provided by the surface facility SSCs that will be designed to prevent events that could exceed the cask design basis during preclosure. For canistered DOE fuels, the canister also provides containment within the cask.

##### 4.4.1.2.2.2 Transfer of Waste to Blending Inventory or to the Disposal Container

This function covers the period from waste removal from the transportation cask to the blending inventory area (as required) and then from this area to placement in a disposal container, followed by transfer of the disposal container to the sealing area.

After the cask lid bolts are removed within the WHB, prevention is provided by the surface facility SSCs that will be designed to prevent events that result in an unsealed transportation cask drop within the preclosure operations time period, reliably handle transfers of SNF when not in the pools, and reliably maintain pool water levels. Containment/confinement is provided by the pool water and the WHB HEPA filter for commercial SNF.

DOE SNF is received in sealed canisters inside transportation casks. The sealed canisters within the unsealed transportation cask provide containment during cask transfer to the canister transfer area and during disposal container transfer to the sealing area. During canister transfer from the cask to storage or the disposal container, prevention is provided by the surface facility SSCs that will make a canister breach a beyond-design-basis event. Confinement is also provided by the WHB HEPA filters.

#### 4.4.1.2.2.3 Packaging/Sealing of the Disposal Container

This function covers the handling of waste from the initial receipt in the disposal container sealing area to loading into the transporter.

For commercial SNF, prevention of a DBE is provided by the surface facility SSCs that will be designed to reliably handle disposal containers. Confinement is provided by WHB HEPA filters.

For canistered DOE fuels, containment is provided by the canister. Prevention is provided by the surface facility SSCs that will be designed to prevent disposal container events that are beyond the canister design basis during preclosure. Confinement is also provided by the WHB HEPA filters.

#### 4.4.1.2.2.4 Transfer of Waste Package to an Emplacement Drift

This function covers the handling of waste from the time the waste package is loaded into the transporter through parking the transporter at the entrance to the emplacement drift. Three locations are considered:

A. Before descent on ramp–Containment is provided by the waste package. For waste packages containing canistered DOE fuel, the canister also provides containment. Prevention is provided by the transporter and rail system to ensure that no credible events can occur that are beyond the waste package design basis during preclosure.

B. During descent–Containment is provided by the waste package. For waste packages containing canistered DOE fuel, the canister also provides containment. Prevention is provided by the transporter and rail system to ensure that no credible events can occur that are beyond the waste package design basis during preclosure.

C. During parking at emplacement drift–Containment is provided by the waste package. For waste packages containing canistered DOE fuel, the canister also provides containment. Prevention is provided by the transporter, rail system, emplacement handling, pallet and ground support to ensure that no credible events can occur that are beyond the waste package design basis during preclosure.

#### 4.4.1.2.2.5 Waste Package Emplacement in Drift

This function covers handling the waste package from the entrance to the emplacement drift through emplacement and storing in the emplacement drift.

Containment is provided by the waste package. For waste packages containing canistered DOE fuel, the canister also provides containment. Prevention is provided by the rail system, emplacement handling, and pallet and ground support to ensure that no credible events can occur that are beyond the waste package design basis during preclosure.

### 4.4.1.3 Design Basis Event Analysis

A preliminary analysis of MGR DBEs (CRWMS M&O 1998a) has been performed to determine the effects of internal and external events on facility safety and in the classification of MGR SSCs. The DBE analysis, presented in Section 5, addresses both the DBE frequencies and dose consequences at the site boundary.

The classification analyses utilizes the results of the DBE analysis to evaluate MGR SSCs against the classification criteria of procedure QAP-2-3. It should be noted that the performance of transportation casks and the standard DOE canister to provide radionuclide containment and criticality control is accounted for in the assessment of DBEs and the selection of SSCs that are important to safety.

## 4.4.2 Important to Safety Structures, Systems, and Components

Using the method described in Section 4.4.1, MGR SSCs were evaluated against the criteria of procedure QAP-2-3 to determine the item's Quality Assurance classification level. QL-1, QL-2, and QL-3 SSCs are shown in Tables 4-1, 4-2, and 4-3, respectively.

Table 4-1. Quality Level 1 Important to Safety Items

| SSC | MGR System | Important to Safety Function | Reference |
|---|---|---|---|
| Assembly Transfer Baskets | Assembly Transfer System | Provide criticality control for SNF assemblies. | CRWMS M&O 1999b |
| Basket Staging Racks | Assembly Transfer System | Provide criticality control for SNF assemblies. | CRWMS M&O 1999b |
| Disposal Containers | Canistered SNF Disposal Container | Provide containment and criticality control for SNF assemblies. | CRWMS M&O 1999c |
| | Defense HLW Disposal Container | | CRWMS M&O 1999d |
| | DOE SNF Disposal Container | | CRWMS M&O 1999e |
| | Non-Fuel Components Disposal Container | | CRWMS M&O 1999f |
| | Uncanistered SNF Disposal Container | | CRWMS M&O 1999g |
| | Naval SNF Disposal Container | | CRWMS M&O 1999h |
| Locomotives | Waste Emplacement System | Prevent uncontrolled descent of the waste package transporter and the possible impact of a waste package with the subsurface facility structure or other facility equipment resulting in radiological release. | CRWMS M&O 1999i |
| Small Canister Staging Racks | Canister Transfer System | Provide criticality control for defense HLW canisters. | CRWMS M&O 1999j |
| Waste Package Transporter | Waste Emplacement System | Prevent uncontrolled descent of the waste package transporter and the possible impact of a waste package with the subsurface facility structure or other facility equipment resulting in radiological release. | CRWMS M&O 1999i |
| WHB Structure | WHB System | Provides containment of radioactive materials, radiation shielding, and protection of equipment from internal and external hazards. | CRWMS M&O 1999k |
| Waste Retrieval System | Waste Retrieval System | Prevent uncontrolled decent of the waste package transporter and the possible impact of a waste package with the subsurface facility structure or other facility equipment resulting in radiological release. | CRWMS M&O 1999l |

Table 4-2. Quality Level 2 Important to Safety Items

| SSC | MGR System | Important to Safety Function | Reference |
|---|---|---|---|
| Assembly Drying System | Assembly Transfer System | Collect and manage site-generated radioactive waste produced in the assembly drying process | CRWMS M&O 1999b |
| Backfill Emplacement System | Backfill Emplacement System | Maintain structural integrity in the event of a DBE, e.g., seismic event. | CRWMS M&O 1999m |
| Bridge Cranes | Assembly Transfer System<br>Carrier/Cask Handling System<br>CPB Materials Handling System<br>Canister Transfer System<br>Disposal Container Handling System<br>Waste Package Remediation System | Maintain structural integrity in the event of a DBE, e.g., seismic event.<br>Prevent interactions with QL-1 structures, systems, and components. | CRWMS M&O 1999b<br>CRWMS M&O 1999n<br>CRWMS M&O 1999o<br>CRWMS M&O 1999j<br>CRWMS M&O 1999p<br>CRWMS M&O 1999q |
| Control and Tracking System | Assembly Transfer System | Prevent drop of assembly transfer basket during transfer of SNF assemblies. | CRWMS M&O 1999b |
| Control and Tracking System | Canister Transfer System<br>Disposal Container Handling System<br>Waste Package Remediation System | Support site-generated radiological waste collection and management functions. | CRWMS M&O 1999j<br>CRWMS M&O 1999p<br>CRWMS M&O 1999q |
| Cooling System | Assembly Transfer System | Collect and manage the site-generated radioactive waste generated in the SNF container cooling process. | CRWMS M&O 1999b |
| Disposal Container Inerting System | Disposal Container Handling System | Collect and manage the site-generated radioactive waste generated in the disposal container inerting process. | CRWMS M&O 1999p |
| Disposal Container Loading Port Mating Device | Assembly Transfer System | Maintain structural integrity in the event of a DBE, e.g., seismic event. | CRWMS M&O 1999b |
| Disposal Container Weld Station Jib Crane | Disposal Container Handling System | Maintain structural integrity in the event of a DBE, e.g., seismic event. | CRWMS M&O 1999p |

Table 4-2. Quality Level 2 Important to Safety Items (Continued)

| SSC | MGR System | Important to Safety Function | Reference |
|---|---|---|---|
| Decontamination Systems | Assembly Transfer System | Collect and manage the site-generated radioactive wastes generated in the process of facility and equipment decontamination. | CRWMS M&O 1999b |
| | Canister Transfer System | | CRWMS M&O 1999j |
| | Disposal Container Handling System | | CRWMS M&O 1999p |
| | WHB System | | CRWMS M&O 1999k |
| | Waste Package Remediation System | | CRWMS M&O 1999q |
| Dry Assembly Transfer Machine | Assembly Transfer System | Maintain structural integrity in the event of a DBE, e.g., seismic event. Prevent drop of assembly transfer basket during transfer of SNF assemblies. | CRWMS M&O 1999b |
| Dual-Purpose Canister Lid Severing Tool | Assembly Transfer System | Collect and manage radiologically contaminated metal chips generated during dual-purpose canister lid removal operations. | CRWMS M&O 1999b |
| Emergency Power Source and Distribution System | WHB Electrical System | Support the WHB primary ventilation system to mitigate the consequences of a facility DBE. | CRWMS M&O 1999r |
| Emplacement Drift Ground Control | Ground Control System | Prevent breach of waste package in emplacement drift due to rockfall | CRWMS M&O 1999s |
| Emplacement Gantry | Waste Emplacement System | Maintain structural integrity in the event of a DBE, e.g., seismic event. | CRWMS M&O 1999i |
| Fire Detection Systems | WHB Fire Protection System | Protect QL-1 SSCs from the effects of fire. | CRWMS M&O 1999t |
| Fire Suppression Systems | WHB Fire Protection System | Protect QL-1 SSCs from the effects of fire. | CRWMS M&O 1999t |
| Lifting Fixtures, Cask and Dual-Purpose Canister Preparation System | Assembly Transfer System | Maintain structural integrity in the event of a DBE, e.g., seismic event. | CRWMS M&O 1999b |
| Lifting Fixtures, Disposal Container Handling System | Disposal Container Handling System | | CRWMS M&O 1999p |

Table 4-2. Quality Level 2 Important to Safety Items (Continued)

| SSC | MGR System | Important to Safety Function | Reference |
|---|---|---|---|
| Lifting Fixtures, Dry Assembly Handling System | Assembly Transfer System | Prevent drop of assembly transfer basket during transfer of SNF assemblies. | CRWMS M&O 1999b |
| Liquid LLW System | Site Generated Radiological Waste Handing System | Collect and manage site-generated radioactive wastes generated in the operation of MGR facilities. | CRWMS M&O 1999u |
| Mixed LLW System | Site Generated Radiological Waste Handing System | Collect and manage site-generated mixed wastes generated in the operation of MGR facilities. | CRWMS M&O 1999u |
| MGR Operations Monitoring and Control System | MGR Operations Monitoring and Control System | Mitigate the consequences of a facility DBE. | CRWMS M&O 1999v |
| Pool Water Treatment | Pool Water Treatment and Cooling System | Collect and manage the site-generated radioactive wastes generated in the process of pool water treatment. | CRWMS M&O 1999w |
| Site Fire Protection System | Site Fire Protection System | Protect QL-1 SSCs from the effects of fire. | CRWMS M&O 1999x |
| Solid LLW System | Site Generated Radiological Waste Handing System | Collect and manage site-generated radioactive wastes generated in the operation of MGR facilities. | CRWMS M&O 1999u |
| WHB Confinement Area Ventilation System | WHB Ventilation System | Mitigate the consequences of a facility DBE. | CRWMS M&O 1999y |
| Waste Package/Disposal Container Weld Preparation and Opening System | Waste Package Remediation System | Collect and manage radiologically contaminated metal chips generated during lid removal operations. | CRWMS M&O 1999q |
| Waste Package Horizontal Lifting System | Disposal Container Handling System | Maintain structural integrity in the event of a DBE, e.g., seismic event. | CRWMS M&O 1999p |
| WTB Confinement Area Ventilation System | WTB Ventilation System | Collect and manage site-generated radioactive wastes generated in the operation of MGR facilities. | CRWMS M&O 1999aa |
| WTB System | WTB System | Collect and manage site-generated radioactive wastes generated in the operation of MGR facilities. | CRWMS M&O 1999z |
| Wet Assembly Transfer Machine | Assembly Transfer System | Maintain structural integrity in the event of a DBE, e.g., seismic event. | CRWMS M&O 1999b |

Table 4-3. Quality Level 3 Important to Safety Items

| SSC | MGR System | Important to Safety Function | Reference |
|---|---|---|---|
| Area Radiation Monitoring System | Site Radiological Monitoring System | Provides an alarm to warn of significant increases in MGR radiation levels. Minimize onsite worker dose as a result of normal operations and Category 1 DBEs, including planned recovery operations. | CRWMS M&O 1999ab |
| Continuous Air Monitoring System | Site Radiological Monitoring System | Provide an alarm to warn of significant increases in MGR concentrations of airborne radioactive materials. Minimize onsite worker dose as a result of normal operations and Category 1 DBEs, including planned recovery operations. | CRWMS M&O 1999ab |
| Performance Confirmation Emplacement Drift Monitoring System | Performance Confirmation Emplacement Drift Monitoring System | Monitor variables to verify that operating conditions are within technical specifications. Also, functions as part of the radiological monitoring system required to assess radionuclide dispersion following a DBE. | CRWMS M&O 1999ac |
| Inspection Gantry | Performance Confirmation Emplacement Drift Monitoring System | Monitor variables to verify that operating conditions are within technical specifications. Also, assists in determining the cause and/or consequences of DBEs during accident investigations. | CRWMS M&O 1999ac |
| Waste Package Monitoring System | Performance Confirmation Emplacement Drift Monitoring System | Monitor variables to verify that operating conditions are within technical specifications; obtain and transfer waste package visual, thermal, and radiological monitoring data; function as part of the monitoring systems required to assess radionuclide release or dispersion following a DBE. | CRWMS M&O 1999ac |
| Meteorological Monitoring System | Surface Environmental Monitoring System | Monitor variables to verify that operating conditions are within technical specifications. Also, functions as part of the radiological monitoring system required to assess radionuclide dispersion following a DBE. | CRWMS M&O 1999ad |
| Performance Confirmation Data Acquisition/Monitoring System | Performance Confirmation Data Acquisition/Monitoring System | Monitor variables to verify that operating conditions are within technical specifications. | CRWMS M&O 1999ae |
| Performance Confirmation Waste Isolation Verification/Validation System | Performance Confirmation Waste Isolation Verification/Validation System | Monitor variables to verify that operating conditions are within technical specifications. | CRWMS M&O 1999af |
| Pool Water Leak Detection | Pool Water Treatment and Cooling System | Monitor variables to verify that operating conditions are within technical specifications. | CRWMS M&O 1999w |
| Pool Water Level Control | Pool Water Treatment and Cooling System | Monitor variables to verify that operating conditions are within technical specifications. | CRWMS M&O 1999w |
| Pool Water Makeup | Pool Water Treatment and Cooling System | Monitor variables to verify that operating conditions are within technical specifications. | CRWMS M&O 1999w |

Table 4-3. Quality Level 3 Important to Safety Items (Continued)

| SSC | MGR System | Important to Safety Function | Reference |
|---|---|---|---|
| Radiological Emergency Response System | Emergency Response System | Monitor area radiological conditions and support the management of emergency response to adverse radiological conditions. Minimize onsite worker dose as a result of normal operations and Category 1 DBEs, including planned recovery operations. | CRWMS M&O 1999ag |
| Radiological Monitoring System | WTB System | Provide an alarm to warn of significant increases in MGR concentrations of airborne radioactive materials. Minimize onsite worker dose as a result of normal operations and Category 1 DBEs, including planned recovery operations. | CRWMS M&O 1999z |
| Remote Manipulators | Assembly Transfer System<br>Carrier/Cask Handling System<br>Canister Transfer System | Permit remote operations and increased distance between facility operators and radiation source. Minimize onsite worker dose as a result of normal operations and Category 1 DBEs, including planned recovery operations. | CRWMS M&O 1999b<br>CRWMS M&O 1999n<br>CRWMS M&O 1999j |
| Sample Collection System | Surface Environmental Monitoring System | Monitor variables to verify that operating conditions are within technical specifications. Also, functions as part of the radiological monitoring system required to assess radionuclide dispersion following a DBE. | CRWMS M&O 1999ad |
| Seismic Monitoring System | Surface Environmental Monitoring System | Monitor variables to verify that operating conditions are within technical specifications. | CRWMS M&O 1999ad |
| WHB Exhaust Stack Radiation Monitor | Site Radiological Monitoring System | Provide alarm to warn of significant increases in MGR concentrations of airborne radioactive materials. Monitor variables to verify that operating conditions are within technical specifications. Assess radionuclide dispersion following a DBE. Maintain levels of radioactive material in effluent to unrestricted areas ALARA during normal operations. | CRWMS M&O 1999ab |
| WTB Exhaust Stack Radiation Monitor | Site Radiological Monitoring System | Provide alarm to warn of significant increases in MGR concentrations of airborne radioactive materials. Monitor variables to verify that operating conditions are within technical specifications. Assess radionuclide dispersion following a DBE. Maintain levels of radioactive material in effluent to unrestricted areas ALARA during normal operations. | CRWMS M&O 1999ab |

## 4.5 UTILITY AND AUXILIARY SYSTEMS

Utility and auxiliary systems are those conventional quality systems required to support operation of the MGR. These conventional quality systems do not perform an important to safety function. MGR utility and auxiliary systems are as follows:

- Water distribution and collection systems
- Electrical systems
- Compressed air and gas systems
- Non-nuclear HVAC systems
- Instrument and control systems
- Sanitary waste collection and treatment systems
- Site transportation systems
- Subsurface fire protection system
- Worker health and industrial safety systems
- Administrative systems

## 4.6 REFERENCES

### 4.6.1 Documents Cited

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# 5. HAZARD AND DESIGN BASIS EVENT ANALYSIS

Proposed rule 10 CFR Part 963 (64 FR 67086) provides the requirements for a preclosure suitability evaluation for the Yucca Mountain site. Specifically, 10 CFR 963.13(b)(3) requires that the preclosure safety evaluation consider a preliminary description of potential hazards, event sequences, and their consequences.

This section addresses the above criterion and provides a comprehensive assessment of facility hazards, event sequences and radiological consequences. In addition to the hazards analysis and DBE analysis discussed in this section, criticality safety is addressed in Section 6, radiation protection is addressed in Section 7, and fire safety is addressed in Section 8. Together, these items encompass many of the requirements of an Integrated Safety Analysis, as defined in Section 112 of *Interim Guidance Pending Issuance of New U. S. Nuclear Regulatory Commission (NRC) Regulations for Yucca Mountain, Nevada* (Dyer 1999).

Section 112 of Dyer (1999) provides specific requirements for the identification and evaluation of potential hazards and the development and selection of DBEs. This section provides the following information as required by Dyer (1999), Section 112:

- Section 112(b)–An identification and systematic analysis of naturally occurring and human-induced hazards at the geologic repository operations area, including a comprehensive identification of potential DBEs (addressed in Section 5.1)

- Section 112(d)–The technical basis for either inclusion or exclusion of specific, naturally occurring and human-induced hazards in the safety analysis (addressed in Section 5.2)

- Section 112(e)–An analysis of the performance of the major design SSCs, both surface and subsurface, to identify those that are important to safety, including identification and description of controls that are relied on to limit or prevent potential DBEs or mitigate their consequences, and including identification of measures taken to ensure the availability of identified safety systems (addressed in Section 5.3).

Section 5.4 of this report discusses beyond DBEs to provide a perspective of the residual risk associated with the operation of the MGR.

## 5.1 HAZARD ANALYSIS

Hazards analyses were performed to identify and document the internal and external hazards having the potential to initiate radiological event sequences associated with preclosure operations of the MGR. Internal hazards are those hazards presented by operation of the facility and associated processes. External hazards involve natural phenomena and external man-made hazards such as those posed by aircraft and nearby military/industrial facilities. The methodology used in these hazards analyses provides a systematic means to identify facility hazards and associated event sequences that may result in radiological consequences to the public and facility worker during the MGR preclosure period.

The MGR internal and external hazards analyses are documented in *Monitored Geologic Repository Internal Hazards Analysis* (CRWMS M&O 2000c) and *MGR External Events*

*Hazards Analysis* (CRWMS M&O 2000b), respectively. Section 5.1.1 describes the hazards analysis methodology and Section 5.1.2 provides a summary of the hazards analyses results. As the MGR design progresses, these hazards analyses will be reviewed and modified if necessary to ensure no new hazards are introduced and that previously evaluated hazards have not increased in severity.

### 5.1.1 Hazard Analysis Methodology

The internal and external hazards analyses are performed utilizing hazard analysis methodologies described in the *System Safety Analysis Handbook* (System Safety Society 1997) and *Guidelines for Hazard Evaluation Procedures: With Worked Examples* (American Institute of Chemical Engineers 1992). The process steps include (1) defining/describing the MGR site and facilities, (2) developing a generic events checklists, and (3) determining the applicability of the generic events to the MGR. A description of each process step is provided in the following sections.

#### 5.1.1.1 Define/Describe MGR Site and Facilities

To facilitate identification of MGR hazards, the MGR site and facilities are initially defined. For the internal hazards analysis, the MGR facilities are divided into functional areas. These functional areas are defined by a specific function and/or physical boundaries of the facility. Following the definition of functional areas, facility design configuration and operations within those areas are established and documented prior to hazard identification activities. MGR functional areas defined for the internal hazards analysis are as follows:

- Waste Receipt and Carrier/Cask Transport
- Carrier/Cask Preparation
- Waste Handling - Carrier Bay
- Waste Handling - Canister Transfer
- Waste Handling - Assembly Transfer
- Waste Handling - Disposal Container Handling and Waste Package Remediation
- Subsurface Transport, Emplacement, and Monitoring
- Site-Generated Waste Treatment - Liquid LLW
- Site-Generated Waste Treatment - Solid LLW

Functional area operations and design configurations are based upon the MGR facility description provided in Section 4 of this document.

To facilitate the identification of external hazards, the MGR site is initially described. This includes a description of the MGR site, location of facilities within the site and its proximity to the public and other facilities that may impact the MGR. A description of site meteorology, hydrology, geology is also included to the extent needed for identification of applicable natural phenomena. A description of the MGR site is provided in Section 3 of this document.

#### 5.1.1.2 Develop Generic Events Checklist

Once the MGR site, facilities and operations are defined, a list of generic internal and external events is developed that, if determined to be applicable, could result in radiological consequences to the public or the facility worker. This generic list attempts to identify

potentially hazardous events and is not facility specific. The intent is to provide the most comprehensive list to ensure thorough treatment of possible initiators of radiological event sequences. The generic lists developed for the internal and external hazard analyses are based upon hazard evaluation techniques described in *System Safety Analysis Handbook* (System Safety Society 1997) and *Guidelines for Hazard Evaluation Procedures: With Worked Examples* (American Institute of Chemical Engineers 1992). Tables 5-1 and 5-2 provide the internal and external generic events, respectively.

Table 5-1. Generic Internal Events

| Internal Event |
|---|
| Collision/Crushing |
| Chemical Contamination/Flooding |
| Explosion/Implosion |
| Fire |
| Radiation/Magnetic/Electrical/Fissile |
| Thermal |

Table 5-2. Generic External Events

| External Events | | |
|---|---|---|
| Aircraft crash | High river stage | Seismic activity, uplifting (tectonic) |
| Avalanche | Hurricane | Seismic activity, earthquake |
| Coastal erosion | Inadvertent future intrusions (man-made) | Seismic activity, surface fault displacement |
| Dam failure | Industrial activity induced accident | Seismic activity, subsurface fault displacement |
| Debris avalanching | Intentional future intrusions (man-made) | Static Fracturing |
| Denudation | Landslides | Stream Erosion |
| Dissolution | Lightning | Subsidence |
| Epeirogenic displacement | Loss of offsite/onsite power | Tornado |
| Erosion | Low lake level | Tsunami |
| Extreme wind | Low river level | Undetected past intrusions (man-made) |
| Extreme weather fluctuations | Meteorite impact | Undetected Geologic features |
| Range fire | Military activity induced accident | Undetected Geologic processes |
| Flooding (storm, river diversion) | Orogenic Diastrophism | Volcanic Eruption |
| Fungus, bacteria, and algae | Pipeline accident | Volcanism, intrusive magmatic activity |
| Glacial erosion | Rainstorm | Volcanism, ashflow (extrusive magmatic activity) |
| Glaciation | Sandstorm | Volcanism, ashfall |
| High Lake Level | Sedimentation | Waves (aquatic) |
| High Tide | Seiche | |

### 5.1.1.3 Determine Applicability of Generic Events to MGR

This portion of the analysis reviews the MGR facility design/operations and the MGR site against specified criteria, to determine the applicability of generic initiating events that could potentially result in radiological consequences.

Criteria are developed for the internal and external generic initiating events to support the applicability determination. If the criteria are satisfied, the generic event has the potential for radiological consequence and specific preliminary events are then identified. In addition, a general review of previously performed safety evaluations of repository operations was conducted to determine event sequences that may be applicable to the MGR. These evaluations included the following:

- Preliminary Worst–Case Analysis to Support the Conceptual Design of a Potential Repository in Tuff (Jackson et al. 1984)
- Site Characterization Plan Conceptual Design Report–Volume 4 Appendices F - O (MacDougall et al. 1987)
- Yucca Mountain Site Characterization Project Identification of Structures, Systems, and Components Important to Safety at the Potential Repository at Yucca Mountain (Hartman and Miller 1991)
- Preclosure Radiological Safety Analysis for Accident Conditions of the Potential Yucca Mountain Repository–Underground Facilities (Ma et al. 1992)
- Preclosure Radiological Safety Evaluation–Exploratory Studies Facility (Schelling and Smith 1993).

#### 5.1.1.3.1 Internal Events Applicability Criteria

The criteria used to determine the applicability of internal hazards as initiators of event sequences are listed below for each event category. Applicability to a functional area of design is determined by a positive (yes) response to all questions within a given hazard category or subcategory as appropriate.

A. Collision/Crushing

1. Is kinetic or potential energy present?
2. Can the kinetic or potential energy be released in an unplanned way?
3. Can the release of kinetic or potential energy interact with the waste form?

B. Chemical Contamination/Flooding

1. Category 1, Reactions

   a. Are corrosive/reactive chemicals or materials present?
   b. Can these chemicals or materials be released?

c. Can the chemicals or materials interact with the waste form?

2. Category 2, Off-Gassing

   a. Are volatile/condensable materials present?
   b. Can these materials be released?
   c. Can these materials interact with the waste form?

3. Category 3, Venting

   a. Is there potential for venting materials in the area?
   b. Can the materials interact with the waste form?

4. Category 4, Debris/Leaks

   a. Is there potential for debris or leaks in the area?
   b. Can the debris or fluids interact with the waste form?

5. Category 5, Flooding

   a. Are sources of water present in the area?
   b. Is there a potential to release the water?
   c. Can the released water interact with the waste form with potential for criticality?

C. Explosion/Implosion

1. Are pressure, electrical, chemical, or mechanical energy present?
2. Can an event occur that results in an explosion or implosion energy release?
3. Can the released energy impact the waste form directly?

D. Fire

1. Are fuel, oxidizers, and ignition sources present?
2. Is there sufficient fuel and oxidizer to sustain fire?
3. Can fire interact with the waste form?

E. Radiation/Magnetic/Electrical/Fissile

1. Are radiation/magnetic/electrical energy sources present external to the waste form? Is fissile material present?

2. Is a mechanism present to release radioactive/magnetic/electrical energy?

3. Can the release of radiation/magnetic/electrical energy interact with the waste form? Can fissile material be arranged in a such a manner as to result in criticality?

F. Thermal

1. Are external heat energy sources present?
2. Can heat energy be released?
3. Can the heat energy affect the waste form?

#### 5.1.1.3.2 External Events Applicability Criteria

The criteria used to determine external events applicability to the MGR are listed below. The external event is considered a potential initiator of a DBE if all of the following are determined to be true:

A. The potential exists and is applicable to the Yucca Mountain site.

B. The rate of the process is sufficient to affect the 100-year operational period. (Example: Is erosion expected to occur at the MGR during the 100-year operational phase?)

C. The consequence of the process is significant enough to affect the 100-year operational period. (Example: Can the consequences of erosion lead to a radiological release at the MGR during the 100-year operational phase?)

D. The event frequency $\geq$ 1E-6 events per year. (Example: Is any event associated with erosion expected to occur at a rate greater than or equal to once in a million years?)

If all of the above statements are true for any external event, then the event is considered applicable to the MGR. If, for any external event, any one of the above statements is false then the event is not considered applicable to the MGR. If any statement is indeterminate (its validity cannot be determined at this time) then the statement is treated as equivalent to true (cannot be screened out at that point).

### 5.1.2 Hazard Analysis Results

Applying the methods described in Section 5.1.1, the following internal and external events are determined to be applicable to the MGR. As the MGR design progresses, existing hazards analyses will be used as a basis for future hazards analysis performed to identify changes in the hazards posed by the facility and the introduction of new hazards. The results of this analysis are used as input to MGR DBE selection process.

#### 5.1.2.1 Internal Events

#### 5.1.2.1.1 Waste Receipt and Carrier/Cask Transport

A. Collision/Crushing: cask collision, railcar derailment involving transportation cask, overturning of truck trailer involving transportation cask.

B. Fire, Thermal: diesel fuel fire.

C. Radiation: radiation exposure of facility worker.

D. Fissile: criticality associated with cask collision, railcar derailment, or overturned truck trailer and rearrangement of cask internals.

#### 5.1.2.1.2 Carrier/Cask Preparation

A. Collision/Crushing: handling equipment drops on transportation cask, cask collision
B. Fire, Thermal: diesel fuel fire.
C. Radiation: radiation exposure of facility worker.
D. Fissile: criticality associated with cask collision and rearrangement of cask internals.

#### 5.1.2.1.3 Waste Handling–Carrier Bay

A. Collision/Crushing: transportation cask drop, transportation cask slapdown (tip over), cask collision, isolation door closes on transportation cask, handling equipment drops on transportation cask.

B. Fire, Thermal: diesel fuel fire.

C. Radiation: radiation exposure of facility worker.

D. Fissile: criticality associated with cask collision/drop and rearrangement of cask internals.

#### 5.1.2.1.4 Waste Handling–Canister Transfer

A. Collision/Crushing: transportation cask slapdown, disposal container slapdown, canister drop, canister slapdown, canister collision, canister drops onto disposal container, canister drop on sharp object, canister drop onto another canister at small canister staging rack, shield door closes on transportation cask, shield door closes on disposal container, handling equipment drops on transportation cask, canister or disposal container.

B. Radiation: radiation exposure of facility worker.

C. Fissile: criticality associated with small canister staging rack, criticality associated with collision or drop of cask/canister and rearrangement of container internals.

#### 5.1.2.1.5 Waste Handling–Assembly Transfer

A. Collision/Crushing: transportation cask drop, transportation cask slapdown, cask collision, spent fuel assembly (SFA) drop onto pool floor, SFA drop onto SFA staging rack, SFA drop onto assembly cell floor, SFA drop onto assembly dryer, SFA drop onto disposal container, SFA collision, loaded SNF assembly basket drop onto pool floor, loaded SNF assembly basket drop onto SFA staging rack, loaded SNF assembly basket drop onto assembly cell floor, loaded SNF assembly basket drop onto assembly dryer, loaded SNF assembly basket collision, uncontrolled descent of loaded incline basket transfer cart.

B. Flooding: uncontrolled pool water draindown/fill.

C. Fire, Thermal: SNF overheating in an assembly transfer basket or dryer.

D. Radiation: radiation exposure of facility worker.

E. Fissile: criticality associated with cask collision/drop and rearrangement of cask internals, criticality associated with SFA staging rack, criticality associated with misload of assembly dryer, criticality associated with misload of disposal container.

#### 5.1.2.1.6 Waste Handling–Disposal Container Handling and Waste Package Remediation

A. Collision/Crushing: waste package drop, waste package slapdown, waste package drop onto sharp object, waste package collision, equipment drops onto waste package, disposal container drop, disposal container slapdown, disposal container drop onto sharp object, disposal container collision, handling equipment drops on disposal container.

B. Fire, Thermal: fuel damage by burn through during welding process, SNF overheating in a disposal container.

C. Radiation: radiation exposure of facility worker.

D. Fissile: criticality associated with the disposal container/waste package staging area, criticality associated with collision or drop of disposal container/waste package and rearrangement of container internals.

#### 5.1.2.1.7 Subsurface Transport, Emplacement, and Monitoring

A. Collision/Crushing: transporter derailment outdoors, transporter derailment in ramp or main drift, transporter collision with other stationary or moving equipment, waste package reusable rail car rolls out of transporter, runaway transporter, rockfall onto transporter, loaded emplacement gantry derailment, waste package drop from emplacement gantry, waste package/emplacement gantry collision with equipment or another waste package, rockfall onto waste package, steel set drop onto waste package, failure of isolation air locks due to rockfall, equipment collision, or other impacts as a result of development operations.

B. Flooding: flooding from water pipe break originating on development or emplacement sides.

C. Fire, Thermal: fire associated with waste package transporter/locomotive or development equipment.

D. Radiation: radiation exposure of facility worker, or early waste package failure and resultant release of radioactive material.

E. Fissile: criticality associated with collision or drop of waste package and rearrangement of package internals.

#### 5.1.2.1.8 Site-Generated Waste Treatment Liquid Low–Level Radioactive Waste

A. Collision/Crushing: handling equipment drops on liquid LLW.
B. Flooding: uncontrolled release of liquid LLW.
C. Radiation: radiation exposure of facility worker.

#### 5.1.2.1.9 Site-Generated Waste Treatment–Solid Low-Level Radioactive Waste

A. Collision/Crushing: solid LLW drop, handling equipment drops on solid LLW.
B. Fire, Thermal: fire involving combustible LLW.
C. Radiation: radiation exposure of facility worker.

### 5.1.2.2 External Events

The 53 external events listed in Table 5-2 were subjected to the screening process described in Section 5.1.1.3.2, and to supplemental analyses described in Section 3.4, to reduce the list to 20 candidate events. The list was reduced to the following 12 by grouping events. The following external events are determined to be applicable to the MGR as potential initiators of event sequences leading to release of radioactivity:

- Debris Avalanching
- Extreme Wind (including sandstorms)
- Flooding (including rainstorm and river diversion)
- Industrial-Activity-Induced Accident
- Landslide
- Lightning
- Loss of Offsite/Onsite Power
- Military-Activity-Induced Accident
- Seismic Activity, Earthquake
- Seismic Activity, Surface Fault Displacement
- Seismic Activity, Subsurface Fault Displacement (including subsidence)
- Tornado (winds, missiles)

The list is further reduced by screening analyses, or by combining events as described below.

**Industrial/Military Activities**–An analysis entitled Industrial/Military Activity-Initiated Accident Screening Analysis (CRWMS M&O 1999m) was performed to establish whether this external event could be screened from further consideration. The study concluded that because of the remote location of the YMP site, the near-by-industrial operations, transportation routes, and operations on the NTS and Nellis Air Force Range were found to have no events that would impact the MGR. The remote location of the MGR (5+ miles from NTS facilities, 13+ miles from near-by industrial operations and US 95, and 25+ miles from Nellis Air Force Range facilities) is the major reason none of the postulated events (e.g., explosions, fires, chemical releases) impact the MGR.

**Lightning**–Potential event sequences initiated by lightning are either covered under other events or have been screened out. For example, indirect effects of lightning strikes include loss of offsite/onsite power and fires, both of which will be addressed in the design bases. Potential direct strikes of lightning on waste forms within the confines of the waste handling building are precluded by the lightning protection system. A direct strike on the waste package during transport to the subsurface has been shown to be below the threshold for credible events (CRWMS M&O 1997b).

**Debris Avalanching and Landslide**–The site of the surface facilities and the North Portal will be stabilized against such events. For purposes of the preclosure safety analysis, these events are grouped with flooding.

**Extreme Wind and Tornadoes**–Tornadoes pose two kinds of hazards: wind loading; and missile generation. For purposes of the preclosure safety analysis, extreme wind is grouped with tornado wind and tornado missile is broken out as a separate hazard.

As described in Section 3.4, other potential external events will be addressed in the MGR design bases for reasons beyond preclosure safety analysis. Inadvertent/Intentional Human Intrusion will be addressed in the MGR Safeguards and Security plan. External Fires (range or other) will be addressed in the Fire Hazards Analysis (see Section 8) and appropriate prevention and mitigation controls will be provided in the design.

The grouping discussed above reduces the list of external events to the six categories shown in Table 5-4.

## 5.2 DESIGN BASIS EVENT SELECTION

DBE analysis involves the detailed frequency and consequence analysis of event sequences that have the potential to result in a radiological release. Frequency analysis is used to categorize event sequences as Category 1 DBEs, Category 2 DBEs, or Beyond Design Basis Events (BDBEs), as defined below:

Table 5-3. DBE Frequency Categories

| DBE Category | Frequency of Occurrence | Definition (Dyer 1999, Section 2) |
|---|---|---|
| Category 1 DBE | Greater than, or equal to, once every 100 years | "Those natural events and human-induced event sequences that are expected to occur one or more times before permanent closure of the geologic repository operations area." |
| Category 2 DBE | Less than once every 100 years, but greater than, or equal to, once every 1 million years | "(a) Other human-induced event sequences that have at least one chance in 10,000 of occurring before permanent closure of the geologic repository, and (b) appropriate consideration of natural events (phenomena) that have been historically reported for the site and the geologic setting." |
| Beyond Design Basis Event | Less than once every 1 million years | Not Applicable |

The frequency ranges shown in Table 5-3 for each DBE category correlate with the probability-based definitions from Section 2 of Dyer (1999), assuming a 100-year preclosure period. A

more thorough discussion of the 100-year preclosure period and the impact of assuming a longer or shorter preclosure period is provided in Section 4.4.1.2.1.

Consequence analysis is used to identify the bounding Category 2 event sequences and to allow for comparison of Category 1 and Category 2 DBEs with regulatory dose limits. Bounding event sequences are those that result in the largest radiological consequences for a group of physically similar event sequences. All Category 1 event sequences that result in a radiological consequence are analyzed, as opposed to just the bounding Category 2 event sequences, because Category 1 event consequences are annualized, summed, and compared to an annual dose limit. Dose limits for Category 2 DBEs, by contrast, are on a per event basis.

The overall process for implementing an Integrated Safety Analysis for preclosure repository operations is graphically illustrated in Figure 5-1. The DBE selection activity results from internal and external event analyses, as shown in Figure 5-1.

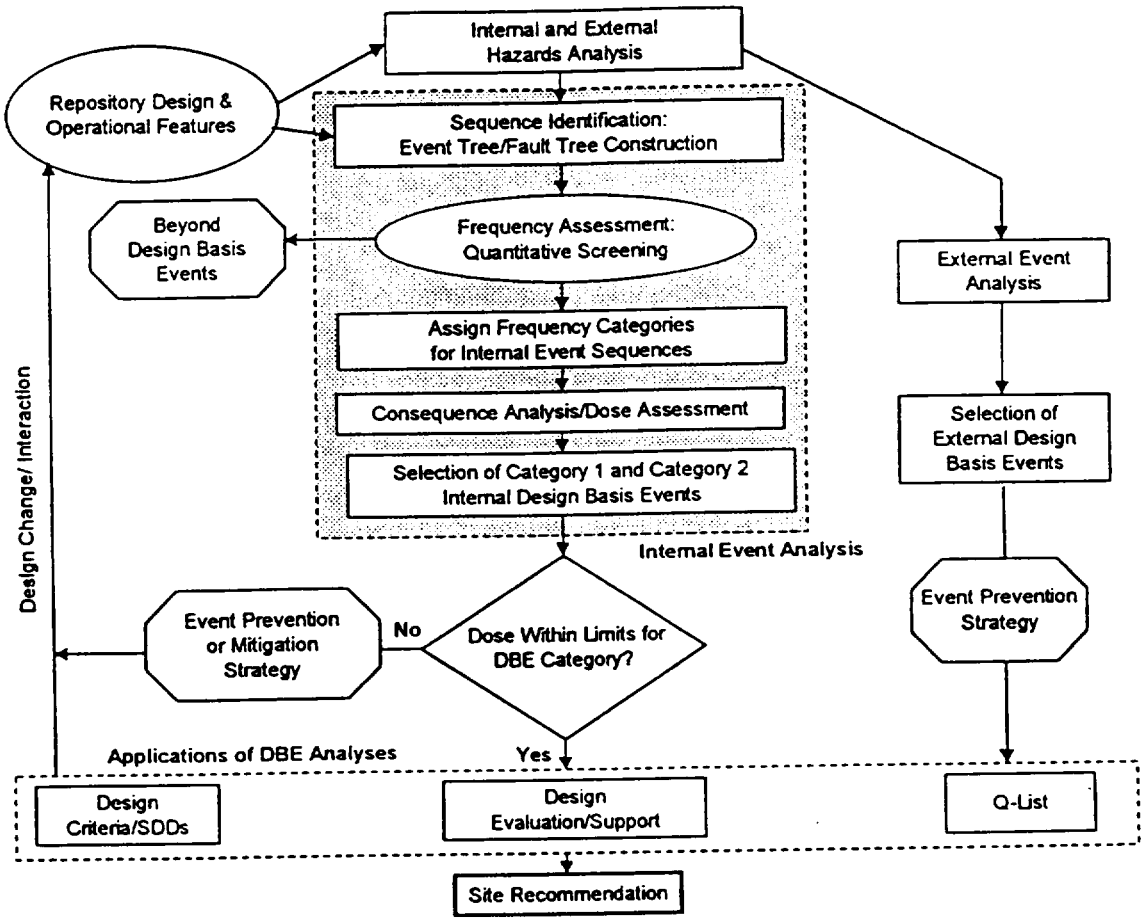


Figure 5-1. Integrated Safety Analysis Process (Preclosure)

The preclosure safety strategy for external events and natural phenomena is to prevent a radiological release by designing the SSCs important to safety to withstand the bounding criteria associated with external DBEs (e.g., maximum wind loadings resulting from the design basis tornado). For internal event sequences, the preclosure safety strategy is either prevention or

mitigation of event sequences that result in a release, depending on the waste form involved (see Section 4.4.1.2.2).

The process described above and illustrated in Figure 5-1 is iterative in nature and results in a safety analysis that is integrally tied to the facility design. Consequently, the preclosure safety analysis and DBEs will continue to evolve in lock step with design.

### 5.2.1 External Design Basis Events

The external events and natural phenomena in Table 5-4 are credible initiating events that could potentially lead to a radiological release. The general strategy for external DBEs, which is based on deterministic NRC licensing precedence, is to design the SSCs important to safety to withstand these events so that no release scenarios are initiated by them and no loss of containment or confinement of radioactive material results. This strategy ensures that there are no credible radiological release scenarios associated with external DBEs and natural phenomena at the MGR, even though the initiating events are deemed credible.

Table 5-4. External DBEs and Natural Phenomena

| Design Basis Event/ Natural Phenomena | Location | Initiating Event Frequency Category* | Discussion Section |
|---|---|---|---|
| Loss-of-Offsite Power | Surface and Subsurface Facilities | 1 | 5.2.1.1 |
| Earthquake – Vibratory Ground Motion | Surface and Subsurface Facilities | 1, 2** | 5.2.1.2 |
| Earthquake -Fault Displacement | Surface and Subsurface Facilities | 1, 2** | 5.2.1.3 |
| Flood | Surface and Subsurface Facilities | 2 | 5.2.1.4 |
| Tornado Missiles | Surface Facilities | 2 | 5.2.1.5 |
| Tornado Wind | Surface Facilities | 2 | 5.2.1.6 |

* For external events, the initiating event (e.g., earthquake) frequency is considered instead of the event sequence.

** Two design basis earthquakes are considered; Frequency-Category 1 with a recurrence frequency of $10^{-3}$ per year and Frequency-Category 2 with a recurrence frequency of $10^{-4}$ per year (YMP 1997).

#### 5.2.1.1 Loss-of-Offsite Power

Event Description: This event results in the total loss of external alternating current power, short term and long term, to the MGR. This event is postulated to occur as a result of an external event (e.g., lightning) or an internal event (e.g., fire or random equipment failure). A Loss-of-Offsite Power event will, at a minimum, temporarily halt the receiving and transferring of waste. Loss-of-Offsite Power at the MGR is assumed to occur one or more times during the preclosure operations, therefore, it is a Category 1 event.

Safety Strategy: The strategy for this event is to prevent credible release scenarios by design of the facility and, if necessary, invoking administrative controls. MGR SSCs important to safety

are designed to prevent load drops during a Loss-of-Offsite Power event. Emergency backup power sources and redundant offsite power lines/sources may be used to ensure continuous power is supplied to SSCs important to safety. The MGR design may also include features such as external lightning rods to protect against a lightning-initiated Loss-of-Offsite Power event.

#### 5.2.1.2 Earthquake - Vibratory Ground Motion

Event Description: This is a natural event involving trembling and shaking of the earth due to the shifting of tectonic plates. This event can potentially impact SSCs in the surface and subsurface facilities and lead to a radiological release. The possible consequences of this event include a collapse of structures, concrete cracking, loss-of-offsite power, ground displacement, and subsurface rockfall.

*Preclosure Seismic Design Methodology for a Geologic Repository at Yucca Mountain* (YMP 1997) defines two design basis earthquake categories: Frequency-Category-1 (FC-1) with a recurrence frequency of 1E-3 per year and Frequency-Category-2 (FC-2) with a recurrence frequency of 1E-4 per year.

Safety Strategy: The safety strategy for the surface facilities is to design the SSCs important to safety to withstand the effects of either the FC-1 or FC-2 design basis earthquake, depending on the potential consequences of a failure.

#### 5.2.1.3 Earthquake - Fault Displacement

Event Description: A fault displacement is a fracture or a zone of fractures along which there is potential for displacement of the sides relative to one another, parallel to the fracture. Because several faults intersect the designated repository area, this event is applicable to Yucca Mountain.

Safety Strategy: Fault avoidance is the primary strategy to prevent fault displacement hazards (YMP 1997). Faults that are capable of impacting the design of an SSC are defined as Type I faults – those "*subject to displacement and of sufficient length and located such that they may affect repository design and/or performance*" of SSCs important to safety, containment, or waste isolation (CRWMS M&O 1998b).

#### 5.2.1.4 Flood

Event Description: An external flood may be initiated by intense precipitation, runoff, or a landslide. The Probable Maximum Flood (PMF) is defined as the hypothetical flood (peak discharge, volume, and hydrograph shape) that is considered to be the most severe reasonably possible flood, based on a probable maximum precipitation and other hydrologic factors favorable for maximum flood runoff, such as sequential storms and snowmelt (ANSI/ ANS-2.8-1992). A 100-year flood is defined as the magnitude of peak discharge at any point on a river or drainage channel that can be expected to occur or be exceeded, on average, once in 100 years (DOE 1999). Inasmuch as the Yucca Mountain area is located inland and has no significant surface-water bodies or water-controlled structures located near the site, there is no potential for events such as surges, seiches, tsunamis, dam failures, or ice jams that could affect the site, nor is there any potential for future dam development. No evidence for past flooding induced by landslides in the vicinity of the site has been reported (YMP 1995). However, floods

can potentially produce heavy loads on buildings, transportation casks, waste packages, or transporters containing waste packages. The consequences of a design basis flood are expected to bound the rainstorm, landslide, and debris avalanche events identified in the *MGR External Events Hazards Analysis* (CRWMS M&O 2000b).

Safety Strategy: The primary safety strategy for the design basis flood event is to preclude a radiological release by placement, layout, and design of the surface facilities. Taking into account the effects of sediment and debris transported during flood events, a series of worst-case flood studies were completed. The North Portal site is adjacent to the Midway Valley Wash. The maximum depth of water in this wash was estimated to be 9-12 feet (CRWMS M&O 1998a) during a PMF with consideration given to the presence of sediment and debris. Although it was determined that a portion of the North Portal Pad is in the flood-prone area, the flood inundation lines will stop at the boundary of the pad because the pad intersects the maximum flood levels at a higher elevation, which will cause the water to flow around the pad. Because the water profiles will rise to account for restrictions of flow caused by the pad being superimposed on the inundation zone, the Waste Handling Building (WHB) and Waste Treatment Building (WTB) will be set approximately 1.5 feet (CRWMS M&O 1998a) above the maximum elevation flood elevation. The pad for the Balance of Plant (BOP) area will be set three feet (CRWMS M&O 1998a) below the floor elevation of the WHB to account for its dock height at the southeast corner. The drainage of the radiological control area will protect this pad from a PMF. Pad protection will be accomplished by providing an underground storm drainage collection system to contain the runoff from this area and prevent spillage over the fill slopes and the BOP area. The drainage design for the BOP area will protect this pad from the 100-year flood. Two existing open channels constructed for the Exploratory Studies Facility will protect the North Portal from the PMF (CRWMS M&O 1998a).

Thus, the WHB, WTB, and CPB are designed to withstand the PMF. Other surface facilities are designed to withstand the 100-year flood, based on standard industrial practice (CRWMS M&O 1998a, p. 8). In addition, the WHB, WTB, and CPB structures are designed to withstand the roof loads resulting from 24 hours of rainfall associated with the 100-year flood. Since the return period of the design basis flood is 100 years, the design basis flood is a borderline Category 2 initiating event.

For defense-in-depth, the following additional surface facility characteristics and/or design features may also be used for flood protection:

- "Hardened" foundations and structures
- Sandbags, flood doors, and bulkheads
- WHB cells located within interior wall
- Administrative controls to suspend operations during severe weather conditions.

#### 5.2.1.5 Tornado Missiles

Event Description: This event involves the impact of a tornado-generated missile on the MGR. The design basis tornado is deterministically classified as a Category 2 DBE.

Safety Strategy: The primary safety strategy is to preclude a radiological release by designing important to safety SSCs that are potentially vulnerable to a tornado missile to withstand the design basis tornado.

SSCs that are vulnerable to tornado missile impact are either protected from the missiles, designed to withstand a missile impact, or shown to not interact with a missile based on probabilistic analysis. The waste package transporter is designed not to allow penetration that could breach a waste package as a result of the impact of a tornado missile, the surface facility foundations and structures are designed to protect the wasteforms inside from a design basis tornado missile, and the WHB Ventilation System is designed to function following a design basis tornado missile impact.

Sections 3.5.1.4 and 3.5.2 of NUREG-0800 (NRC 1987) provide NRC guidance on missiles generated by natural phenomena and externally-generated missiles, respectively. Additional defense-in-depth safety features may include administrative controls to suspend operations in the event of a tornado warning or extreme weather conditions, hardened buildings, and the installation of underground utilities.

#### 5.2.1.6 Tornado Wind

Event Description: Tornado winds are high winds generated during a tornado. This event is associated only with the effects produced by these winds (i.e., pressure drop and wind loading). Extreme wind is defined as the 100-year wind with duration of 6 hours (CRWMS M&O 2000b, Section 6.3.3.10). The extreme wind, "fastest mile" wind, and "basic" wind criteria are bounded by the design basis tornado wind for Yucca Mountain (Ramsdell and Andrews 1986, p. 51). The consequences of this event are pressure loads on the surface facilities, waste package transporter, and transportation cask surfaces. The design basis tornado wind is classified as a Category 2 DBE based on data provided in *Tornado Climatology of the Contiguous United States* (Ramsdell and Andrews 1986, p. 51).

Safety Strategy: Important to safety SSCs that are potentially vulnerable to a tornado are designed to withstand the static loading and pressure drops associated with the design basis tornado. This strategy includes designing the WHB foundations and structures to withstand the design basis tornado and designing the WHB ventilation system to confine and filter particulates following a design basis tornado.

The following NRC documents related to design basis tornadoes are considered in the MGR design process:

- NRC Regulatory Guide 1.76, Design Basis Tornado for Nuclear Power Plants
- NRC Regulatory Guide 1.117, Tornado Design Classification
- NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition (NRC 1987), Sections 2.3.1 (Regional Climatology), 3.3.1 (Wind Loadings), and 3.3.2 (Tornado Loadings)

- NUREG/CR-4461, Tornado Climatology of the Contiguous United States (Ramsdell and Andrews 1986).

The design basis tornado wind for the Yucca Mountain region is 189 mph, with a 1E-6 probability of occurrence and a 90 percent strike probability confidence interval (Ramsdell and Andrews 1986). This wind speed bounds both the 100-year return period "Fastest Mile" wind (100-year, 1-minute gust) referenced in NUREG-0800 (NRC 1987) and the "Basic" wind (50-year, 3-second gust) calculated from methodology in ASCE 7-98.

As with the tornado-generated missile event, potential defense-in-depth safety features to protect against tornado winds may include administrative controls to suspend operations in the event of a tornado warning or extreme weather conditions, hardened buildings, and the installation of underground utilities.

In summary, the MGR SSCs deemed important to safety are either designed to withstand or protected from the bounding external events and natural phenomena as appropriate.

### 5.2.2 Internal Design Basis Event Sequences

Internal event sequences that could potentially occur at the MGR and lead to a radioactive release were selected and screened using the methodology described in Section 5.2.

Event sequence frequencies for postulated scenarios were calculated in *Design Basis Event Frequency and Dose Calculation for Site Recommendation* (CRWMS M&O 2000a). Frequencies were used to bin event sequences into either Category 1 or Category 2. Internal event sequences with a scenario frequency less than once per million years are considered to be "Beyond Design Basis Events" and screened out from further consideration. Figure 5-2 graphically illustrates the decision process for categorization of DBEs by frequency and consequence.

Radiological consequences for the bounding internal DBEs were calculated in CRWMS M&O (2000a). Bounding DBEs include those event sequences that result in the maximum radiological consequences to a member of the public at the preclosure controlled area boundary or to a worker onsite, for a group of similar event sequences. Collectively, the bounding DBEs establish constraints on the facility design to ensure that the SSCs important to safety will perform their intended function during a DBE, and that any radiological releases are within the dose limits specified by Section 111 of Dyer (1999).

Internal event sequences were screened into one of the following three groups based on their frequency of occurrence and potential to result in a radiological release:

1. **Internal Event Sequences with Potential Releases**–These event sequences could potentially result in a release of radioactivity and, therefore, are mitigated by the facility design. Internal event sequences with potential radiological releases are identified in Sections 5.3.2 (Category 1 event sequences) and 5.3.3 (Category 2 event sequences).

2. **Internal Event Sequences with No Release**–Internal event sequences in this category are not expected to result in a radiological release because they are prevented by design. In this case, design features function to either prevent the event sequence from occurring or to prevent a release, should the event sequence occur. Design features to prevent the event sequence can either physically prevent an initiating event from occurring (e.g., use of passive design features or process controls such as eliminating cask/canister lifts) or reduce the event sequence frequency below the credible cutoff frequency of 1E-6 per year (e.g., use of redundant design features and control systems). Design features that prevent a release are based on the premise that credible event sequences will occur and that affected SSCs must be designed to protect the wasteform from releasing radioactivity during such an event. Prime examples of this include the waste package design criteria, which establish design bases for the waste package to ensure that the waste package will not breach as a result of credible event sequences (i.e. waste package DBEs). These event sequences are identified in Section 5.3.4.

3. **Beyond Design Basis Events**–These event sequences are expected to have less than one chance in 10,000 of occurring before permanent closure of the repository, corresponding to a frequency of less than 1E-6 per year for a 100-year preclosure period. BDBEs are evaluated in Section 5.4.

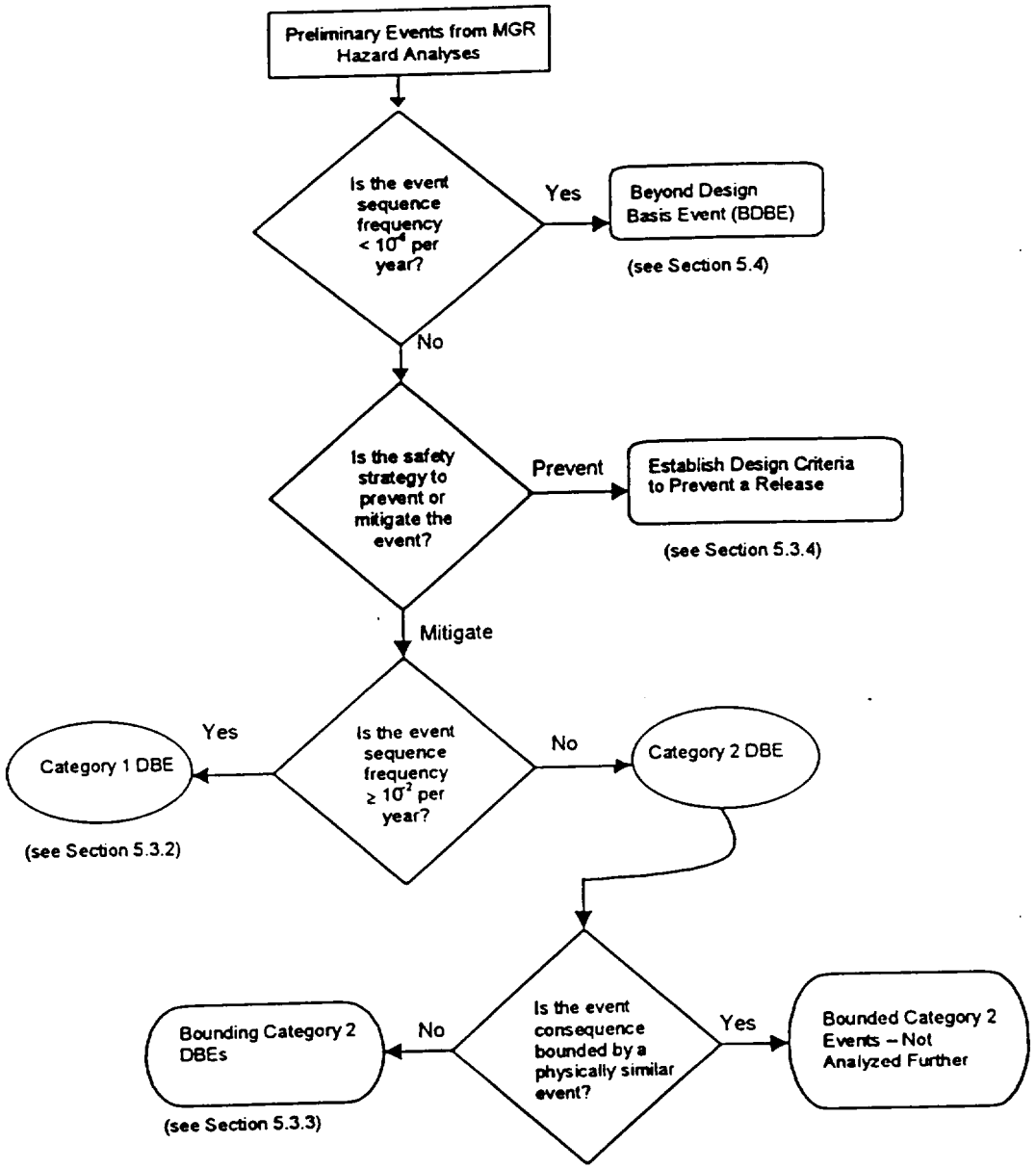


Figure 5-2. DBE Frequency Categorization Flowchart

## 5.3 DESIGN BASIS EVENT ANALYSIS

### 5.3.1 Scenario Development

Potential event sequences of interest begin with an initiating event and conclude with a radiological release. Initially, when postulating the event sequence, no credit is given to design features that could prevent or mitigate the event (i.e., the most severe consequences are evaluated). The scenario development process involves analysis of SSCs that can affect the progression of an accident sequence, including the effects of successful operation or failure of the Heating, Ventilation, and Air Conditioning (HVAC) system with High-Efficiency Particulate

Air (HEPA) filters, where appropriate. Performance of this step relies heavily on the analyst's experience and judgement as well as input from, and review by, the design organizations.

The result of the scenario development process is a series of event sequences that each have a specific frequency of occurrence (Category 1, Category 2, or BDBE). SSCs that are relied upon to prevent or mitigate an event sequence to below the dose limits set forth in Dyer (1999, Section 111) and/or 10 CFR Part 20 are considered important to safety and classified in accordance with Section 4.4. An example of a typical event tree used to determine the frequency and category of potential event sequences is shown in Figure 5-3.

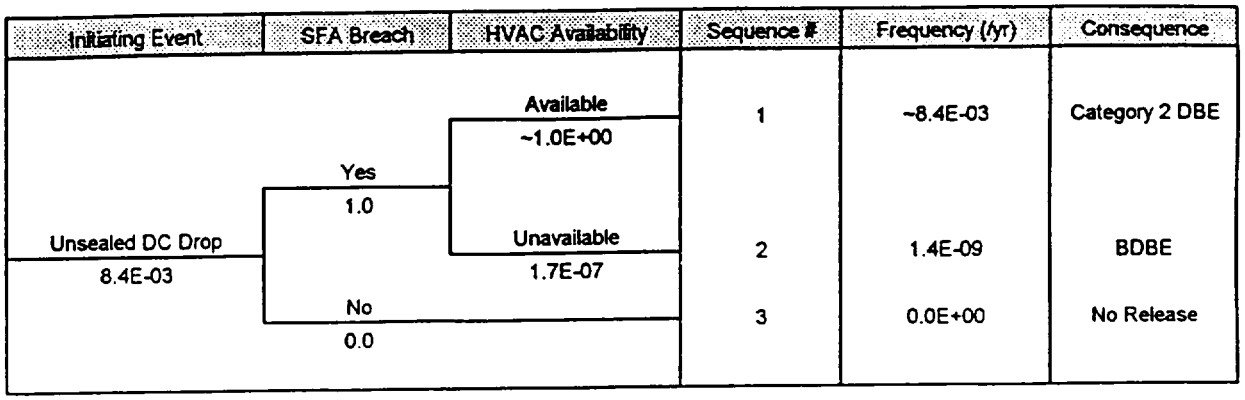


Figure 5-3. Sample Event Tree

In the sample event tree shown in Figure 5-3, the initiating event is an unsealed disposal container drop. The second event represents the conditional probability that a SFA will be breached following an unsealed disposal container drop. The last event in the event tree is the probability that the HVAC system will be available following the initiating event. In the event tree shown above, the three event sequences are labeled 1-3. Event sequence 1, a Category 2 event sequence, represents an unsealed disposal container drop that results in a SFA breach with the HVAC system available to mitigate radiological releases. Event sequence 2 represents a release scenario with HVAC unavailable and a frequency that is beyond design basis. Event sequence 3 represents an unsealed disposal container drop that does not breach the enclosed spent fuel assemblies and, consequently, does not result in a release. The frequency of this event sequence is zero because the probability that a spent fuel assembly does not breach, given an unsealed disposal container drop, is zero.

Credit is taken for the HVAC/HEPA system reliability to determine the frequency of event sequences 1 and 2. Due to the high reliability of the HVAC/HEPA system, event sequences involving failure of the HVAC/HEPA system are BDBEs (event sequence 2 is an example). Therefore, dose consequence limits do not apply to the unfiltered dose. However, because DBE Category 2 event sequences with HEPA filtration (e.g., event sequence 1) must satisfy DBE Category 2 limits, any components required to mitigate the dose to below Category 2 limits would be important to safety. Further, if unfiltered doses exceed the Category 2 limits, then system components relied upon to ensure that event sequences leading to an unfiltered release are beyond design basis would also be important to safety. The results of the scenario development process are described in the subsections that follow.

### 5.3.2 Category 1 Internal Event Sequences

Category 1 DBEs are those event sequences expected to occur one or more times before permanent closure of the geologic repository operations areas, per Section 2 of Dyer (1999). The Category 1 event sequences evaluated for the MGR (shown in Table 5-5) are internal event sequences that occur during handling of bare commercial SFAs or SFA baskets in the Assembly Transfer System (ATS). No Category 1 event sequences have been identified for the subsurface facilities. All Category 1 event sequences occur in confinement areas with HEPA filtration functional in the event sequences. The WHB ventilation system and confinement zones are described in *Surface Nuclear Facilities HVAC Analysis* (CRWMS M&O 1997a). The reliability of the HVAC system, used in event trees to calculate sequence frequencies, is based on the results of *Reliability Assessment of Waste Handling Building HVAC System* (CRWMS M&O 1999g).

#### 5.3.2.1 Sequences Involving Individual SFAs

Bare, unconfined SFAs are individually handled underwater, during transfer from the cask to the ATS basket staging rack, and in a dry environment during transfer from the ATS dryer to the disposal container.

While underwater, SFAs can potentially be dropped or impacted as a result of mechanical failure or control system failure of the wet assembly transfer machine, or as a result of operator error. These event sequences occur in the ATS pool area, which is a confinement area with HEPA filtration. Individual SFA event sequences that occur underwater are identified by sequence numbers 1-01 through 1-04 in Table 5-5.

During transfer from the ATS dryer to the disposal container, individual SFAs can potentially be dropped or impacted as a result of mechanical failure or control system failure of the dry assembly transfer machine, or as a result of operator error. These event sequences occur in the ATS cell, which is a confinement area with HEPA filtration. Individual SFA event sequences that are in the cell are identified by sequence numbers 1-12, 1-13, and 1-14 in Table 5-5.

#### 5.3.2.2 SFA Basket Event Sequences

SFA baskets are first handled underwater during transfer out of the basket staging rack. From there, SFA baskets containing a maximum of 4 Pressurized Water Reactor (PWR) SFAs or 8 Boiling Water Reactor (BWR) SFAs may be transferred and staged in the pool storage area to facilitate aging and blending, or loaded directly into the incline transfer cart. Baskets that are staged in the pool storage area have an additional movement from the storage pool to the incline transfer cart. Once loaded onto the incline transfer cart, SFA baskets are transported out of the pool and into the Assembly Drying Stations, where up to six SFA baskets may be loaded into each of the two assembly dryers. Both the ATS pool and cell are located in confinement areas with HEPA filtration.

SFA baskets can potentially be dropped or impacted in the pool during lifting out of the basket staging racks, during transport to the pool storage area, or during transport up the inclined transfer canal as a result of mechanical failures, control system failures, or operational error.

Event sequences that occur underwater involving SFA baskets are identified by sequence numbers 1-05 through 1-09 in Table 5-5.

In addition, SFA baskets can potentially be dropped or impacted onto the floor or into one of the assembly dryers as a result of mechanical failure or control system failure of the Dry Assembly transfer machine, or operational error. SFA basket sequences that occur in the cell are identified by sequence numbers 1-10 and 1-11 in Table 5-5.

### 5.3.2.3 Summary of Category 1 Internal Event Sequences

Table 5-5 identifies the Category 1 DBEs sequences that are expected to result in radiological releases.

Table 5-5. Category 1 Internal DBE Sequences

| Event Sequence # | Event Description | Location | Frequency (per Year) |
|---|---|---|---|
| 1-01 | SFA Drop Onto Another SFA in Cask | ATS Pool | 2E-1 |
| 1-02 | SFA Collision | ATS Pool | 4E-2 |
| 1-03 | SFA Drop Onto Empty Basket | ATS Pool | 4E-2 |
| 1-04 | SFA Drop Onto Another SFA in Basket Staging Rack (Lowering Into) | ATS Pool | 2E-1 |
| 1-05 | Basket Drop Onto Another Basket in Basket Staging Rack (Lifting Out) | ATS Pool | 4E-2 |
| 1-06 | Basket Drop Onto Another Basket in Pool (Transfer Into Pool Storage) | ATS Pool | 4E-2 |
| 1-07 | Basket Drop Onto Another Basket in Pool (Transfer Out of Pool Storage) | ATS Pool | 4E-2 |
| 1-08 | Basket Drop Onto Transfer Cart or Pool Floor | ATS Pool | 4E-2 |
| 1-09 | Basket Drop Back Into Pool | ATS Pool | 4E-2 |
| 1-10 | Basket Drop Onto ATS Cell Floor | ATS Cell | 4E-2 |
| 1-11 | Basket Drop Onto Another Basket in Dryer | ATS Cell | 4E-2 |
| 1-12 | SFA Drop Onto Another SFA in Dryer | ATS Cell | 2E-1 |
| 1-13 | SFA Drop Onto ATS Cell Floor | ATS Cell | 2E-1 |
| 1-14 | SFA Drop Onto Another SFA in Disposal Container | ATS Cell | 2E-1 |

The annual dose limit for Category 1 event sequences is 25 mrem Total Effective Dose Equivalent (TEDE) to any real member of the public located beyond the preclosure controlled area boundary (Dyer 1999, Section 111). The annual dose limit applies to both normal operational releases and individual Category 1 event releases. Normal operational releases include releases from the surface facilities and releases from the subsurface facility. The Category 1 dose assessment in CRWMS M&O (2000a) credits the ATS Pool for retaining all particulate radionuclides released underwater in the Category 1 pool events (Table 5-5, Events 1-01 through 1-09). The WHB HEPA filters are credited for removing airborne particulate radionuclides released during the Category 1 event sequences that occur in the ATS

Cell (Table 5-5, Events 1-10 through 1-14). The normal operational releases from the WHB are due to Krypton-85 and, therefore, unaffected by the presence of HEPA filters. Normal operational releases from the subsurface facilities are unfiltered as HEPA filters are not assumed to be present in the subsurface design. The dose calculation (CRWMS M&O 2000a) demonstrates that the annualized offsite dose to the public (i.e., sum of annual doses due to frequency weighting of all Category 1 event sequences, normal surface releases, and normal subsurface releases), 6E-2 mrem/year, is more than 400 times below the 25 mrem/year Category 1 DBE dose limit and 160 times below the 10 mrem/year ALARA limit (see Section 5.3.6.1).

### 5.3.3 Category 2 Internal Event Sequences

The bounding Category 2 internal event sequences that are expected to result in radiological releases are identified in Table 5-6 (CRWMS M&O 2000a).

Table 5-6. Category 2 Internal DBE Sequences

| Event Sequence # | Event Description | Location | Frequency (per Year) |
|---|---|---|---|
| 2-01 | SFA Basket Collision During Transfer* | ATS Pool | 7E-3 |
| 2-02 | Uncontrolled Descent of Incline Transfer Cart | ATS Pool | 7E-3 |
| 2-03 | Handling Equipment Drop onto SFA Basket in Pool** | ATS Pool | 2E-3 |
| 2-04 | Handling Equipment Drop onto SFA Basket in Cell** | ATS Cell | 7E-5 |
| 2-05 | Unsealed Disposal Container Collision | Disposal Container Handling Cell | 2E-3 |
| 2-06 | Unsealed Disposal Container Drop and Slapdown | Disposal Container Handling Cell | 8E-3 |
| 2-07 | Handling Equipment Drop onto Unsealed Disposal Container | Disposal Container Handling Cell | 1E-4 |
| 2-08 | Unsealed Shipping Cask Drop into Cask Preparation Pit | ATS Cask Preparation Pit | 9E-3 |
| 2-09 | Unsealed Shipping Cask Drop into Cask Unloading Pool | ATS Pool | 9E-3 |

* This event encompasses two individual basket collision events in CRWMS M&O (2000a), each with the same frequency and consequence.

** Event bounds the consequences of handling equipment drops onto a single SFA and of a handling equipment drop onto a SFA basket in the ATS Pool (CRWMS M&O 2000a).

The Category 2 dose assessment in CRWMS M&O (2000a) evaluated the dose consequences for each of these event sequences. Dose values for the Category 2 event that results in the largest offsite public dose are provided in Section 5.3.6.1. The bounding Category 2 DBEs from Table 5-6 are discussed in more detail in the subsections that follow.

#### 5.3.3.1 SFA Basket Collision During Transfer

Event Description: A SFA basket collides with a wall or other heavy object in the ATS pool causing a breach and subsequent release. This event could occur either during transfer from the assembly basket rack to the pool storage area or from the pool storage area to the incline transfer

cart. The pool water serves as a barrier to particulate release and, consequently, only the radioactive gases are released to Waste Handling Building environment.

Technical Strategy: The primary safety strategy is to confine particulate releases within the ATS pool by designing the pool system in accordance with ANSI/ANS-57.7-1988.

#### 5.3.3.2 Uncontrolled Descent of Incline Transfer Cart

Event Description: A remotely-operated incline transfer cart containing a SFA basket loses control during ascent up the incline transfer canal and results in an uncontrolled descent and impact with the ATS pool, thereby causing a breach and subsequent release. The pool water serves as a barrier to particulate release and, consequently, only the radioactive gases are released to the Waste Handling Building environment.

Technical Strategy: The primary safety strategy is to confine particulate releases within the ATS pool by designing the pool system in accordance with ANSI/ANS-57.7-1988.

#### 5.3.3.3 Handling Equipment Drop onto SFA Basket in Pool

Event Description: A lifting yoke (or other heavy object) is dropped onto a bare SFA in the ATS pool causing a breach and subsequent release. The pool water serves as a barrier to particulate release and, consequently, only the radioactive gases are released to the Waste Handling Building environment.

Technical Strategy: The primary safety strategy is to confine particulate releases within the ATS pool by designing the pool system in accordance with ANSI/ANS-57.7-1988.

#### 5.3.3.4 Handling Equipment Drop onto SFA Basket in Cell

Event Description: A lifting yoke (or other heavy object) is dropped onto a bare SFA in the ATS cell causing a breach and subsequent release.

Technical Strategy: The strategy is to confine particulate releases within the WHB by relying on the HVAC system HEPA filters.

#### 5.3.3.5 Unsealed Disposal Container Collision

Event Description: A loaded, unsealed disposal container collides with a wall, shield door, or other heavy object, resulting in the release of a fraction of its radiological contents.

Technical Strategy: The strategy is to confine particulate releases within the WHB by relying on the HVAC system HEPA filters and to provide design features (e.g., limit switches, redundant controls, emergency switch, etc.) and safe load paths to minimize the likelihood of a collision that could result in a radiological releases.

#### 5.3.3.6 Unsealed Disposal Container Drop and Slapdown

Event Description: A loaded, unsealed disposal container is dropped by the disposal container bridge crane onto a welding fixture or staging fixture. After dropping, the unsealed disposal container is presumed to slap down onto the floor and release a fraction of its radiological

contents. The drop height for this event is the normal handling height in the disposal container Handling Cell.

Technical Strategy: The strategy is to confine particulate releases within the WHB by relying on the HVAC system HEPA filters and to provide design features (e.g., limit switches, interlocks, redundant controls, redundant cables, physical restraints, etc.) that minimize unsealed disposal container drops or minimize the radiological release.

#### 5.3.3.7 Handling Equipment Drop onto Unsealed Disposal Container

Event Description: A lifting yoke (or other heavy object) is dropped onto a loaded, unsealed disposal container resulting in the release of a fraction of its radiological contents.

Technical Strategy: The strategy is to confine particulate releases within the WHB by relying on the HVAC system HEPA filters and to provide design features that minimize handling equipment drops onto spent fuel inside a disposal container.

#### 5.3.3.8 Unsealed Shipping Cask Drop into Cask Preparation Pit

Event Description: A shipping cask, without impact limiters and with its lid unbolted, is dropped from the normal lift height into the cask preparation pit in the ATS pool area.

Technical Strategy: The strategy is to confine particulate releases within the WHB by relying on the HVAC system HEPA filters and to provide design features that prevent or minimize cask drops (e.g., limit switches, interlocks, redundant control circuitry and/or cable restraints) or reduce the impact of a drop (e.g., shock absorber at base of pit). Administrative controls may also be employed to prevent the cask lid from being completely unbolted during the lift out of the cask preparation pit and into the ATS cask unloading pool.

#### 5.3.3.9 Unsealed Shipping Cask Drop into Cask Unloading Pool

Event Description: A shipping cask, without impact limiters and with its lid unbolted, is dropped by the cask bridge crane into the ATS cask unloading pool.

Technical Strategy: The strategy is to confine particulate releases within the ATS pool by designing the pool system in accordance with ANSI/ANS-57.7-1988. In addition, particulate mitigation in the ATS pool area is provided by the secondary HVAC confinement ventilation system.

### 5.3.4 Internal Event Sequences with No Release

Internal event sequences in this category are not expected to result in a radiological release because they are prevented by design. In other words, facility SSCs are credited to prevent a credible radiological release. Internal event sequences that are considered in the MGR design basis but consequently do not result in a release are identified in Table 5-7. Waste-package-related event sequences for the surface and subsurface were evaluated in *Waste Package Design Basis Events* (CRWMS M&O 2000d).

Table 5-7. Internal Event Sequences with No Release

| Event Group | Design Basis Event | Event Location | SSCs Credited to Prevent a Release |
|---|---|---|---|
| Shipping Cask-Related | Cask Carrier/Railcar Accident (with impact limiters) | Between Site Boundary and CPB | Shipping Cask |
| | Sealed Shipping Cask Drop Onto Floor (no impact limiters) | Carrier Bay | Shipping Cask, Carrier/Cask Handling System bridge crane, Lifting Fixtures |
| | Sealed Shipping Cask Drop Into Cask Preparation Pit (no impact limiters) | Cask Preparation Pit | Shipping Cask, Carrier/Cask Handling System bridge crane, Lifting Fixtures |
| | Shipping Cask Collision (no impact limiters) | CPB, Carrier Bay, or En-Route Between | Shipping Cask |
| | Handling Equipment Drops onto Cask (no impact limiters) | CPB or Carrier Bay | Shipping Cask |
| | Shield Door Closes on Cask | ATS or Canister Transfer System (CTS) Airlock | Shipping Cask |
| | Cask Cooldown System Overpressurization | ATS Cask Preparation Pit | Cask and DPC Preparation Cooling System |
| SFA-Related | Flooding Due to Uncontrolled Pool Water Fill/Draindown | ATS Pool & Surrounding Area | Pool Water Level Control |
| Canister-Related | Canister Drop* | CTS Cell | Canisters, CTS bridge crane, Lifting Fixtures |
| | Handling Equipment Drops onto Canister | CTS Cell | Canister, CTS bridge crane, Lifting Fixtures |
| Falling Objects Strike Disposal Container or Waste Package | WHB handling equipment falls on waste package | Disposal Container Handling System (DCHS) Cell | Disposal Containers, DCHS bridge cranes, Lifting Fixtures, Waste Package Horizontal Lifting System |
| | Underground handling equipment falls on waste package | Subsurface | Disposal Containers |
| | Drift liner / ground support falls on waste package | Subsurface | Disposal Containers |
| Waste Package Drops | Aboveground lifting system drops vertically oriented waste package | DCHS Cell | Disposal Containers |
| | Aboveground lifting system drops horizontally oriented waste package | DCHS Cell | Disposal Containers |
| | Bed plate rolls out of waste package transporter | Subsurface | Disposal Containers |
| | Emplacement gantry drops horizontally oriented waste package | Subsurface | Disposal Containers |
| | Waste package falls onto a sharp object | DCHS Cell or Subsurface | Disposal Containers |

* Exception is a naval canister drop, which may breach and cause a release but results in a negligible offsite dose (see Section 5.3.6.1).

Table 5-7. Internal Event Sequences with No Release (Continued)

| Event Group | Design Basis Event | Event Location | SSCs Credited to Prevent a Release |
|---|---|---|---|
| Waste Package Slapdown | Waste package tips over and slaps down on a flat surface | DCHS Cell | Disposal Containers |
| Waste Package Collisions | Waste package collides in lag storage area | DCHS Cell | Disposal Containers |
| | Transporter collisions at normal operating speeds | Subsurface | Disposal Containers |
| | Transporter derails w/o tip-over, but with waste package restraint failure | Subsurface | Disposal Containers |
| | Transporter derails with tip-over | Subsurface | Disposal Containers |
| | Transporter door closes on waste package | Subsurface | Disposal Containers |
| | Operation of emplacement gantry causes waste package collision | Subsurface | Disposal Containers |
| Missiles and Explosive Overpressure | Pressurized system missile strikes waste package | DCHS Cell | Disposal Containers |
| | Waste package missile strike from battery hydrogen explosion | DCHS Cell | Disposal Containers |
| Fire and Thermal Hazards | Thermally overloaded waste package | DCHS Cell | Disposal Containers |
| | Fire in Disposal Container Cell | DCHS Cell | Disposal Containers |
| | Transporter breakdown outside North Portal (insolation) | Between WHB and North Portal | Disposal Containers |
| | Thermally overloaded emplacement drift | Subsurface Emplacement Drift | Disposal Containers |
| | Loss of underground ventilation | Subsurface | Disposal Containers |
| | Rock fall buries waste package with debris | Subsurface Emplacement Drift | Disposal Containers |
| Waste Package Overpressure | Fuel Rod Rupture | DCHS Cell | Disposal Containers |
| Waste Package Criticality | Waste Package Criticality Misload | DCHS Cell | Disposal Containers |

### 5.3.5 Consequence Analysis

Consequence analyses of Category 1 DBEs utilize best estimate parameters and assumptions, where available, whereas consequence analyses of Category 2 DBEs apply conservative parameters and assumptions. The best estimate analyses performed for Category 1 DBEs are conservative because many of the assumptions and parameters used for Category 1 consequence analyses are chosen conservatively due to a lack of technical data or licensing precedent to support best estimate data.

The following sections describe the various parameters involved in the consequence analysis of Category 1 and Category 2 DBEs.

#### 5.3.5.1 Source Term Analysis

Source term analysis is the identification of radionuclides that contribute to the offsite dose during a potential DBE. The source terms evaluated for consequence analysis of MGR DBEs are focused on commercial SNF. Other DOE-owned SNF or HLW that will be handled at the MGR (see Section 5.3.5.1.3) are either bounded by commercial SNF or will not be released in an accident due to preventive and/or mitigative design features.

##### 5.3.5.1.1 Commercial Spent Nuclear Fuel

The commercial SNF source term inventory includes the quantity of each radioactive isotope (curies per fuel assembly) that is potentially at risk of being released. The total material-at-risk in a DBE is a product of the source term inventory per assembly and the number of assemblies involved in the event. The commercial SNF source terms are based on the most common types of SFAs that are handled at the MGR, PWR and BWR SFAs. Category 1 and Category 2 DBEs identified in Sections 5.3.2 and 5.3.3 involve commercial SNF.

For Category 1 DBE analysis, the Average PWR assembly characteristics, as defined in Table 5-8, are used. The average characteristics are representative of a hypothetical PWR SFA that results in the statistical average (in terms of curies per assembly) of the PWR SFAs, that could potentially be handled at the MGR. The average PWR source term was selected for Category 1 dose assessment instead of the average BWR source term because it results in offsite public doses that are a factor of 1.5-3.0 higher, depending on the event (i.e., pool drop or cell drop).

Table 5-8. Average Fuel Characteristics

| Fuel Type | Initial Enrichment (percent) | Burnup (GWd/MTU)* | Decay Period (Years) | Reference |
|---|---|---|---|---|
| Average PWR SFA | 4.0 | 48 | 25 | CRWMS M&O 1999b |

* Gigawatt-Day/Metric Ton of Uranium

For Category 2 event sequences, either the Maximum PWR fuel or the Maximum BWR fuel, as defined in Table 5-9, was used for dose assessment depending on which source term resulted in the highest offsite dose. For event sequences occurring in the pool, Maximum PWR fuel results in the largest offsite dose due to the larger inventory of radioactive gases compared to the Maximum BWR fuel. For event sequences involving particulate releases in a dry environment (e.g., cell), however, the Maximum BWR fuel results in a public offsite dose that is a factor of 1.1-2.2 higher (CRWMS M&O 2000a) than Maximum PWR fuel because of the increased crud inventory. Crud is a mixture of corrosion products that has been found on the exterior surfaces of zirconium alloy cladding of SNF rods. The crud radionuclides included in the dose calculations are Cobalt-60 (half-life of 5.3 years) and Iron-55 (half-life of 2.7 years). Crud dominates the offsite inhalation doses for dry event sequences involving Maximum PWR or BWR fuel because of the conservative release fractions and relatively short decay period (5 years) assumed (CRWMS M&O 2000a).

On average, crud is responsible for 67 percent of the total whole body inhalation dose for a dry event (as opposed to a pool event) involving the postulated breach of 21 PWR SFAs (CRWMS

M&O 2000a). The crud contribution for this same event involving 44 BWR SFAs is 85 percent of the total whole body inhalation dose (CRWMS M&O 2000a). The crud activity (Ci/fuel assembly) for BWR SFAs is higher than for PWR SFAs because of the larger surface area for crud to deposit on. Crud source terms for average and maximum PWR and BWR SFAs are calculated in CRWMS M&O (2000a).

Table 5-9. Maximum Fuel Characteristics

| Fuel Type | Initial Enrichment (percent) | Burnup (GWd/MTU)* | Decay Period (Years) | Reference |
|---|---|---|---|---|
| Maximum PWR SFA | 5.0 | 75 | 5 | CRWMS M&O 1999b |
| Maximum BWR SFA | 5.0 | 75 | 5 | CRWMS M&O 1999c |

* Gigawatt-Day/Metric Ton of Uranium

High-burnup mixed oxide fuel was compared with maximum PWR fuel in Attachment XI of CRWMS M&O (2000a) and found to result in a 12 percent lower offsite TEDE dose. Therefore, the source term associated with maximum PWR fuel is expected to bound that of high-burnup mixed oxide fuel and may be considered adequately conservative for consequence analysis.

#### 5.3.5.1.2 Failed Fuel

Failed fuel refers to the following four categories of "problematic" commercial SNF that are projected to be placed in disposable single element canisters (CRWMS M&O 1999f):

- "Mechanically and Cladding-Penetration Damaged–SNF classified as failed because of (1) mechanical damage that limits the ability to vertically lift the assembly or to fit within the dimension of standard, and/or (2) cladding damage resulting in loss of containment.
- Consolidated/Reconstituted Assemblies–SNF assemblies that were disassembled and, if reassembled, were done so in a form that is dimensionally different from the original.
- Fuel Rods, Pieces, and Debris–Variable-sized pieces of fuel (ranging from a single pellet to a full rod) and debris combining fuel and nonfuel materials.
- Nonfuel Components–Includes in-core assembly components physically separated from the assemblies and shipped separately."

The DBE source terms associated with failed fuel are assumed to be bounded by the radiological consequences of commercial SNF. This assumption is based on several factors, including: (1) failed fuel will have already released its gaseous radionuclides prior to any event at the MGR, (2) failed fuel is expected to be much older and further decayed, on average, than commercial SNF, and (3) failed fuel canisters are expected to be involved in a small percentage of the total number of spent fuel handlings.

#### 5.3.5.1.3 Department of Energy Spent Nuclear Fuel, Vitrified High-Level Waste, and Other Non-Commercial Waste Forms

The DOE and naval SNF, vitrified HLW, and other non-commercial waste forms will arrive at the proposed repository in welded disposable canisters ready for loading directly into disposal containers. Except for naval canisters, the handling of these canisters will rely on a strategy that ensures by design of the facility handling equipment and certification of the canisters that a canister breach is not credible. Therefore, with the exception of naval canisters, a canister drop and breach event is a BDBE that is not considered in the DBE consequence analysis.

Because of the robust nature of naval SNF and detailed knowledge of the fuel inventory and behavior of the waste form, the naval canisters will not require certification to withstand all credible design basis event impacts at the MGR. Therefore, qualified source terms for naval fuel have been established and considered in the DBE consequence analysis (see Section 5.3.6).

The Department of the Navy through the Naval Nuclear Propulsion Program provided the naval SNF source term (Naples 1999). A list of radionuclides that could be released from the breached naval disposable canister and that contribute 99.9 percent of the possible radiation dose was provided and used in the DBE dose calculations for naval SNF (CRWMS M&O 1999i).

Source terms for DOE SNF, high-level waste glass and other non-commercial waste forms were considered for evaluation of BDBE scenarios but were not included in the DBE consequence analysis (see Section 5.3.6).

#### 5.3.5.2 Accident Release Fractions

The accident release fraction is the fraction of the total inventory of a given radionuclide that is released to the environment upon breach of the wasteform. The accident release fraction for commercial SNF is a measure of the inventory of fuel particulates, gases and volatile species present in the fuel-cladding gap region of each breached fuel rod.

The total release fraction for calculating the source term released is a function of the cladding damage fraction (DF), cladding release fraction (CRF), airborne release fraction (ARF), respirable fraction (RF), and the local deposition factor (DEP):

$$\text{Accident Release Fraction} = DF \times CRF \times ARF \times RF \times DEP \qquad \text{(Eq. 5-1)}$$

The DF is the fraction of fuel rods in one spent fuel assembly that is assumed to fail by cladding breach during an event. The CRF is the fraction of the total radionuclide inventory in the fuel-cladding gap that actually escapes from a breached fuel rod. The ARF is the fraction of radionuclide inventory released from the fuel rods that becomes airborne and is available for transport to the external environment through the ventilation exhaust. The RF is the fraction of the ARF that is of respirable size (<10 micron) and which could contribute to the inhalation dose. The DEP is the fraction of material released in a confinement area that escapes from the building and is released to the external environment.

The DF, CRF, and DEP fractions are assumed equal to 1.0. Using a value of 1.0 maximizes the release fraction of radionuclides during an accident. This is a conservative assumption since

each of these factors is realistically expected to provide some degree of radionuclide retention in the event of a SNF assembly/basket drop event in the surface facility. A DEP factor of 1.0 implies that there is no local deposition or gravitational settling of radioactive particles on surfaces in the building.

The release fraction for crud particles is the exception to Equation 5-1. Crud release is not dependent on the DF and CRF. Crud is present on the outer surfaces of spent fuel assemblies, therefore, any event sequence that impacts an assembly has the potential to release crud particles. While a significant fraction of crud might realistically be retained on the cladding surface during impacts, the present analysis assumed 100% of the crud is released. The crud release fraction is the product of the ARF, RF and DEP.

The ARF and RF parameters for Category 1 and 2 event sequences involving commercial SNF releases in air (herein referred to as dry event sequences) are based on *Commercial SNF Accident Release Fractions* (CRWMS M&O 1999d), except for the RF for Category 1 releases in air. In this case, a RF of 1.0 is conservatively assumed since all particle sizes must be considered when calculating the ingestion portion of the Category 1 TEDE. Since Category 2 event sequences result in acute exposures, the ingestion pathway is not included (see Section 5.3.5.4.2) and only the respirable size particles are available for inhalation.

For event sequences occurring in a pool (herein referred to as pool event sequences), an ARF equal to zero is assumed for particulates and volatiles. In these event sequences, only the noble gases are released from the pool. The release fractions for commercial SNF releases in air and water for Category 1 and Category 2 dose assessments are shown in Table 5-10 and Table 5-11, respectively.

Table 5-10. Category 1 DBE Release Fractions

| Release Fractions From Event Sequences That Occur in Air | | | | | | |
|---|---|---|---|---|---|---|
| Nuclide | DF | ARF | DEP | CRF | RF | Effective Release |
| Hydrogen-3 | 1.0 | 0.3 | 1.0 | 1.0 | 1.0 | 3E-1 |
| Krypton-85 | 1.0 | 0.3 | 1.0 | 1.0 | 1.0 | 3E-1 |
| Iodine-129 | 1.0 | 0.3 | 1.0 | 1.0 | 1.0 | 3E-1 |
| Cesium | 1.0 | 2E-4 | 1.0 | 1.0 | 1.0 | 2E-4 |
| Strontium | 1.0 | 3E-5 | 1.0 | 1.0 | 1.0 | 3E-5 |
| Rubidium | 1.0 | 3E-5 | 1.0 | 1.0 | 1.0 | 3E-5 |
| Crud | 1.0 | 1E+0 | 1.0 | 1.0 | 1.0 | 1E+0 |
| Fuel Fines | 1.0 | 3E-5 | 1.0 | 1.0 | 1.0 | 3E-5 |
| Release Fractions From Event Sequences That Occur in Water | | | | | | |
| Nuclide | DF | ARF | DEP | CRF | RF | Effective Release |
| Hydrogen-3 | 1.0 | 0.3 | 1.0 | 1.0 | 1.0 | 3E-1 |
| Krypton-85 | 1.0 | 0.3 | 1.0 | 1.0 | 1.0 | 3E-1 |
| Iodine-129 | 1.0 | 0.3 | 1.0 | 1.0 | 1.0 | 3E-1 |
| Cesium | 1.0 | 0.0 | 1.0 | 1.0 | 1.0 | 0 |
| Strontium | 1.0 | 0.0 | 1.0 | 1.0 | 1.0 | 0 |
| Rubidium | 1.0 | 0.0 | 1.0 | 1.0 | 1.0 | 0 |
| Crud | 1.0 | 0.0 | 1.0 | 1.0 | 1.0 | 0 |
| Fuel Fines | 1.0 | 0.0 | 1.0 | 1.0 | 1.0 | 0 |

Table 5-11. Category 2 DBE Release Fractions

| Release Fractions From Event Sequences That Occur in Air | | | | | | |
|---|---|---|---|---|---|---|
| Nuclide | DF | ARF | DEP | CRF | RF | Effective Release |
| Hydrogen-3 | 1.0 | 0.3 | 1.0 | 1.0 | 1.0 | 3E-1 |
| Krypton-85 | 1.0 | 0.3 | 1.0 | 1.0 | 1.0 | 3E-1 |
| Iodine-129 | 1.0 | 0.3 | 1.0 | 1.0 | 1.0 | 3E-1 |
| Cesium | 1.0 | 2E-4 | 1.0 | 1.0 | 1.0 | 2E-4 |
| Strontium | 1.0 | 3E-5 | 1.0 | 1.0 | 5E-3 | 2E-7 |
| Rubidium | 1.0 | 3E-5 | 1.0 | 1.0 | 5E-3 | 2E-7 |
| Crud | 1.0 | 1E+0 | 1.0 | 1.0 | 3E-1 | 3E-1 |
| Fuel Fines | 1.0 | 3E-5 | 1.0 | 1.0 | 5E-3 | 2E-7 |
| Release Fractions From Event Sequences That Occur in Water | | | | | | |
| Nuclide | DF | ARF | DEP | CRF | RF | Effective Release |
| Hydrogen-3 | 1.0 | 0.3 | 1.0 | 1.0 | 1.0 | 3E-1 |
| Krypton-85 | 1.0 | 0.3 | 1.0 | 1.0 | 1.0 | 3E-1 |
| Iodine-129 | 1.0 | 0.3 | 1.0 | 1.0 | 1.0 | 3E-1 |
| Cesium | 1.0 | 0.0 | 1.0 | 1.0 | 1.0 | 0 |
| Strontium | 1.0 | 0.0 | 1.0 | 1.0 | 1.0 | 0 |
| Rubidium | 1.0 | 0.0 | 1.0 | 1.0 | 1.0 | 0 |
| Crud | 1.0 | 0.0 | 1.0 | 1.0 | 1.0 | 0 |
| Fuel Fines | 1.0 | 0.0 | 1.0 | 1.0 | 1.0 | 0 |

### 5.3.5.3 Atmospheric Dispersion Factors

Atmospheric dispersion factors ($\chi/Q$ values) are used to estimate the dispersion of radionuclides between the facility release point and the receptor at the site boundary. Atmospheric dispersion factors are based on site-specific meteorological data collected at Yucca Mountain between 1993 and 1997, as reported in *Calculations of Acute and Chronic "Chi/Q" Dispersion Estimates for a Surface Release* (CRWMS M&O 1999e). Category 1 event releases are modeled as "chronic" releases and use the maximum sector chronic $\chi/Q$ at the distances evaluated. The chronic $\chi/Q$ is an annual average, or best estimate, $\chi/Q$. Category 2 event releases are modeled as "acute" releases and use the conservative, maximum sector 99.5 percentile acute (0.5 percent exceedance) $\chi/Q$ values for the distances evaluated. The selection of the maximum sector 99.5 percent $\chi/Q$ value was based on being larger than the 95 percent overall site $\chi/Q$ value, in accordance with NRC Regulatory Guide 1.145.

A site boundary distance of 11-kilometers was used to calculate doses due to radiological releases from the WHB (CRWMS M&O 2000a). This is a conservative estimate of the distance from the MGR WHB ventilation exhaust shaft to the nearest point on the proposed YMP Withdrawal Area boundary (to the West, see Figure 5-4) and is assumed to be the closest point that any member of the public could be located at the time of a postulated radiological release. It is assumed that administrative controls will be in place to evacuate any members of the public that could potentially be located within the YMP Withdrawal Area but outside of the Preclosure Controlled Area Boundary (Figure 5-4) following a Category 2 DBE. Personnel located on the Nevada Test Site and Nellis Air Force Range are not considered part of the public. These

personnel are government workers located on government property and subject to evacuation if required.

A site boundary distance of 8 kilometers (5 miles) was used to calculate potential doses due to radiological releases from the subsurface repository. This is a conservative estimate of the approximate distance between the potential repository and the nearest point of public access on the proposed YMP Withdrawal Area boundary (to the West, see Figure 5-4).

The methodology used to calculate the atmospheric dispersion factors uses a straight line Gaussian distribution which does not account for the terrain effects between the source and the receptor. The straight-line Gaussian distribution model provides conservative estimates for atmospheric dispersion factors for the Yucca Mountain Site. The mountains and valleys that exist between the WHB and the potential population would provide some channeling of the radioactive plume and, therefore, may affect both the direction in which the plume travels and the time at which the plume arrives in a given location. Atmospheric dispersion factors were calculated assuming ground level releases. This assumption adds to the conservatism included in the atmospheric dispersion factors. Assuming a stack release would reduce the estimated $\chi/Qs$ due to the increased dispersion associated with elevated releases.

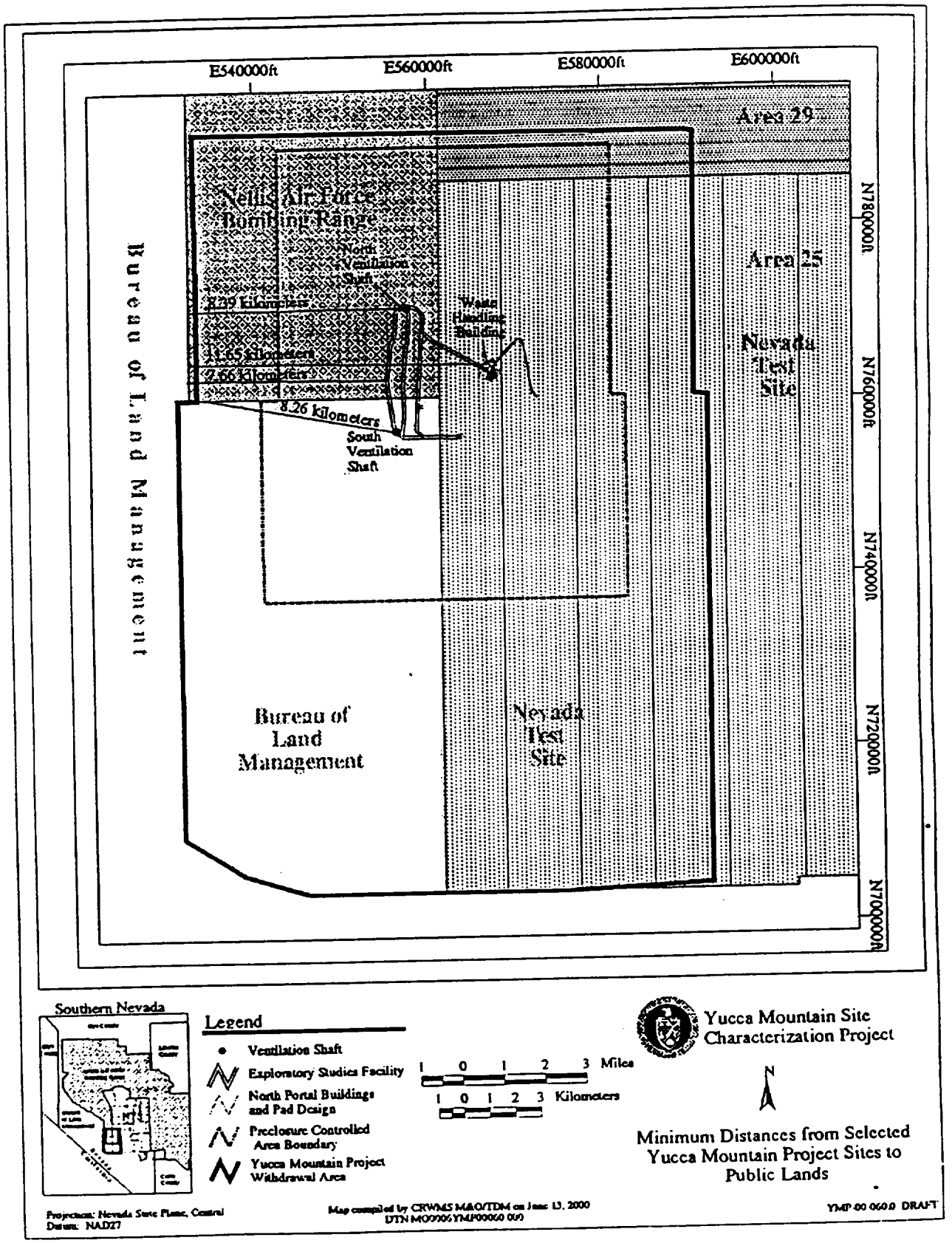


Figure 5-4. Minimum Distances to Site Boundary

### 5.3.5.4 Dose Calculation Models

Two models were used to calculate offsite doses: one for modeling the chronic, annual offsite dose resulting from Category 1 DBE exposures; and the other for modeling the acute offsite dose due to Category 2 DBE exposures. The term "chronic" is used to describe the annual exposure to an individual that is living at the site boundary and continuously exposed to an averaged level of radiation. By comparison, "acute" refers to the maximum exposure that an individual at the site boundary is expected to receive over a period of 2 hours.

The six dose measures applicable to Category 1 and/or Category 2 DBEs at the MGR are defined as follows (10 CFR 20.1003):

A. *Deep-Dose Equivalent (DDE).* Applies to external whole-body exposure and is equal to the dose equivalent at a tissue depth of 1 cm.

B. *Committed Dose Equivalent (CDE).* The dose equivalent to an individual organ or tissue that is received from an intake of radioactive material by an individual during the 50-year period following the intake.

C. *Committed Effective Dose Equivalent.* The sum of the products of the weighting factors applicable to each of the body organs or tissues that are irradiated and the committed dose equivalent to these organs or tissues, commonly referred to as the "whole body dose."

D. *Total Effective Dose Equivalent (TEDE).* The sum of the deep-dose equivalent (for external exposures) and the committed effective dose equivalent (for internal exposures).

E. *Lens Dose Equivalent (LDE).* Applies to the external exposure of the lens of the eye and is taken as the dose equivalent at a tissue depth of 0.3 centimeter. The LDE was not calculated because Krypton-83m is the only nuclide with a submersion exposure-to-dose conversion factor for lens of the eye identified in Federal Guidanee Report No. 11 (Eckerman et al. 1988) and Krypton-83m is not present in the MGR source terms because of its short half-life (1.8 hours) and the minimum decay time (5 years) for SNF handled at the MGR. Furthermore, Federal Guidance Report No. 12 (Eckerman and Ryman 1993), which updates the dose coefficients for submersion given in Federal Guidance Report No. 11, does not include any dose coefficients for lens of the eye.

F. *Skin & Extremities.* Applies to the dose to the skin and extremities due to the air submersion pathway.

G. *Annual Dose.* This term applies to the yearly dose (rem/year) received by a receptor (worker or individual member of the public) as a result of exposure to Category 1 events and normal operational releases, from all applicable dose pathways.

#### 5.3.5.4.1 Category 1 Dose Model

Dose calculations for Category 1 event sequences consider all of the potential exposure pathways, including inhalation, ingestion, submersion, and groundshine. Category 1 doses are calculated with the GENII-S Code (CRWMS M&O 1998c) assuming the dose receptor is a hypothetical subsistence farmer living at the site boundary. The input parameters and methodology for Category 1 dose assessment is provided in CRWMS M&O (2000a).

The total Category 1 annual dose is based on contributions from three sources that are expected to contribute during the facility operational lifetime: Category 1 DBEs, normal operational (routine) releases from the WHB and WTB, and normal operational (routine) releases from the subsurface repository. The total Category 1 annual dose (mrem/yr) is described by the following equation:

$$D_{Cat.1}^{TOT} = D_{Cat.1}^{DBEs} + D_{NO}^{WHB} + D_{NO}^{Sub} \qquad \text{(Eq. 5-2)}$$

where

$D_{Cat.1}^{DBEs}$ = Annual dose due to all Category 1 DBE releases [mrem/yr].

$D_{NO}^{WHB}$ = Annual dose due to normal operational releases from the WHB [mrem/yr].

$D_{NO}^{Sub}$ = Annual dose due to normal operational releases from the subsurface repository [mrem/yr].

The radiological release (Ci/yr) estimates for each of these components were used as an input to the GENII-S dose calculation (CRWMS M&O 2000a, Attachment IV). The annual release due to individual Category 1 DBEs is based on the following equation:

$$R_{DBEs}^{TOT} = \sum_{i=1}^{n} R_i^{DBE} \cdot f_i \qquad \text{(Eq. 5-3)}$$

where

i = Index for a given Category 1 event sequence (i=1,2...n).
n = Total number of Category 1 event sequences.
$R_i^{DBE}$ = Radiological release due to event sequence i [Ci/event].
$f_i$ = Frequency of event sequence i [events/yr].

#### 5.3.5.4.2 Category 2 Dose Model

Only the inhalation and air submersion offsite dose are considered for Category 2 DBEs. The potential doses from ingestion, water immersion, and contaminated soil are not considered consistent with industry and regulatory practice.

The TEDE dose for Category 2 DBEs is expressed as:

$$TEDE = \sum_j D^{inh}_{j,effective} + \sum_j D^{sub}_{j,effective} \qquad \text{(Eq. 5-4)}$$

where

$D^{inh}_{j,effective}$ = Whole body "effective" inhalation dose [rem] from the $j^{th}$ isotope, also referred to as the committed effective dose equivalent.

$D^{sub}_{j,effective}$ = Whole body "effective" submersion dose [rem] from the $j^{th}$ isotope, also referred to as the DDE.

The sum of the DDE and the CDE to any individual organ or tissue (other than the lens of the eye), is expressed as:

$$CDE_k + DDE = \sum_j D^{inh}_{j,k} + \sum_j D^{sub}_{j,effective} \qquad \text{(Eq. 5-5)}$$

where

$CDE_k$ = Committed dose equivalent to the $k^{th}$ organ or tissue [rem].

$D^{inh}_{j,k}$ = Radiation dose from the $j^{th}$ isotope to the $k^{th}$ organ or tissue due to inhalation [rem].

$k$ = organ or tissue index, including gonads, breast, lungs, red marrow, bone surface, thyroid, and remainder†.

† The remainder represents the next five remaining organs or tissues (such as liver, kidneys, spleen, brain, small intestine, upper large intestine, lower large intestine, etc., but excluding the skin, lens of the eye, and the extremities) receiving the highest doses (Eckerman et al. 1988).

The inhalation and air submersion doses shown above can be further expressed as:

$$D^{inh}_{j,k} = ST_j \times SFA \times \frac{\chi}{Q} \times BR \times conv \times DCF^{inh}_{j,k} \qquad \text{(Eq. 5-6)}$$

$$D^{sub}_{j,k} = ST_j \times SFA \times \frac{\chi}{Q} \times conv \times DCF^{sub}_{j,k} \qquad \text{(Eq. 5-7)}$$

where

$ST_j$ = Inventory source term release per spent fuel assembly for the $j^{th}$ isotope[Ci/SFA]. Equal to the inventory source term multiplied by the accident release fraction (Eq. 5-1).

$SFA$ = Number of spent fuel assemblies involved in the release [# SFAs].

$\frac{\chi}{Q}$ = Atmospheric dispersion factor [seconds/cubic meter].

$BR$ = Breathing rate [cubic meters/second].

$DCF_{j,k}^{inh}$ = Inhalation dose conversion factor of the $j^{th}$ isotope for the $k^{th}$ organ [Sv/Bq] (Eckerman et al. 1988).

$DCF_{j,k}^{sub}$ = Air submersion dose coefficient of the $j^{th}$ isotope for the $k^{th}$ organ [(Sv-cubic meter)/(Bq-s)] (Eckerman and Ryman 1993).

$conv$ = DCF unit conversion factor: 3.7E12 (rem-Bq)/(Ci-Sv) (Eckerman et al. 1988).

The skin dose is equal to

$$SKIN = \sum_j D_{j,skin}^{sub} \qquad \text{(Eq. 5-8)}$$

where

$SKIN$ = Dose to the skin [rem].

$D_{j,skin}^{sub}$ = Radiation dose from the $j^{th}$ isotope to the skin due to air submersion [rem]; obtained using Eq. 5-7.

### 5.3.6 Comparison to Dose Limits

The regulatory dose limits applicable to the MGR for offsite public exposures are contained in the proposed 10 CFR Part 63 interim guidance (Dyer 1999, Section 111), proposed 40 CFR 197.4(a) (64 FR 46976), and 10 CFR 20.1301. Occupational dose limits and ALARA requirements are established in 10 CFR 20.1201 and 10 CFR 20.1101, respectively.

The regulatory dose limits are exclusive of the dose contributions from natural background radiation or other artificial radiation sources. For relative comparison, the typical per capita radiation dose from all sources of exposure in the United States is approximately 160 mrem/year: (Eisenbud 1987)

- Natural radioactivity (e.g., radon, cosmic rays): 80 mrem/year
- Medical and dental (diagnostic): 75 mrem/year
- Fallout from weapons testing: 4-5 mrem/year
- Consumer products: 1-4 mrem/year
- Nuclear power production: <1 mrem/year

#### 5.3.6.1 Public Dose

The radiation dose limits applicable to an individual member of the public from Category 1 DBEs include a 25 mrem/year TEDE limit from the proposed 10 CFR Part 63 interim guidance (Dyer 1999, Section 111(a)(2)), a CEDE limit of 15 mrem/year from the proposed 40 CFR 197.4(a) (64 FR 46976), a TEDE limit of 100 mrem/year from 10 CFR 20.1301(a)(1), and a 10

mrem/year constraint specified in 10 CFR 20.1101(d) for implementation of the ALARA requirement for air emissions of radioactive material to the environment. For external radiation exposures of the public, a limit of 2 mrem/hour is specified in 10 CFR 20.1301(a)(2).

The YMP Withdrawal Area boundary (Figure 5-4) is used as the site boundary for calculating public doses (see Section 5.3.5.3) due to DBE and normal operational releases. The calculated annual TEDE dose for all Category 1 DBEs, including normal operational releases from the MGR surface and subsurface facilities, is 6E-2 mrem per year (CRWMS M&O 2000a). This dose is well below the dose limits specified above and indicates that the MGR is able to comply with all of the applicable regulatory dose limits to the public.

The external radiation dose to the public is expected to be orders of magnitude lower than 2 mrem/hour limit specified in 10 CFR 20.1301. This expectation is attributed to several factors including:

1. The distance between the Waste Handling Building and the nearest unrestricted area of the site boundary (approximately 11 km);
2. A maximum allowable radiation dose of 10 mrem/hour at a distance of 2 meters from the surface of transportation casks (49 CFR 173.441(b)(3));
3. Shielding of source terms in the Waste Handling Building;
4. Shielding surrounding the waste package transporter.

The Category 2 DBE dose limits for any individual located on or beyond any point on the boundary of the site, as specified in Section 111 of Dyer (1999), include:

- The more limiting of a TEDE of 5 rem, or
- The sum of the DDE and the maximum CDE to any individual organ or tissue (other than the lens of the eye) of 50 rem; and
- An LDE of 15 rem; and
- A shallow dose equivalent to skin of 50 rem.

Dose calculations for Category 2 DBEs involving commercial SNF indicate the following maximum doses for any one event sequence (CRWMS M&O 2000a):

- TEDE = 2E-2 rem
- DDE + Maximum CDE = 1E-1 rem (maximum CDE is to the Lung)
- Shallow Dose Equivalent to Skin = 4E-2 rem

The Category 2 dose results shown above are based on exposure to a HEPA-filtered release caused by the bounding-consequence Category 2 event (Table 5-6, Event 2-08) , the drop and breach of a shipping cask containing 68-BWR assemblies with the maximum source term (see

Section 5.3.5.1.1). The LDE was not calculated in CRWMS M&O (2000a) because Krypton-83m is the only nuclide having a submersion exposure-to-dose conversion factor and the Krypton-83m source term is negligible. The Krypton-83m source term is negligible because of its short half-life (1.8 hours) and the minimum decay requirement (5 years) for receipt of SNF at the MGR.

The dose assessment of naval fuel (CRWMS M&O 1999i) indicates the following bounding offsite dose results for HEPA-filtered source terms released to the environment during a postulated canister drop event (see Table 5-7 footnote) that breaches a single naval canister:

- TEDE = 3E-7 rem
- DDE + Maximum CDE = 2E-6 rem
- Shallow Dose Equivalent to Skin = 3E-9 rem

### 5.3.6.2 Worker Dose

The occupational dose limits for adults, specified in 10 CFR 20.1201, include:

1. An annual limit of either (whichever is more limiting):

   - TEDE of 5 rem, or
   - The sum of the DDE and the CDE to any individual organ or tissue, other than the lens of the eye, of 50 rem.

2. Annual limits to the lens of the eye, to the skin, and to the extremities of:

   - An LDE of 15 rem, and
   - A shallow-dose equivalent of 50 rem to the skin or to any extremity.

The dose limits for workers apply to Category 1 event sequences, which are expected to occur during the preclosure lifetime of the MGR facilities, and normal operational exposures. A dose assessment of Category 1 event sequences was performed in CRWMS M&O (2000a) to estimate the worker dose from inhalation and submersion pathways at an assumed distance of 100 meters. This distance is typical of nuclear facility dose assessments for noninvolved workers. The noninvolved worker is assumed to be a worker not directly involved with the waste handling operations for which the accident is postulated. Category 1 event sequences considered a ground-level, HEPA-filtered release of commercial SNF. The results of the worker dose calculations in CRWMS M&O (2000a) are provided below:

- TEDE = 1E-2 rem/year
- DDE + Maximum CDE = 1E-1 rem/year
- Shallow Dose Equivalent to Skin = 1E-1 rem/year

These results indicate that the MGR is able to comply with the applicable regulatory dose limits for workers specified in 10 CFR 20.1201.

The LDE was not calculated in CRWMS M&O (2000a) for the reasons discussed in Section 5.3.6.1. The 100 meter distance used to calculate the dose to a noninvolved or colocated

site worker is typically used in Nuclear Regulatory Commission licensing submittals to establish compliance with worker dose limits. Closer distances are not valid for inhalation and submersion exposures because of restrictions on dispersion modeling at close-in distances. All of the Category 1 events that contribute to the worker dose occur either in cells, where workers are not present and are protected by shield walls, or in pool areas where particulate radionuclides are retained by the pool water. In addition, the Waste Handling Building ventilation system is designed to control airflow, filter particulates, and vent radiological releases through an elevated stack to the external environment. Therefore, the potential radiological exposure during an accident for workers located less than 100 meters from a radiological release (e.g., inside the Waste Handling Building) is expected to be minimal.

The worker dose assessment in CRWMS M&O (2000a) does not include contributions from direct radiation exposures to workers during normal operations. The direct radiation dose to workers from normal operations is limited by the facility design and administrative controls, as implemented by the radiation protection program (Section 7). For example, most of the waste handling operations are performed remotely in cells with sufficient shielding to protect operators. Maintenance operations in the cells is prohibited during waste handling operations. In addition, workers will be continuously monitored for radiation doses and relocated or reassigned to less-hazardous duties if their individual doses exceed operational control limits. Finally, the use of site radiological monitoring and control systems, training, and procedures will also ensure that workers are promptly evacuated and minimally exposed to radiological hazards. The worker doses reported herein are preliminary and subject to further analysis as additional design and operational details become available. However, the maximum worker dose will be lower than the limits established in 10 CFR 20.1201.

### 5.3.7 Identification of Important to Safety Structures, Systems, and Components

The results of the DBE analysis in Section 5.3 were used to identify SSCs that are important to safety. Refer to Section 4.4.1 for a complete discussion of SSCs important to safety.

## 5.4 BEYOND DESIGN BASIS EVENTS

BDBE sequences are internal event sequences that have less than one chance in 10,000 of occurring during the preclosure period. Assuming a preclosure period of 100 years, this corresponds to an annual frequency of 1E-6 per year. By definition, BDBE sequences are not subject to the DBE dose limits in Section 111 of Dyer (1999). BDBEs are discussed in this analysis for completeness.

BDBEs considered in this section are internal event sequences that the MGR has specifically addressed in the preliminary design process in order to ensure that the event sequence does not occur (i.e., ensure frequency is below 1E-6 per year). The event sequences identified in Table 5-12 are BDBEs because of design features, physical barriers, administrative controls, or a combination thereof that ensure that the sequence of events necessary to result in a radiological release is beyond design basis. Should the prevention or mitigation features be altered as the design evolves, some of the BDBE sequences could become credible event sequences and subject to DBE dose limits.

Table 5-12. Beyond Design Basis Events

| Event Group | Event [1] | Location | Design/ Mitigation Feature |
|---|---|---|---|
| Fire | Fire in Surface Facilities Resulting in Radiological Release | WHB or WTB | Design layout and administrative controls ensure that a credible fire cannot result in a radiological release. See Section 8 for discussion of MGR Fire Protection program. |
| | Fire in Subsurface Facilities Resulting in Radiological Release | Subsurface | Design layout and administrative controls ensure that a credible fire cannot result in a radiological release. See Section 8 for discussion of MGR Fire Protection program. |
| Shipping Cask-Related | Drop of Shipping Cask from Beyond its Design Basis Height (w/o impact limiters)(i.e., Two-Block Drop), No Filtration | Carrier Bay | Shipping Cask, Carrier/Cask Handling System bridge crane, Lifting Fixtures |
| | Cask Drop into Cask Preparation Pit, No Filtration | ATS Cask Preparation Pit | WHB Confinement Area Ventilation System, Carrier/Cask Handling System bridge crane, Lifting Fixtures |
| | Non-Mechanistic Shipping Cask Leak [3] | Carrier Bay | Shipping Cask |
| | Diesel Fire/Explosion Resulting in Breach of a Shipping Cask | Outside CPB | Shipping Cask; No ignition source present to initiate a fire or explosion capable of breaching a cask |
| SFA-Related | SFA Basket Drop onto Another SFA Basket, No Filtration | ATS Dryer | WHB Confinement Area Ventilation System, Dry Assembly Handling System, Dry Assembly Transfer Machine |
| | Catastrophic Pool Failure | ATS Pool | WHB Structure, Pool Water Level Control |
| | Criticality Event in Pool | ATS Pool | Assembly Transfer Baskets, Basket Staging Racks |
| | Loss of Pool Water Resulting in Zirconium Alloy Cladding Fire | ATS Pool | None – analysis expected to demonstrate there is insufficient heat output to initiate cladding fire (565°C) |
| | Cladding Failure in the ATS Dryer | ATS Dryer | None – analysis expected to demonstrate there is insufficient heat output to ignite cladding (565°C) or initiate preclosure cladding failure |
| | Welding Burnthrough of SNF | DCHS Welding Station | None – analysis expected to demonstrate that a welding error resulting in breach of the SNF cladding is physically impossible |
| Canister-Related | Impact to Disposable Canister that Exceeds its Design Basis (e.g., Two-Block Drop) [2] | CTS Cell | Canisters, Canister Transfer System bridge crane, Lifting Fixtures |
| | Criticality Associated with Small Canister Staging Rack | CTS Cell | Small Canister Staging Racks |

Table 5-12. Beyond Design Basis Events (Continued)

| Event Group | Event [1] | Location | Design/ Mitigation Feature |
|---|---|---|---|
| Disposal Container/Waste Package-Related | Impact to Waste Package that Exceeds its Design Basis (e.g., Two-Block Drop) | Disposal Container Cell | Disposal Containers, DCHS bridge cranes, Waste Package Horizontal Lifting System, Lifting Fixtures |
| | Unsealed Disposal Container Drop (from normal handling height) onto Cell Floor, No Filtration | Disposal Container Cell | WHB Confinement Area Ventilation System, DCHS bridge cranes, Lifting Fixtures |
| | Preclosure Early Failure of Waste package [3] | Subsurface | Disposal Containers |
| | Criticality Due to Waste Package Internal Geometry Failure | DCHS Cell or Subsurface | Disposal Containers |
| | Criticality Due to Waste Package Flooding | DCHS Cell | Waste Package Design; No water pipes located in DCHS; waste package decontamination process does not utilize water |
| | Greater-than-6-MT Rock Fall on Waste Package | Subsurface - Emplacement Drift | Disposal Containers, Emplacement Drift Ground Control |
| | Greater-than-6-MT Rock Fall on Transporter | Subsurface - Main Drift | Disposal Containers, Waste Package Transporter, Locomotives |
| | Transporter Runaway | Subsurface – North Ramp | Waste Package Transporter |

(1) Unless "No Filtration" is shown, HEPA filters are assumed to be available in the event sequence.

(2) Exception is the naval canister, which is not certified to withstand all credible handling impacts. However, the consequences of a naval canister breach are acceptable, as indicated in Section 5.3.6.1.

(3) Event consequences were analyzed deterministically in CRWMS M&O (2000a).

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### 5.5.2 Codes, Standards, Regulations, and Procedures

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# 6. CRITICALITY SAFETY

This section addresses the MGR preclosure criticality safety strategy which considers Yucca Mountain site characteristics, and natural and human-induced events and processes. The report provides an overview of the proposed MGR preclosure criticality safety strategy that is relevant to a site suitability determination as follows:

- Major design requirements and guidelines for criticality control
- Processes performed during the design phase to ensure criticality control
- General and MGR-specific criticality control design features that could be relied upon to satisfy safety goals
- Description of the Criticality Safety Program required to achieve safety goals.

The section then describes how the preclosure criticality safety strategy addresses potential impacts that Yucca Mountain site characteristics (i.e., geology, hydrology, climatology, and meteorology), including several natural and human-induced event sequences, could have on design and procurement of SSCs required for criticality control.

## 6.1 DESCRIPTION OF THE MGR PRECLOSURE CRITICALITY SAFETY STRATEGY

MGR criticality safety during preclosure operations will, in general, be designed into the MGR facility and fuel handling hardware itself. Compliance with regulatory requirements and adherence to criticality safety principles will also be assured by a comprehensive criticality safety program, which includes administrative controls and criticality safety training for personnel that handle fissionable materials.

### 6.1.1 MGR Facility

There are five basic SNF handling operations identified for the MGR that may affect preclosure criticality safety.

1. Operations in the carrier/cask handling system
2. Operations in the ATS
3. Operations in the CTS
4. Operations in the DCHS
5. Emplacement of waste packages and subsequent storage in Yucca Mountain.

Based on this design and operational concept, assumptions can be made regarding basic criticality control features that will be required to support preclosure operations. Criticality control in the carrier/cask handling system is largely provided by transportation casks designed and licensed in accordance with 10 CFR Part 71, "Packaging and Transportation of Radioactive Material," requirements and by spent fuel loading procedures implemented at the transportation cask's point of origin. The primary criticality control function in the assembly transfer system is provided by several components as follows: (a) by the transportation cask or canister prior to

unloading the fuel in the assembly staging pool, (b) by the assembly basket and basket staging rack when being staged in the assembly staging pool, (c) by the assembly basket when baskets are transferred from the assembly staging pool to the assembly handling cell, (d) by the assembly basket and assembly drying vessel during drying operations, and (e) by the disposal container basket following loading into the disposal container. The primary criticality control function in the CTS is provided by several components as follows: (a) by the transportation cask and disposable canisters prior to unloading the canisters, (b) by the canisters and canister staging rack when being staged, and (c) by the canisters and disposal container basket following loading into the disposal container. During operations in the disposal container handling system and emplacement/storage of waste packages, the primary criticality control function is provided by the disposal container and its basket. Other site and facility features may also be credited, either directly or indirectly, as criticality control features, such as the site elevation above the flood plane, structural components and barriers credited for preventing water intrusion into normally dry fuel handling areas or containers, structural components or devices credited for protection of criticality control components from heavy load drops or external missile hazards, and design and layout of fire suppression systems. In addition to the design features described above, criticality safety during preclosure operations also relies on management measures to ensure compliance with limits, such as fissionable mass loading limits, that may be established for specific design features.

The preceding discussion presents an overview of criticality safety provisions likely to be required as part of major repository facility systems and preclosure operations. In addition to these major operational components, support systems, such as facility ventilation or radioactive waste processing systems, will also likely require design and operational provisions to be implemented related to criticality control. The need for such support system criticality control provisions shall be identified as the repository design develops.

### 6.1.2 Regulatory Requirements and Industry Codes and Standards

#### 6.1.2.1 Regulatory Requirements

The regulatory requirements relating to criticality in *Interim Guidance Pending Issuance of New U.S. Nuclear Regulatory Commission (NRC) Regulations for Yucca Mountain, Nevada* (Dyer 1999) are general in nature and focus on the need to perform an Integrated Safety Analysis, which includes identification of all means employed to control criticality. As described Section 112, "the integrated safety analysis of the geologic repository operations area shall include (e) an analysis of the performance of the major design SSCs, both surface and subsurface, to identify those that are important to safety, including identification and description of controls that are relied on to limit or prevent potential accidents or mitigate their consequences, and including identification of measures taken to ensure the availability of identified safety systems." Compliance with the preclosure performance objectives will be demonstrated through an Integrated Safety Analysis of the geologic repository operations area, which involves a systematic examination of potential hazards including criticality.

The regulatory requirements applicable to operations involving fuel assemblies or canisters in transportation casks are derived from 10 CFR Part 71 regulations, which are not specifically addressed as part of this report. The carrier/cask handling system and ATS shall be designed to handle transportation casks in accordance with applicable Certificates of Compliance.

### 6.1.2.2 Regulatory Guides and NUREG Reports

NRC Regulatory Guide 3.71, *Nuclear Criticality Safety Standards for Fuels and Material Facilities*, was used in the development of preclosure criticality safety strategies. This regulatory guide provides guidance on complying with NRC criticality safety regulations by describing procedures for preventing nuclear criticality accidents in operations involving handling, processing, storing, and transporting special nuclear material at nuclear fuel and material facilities.

Guidance from NUREG/CR-6361, *Criticality Benchmark Guide for Light-Water- Reactor Fuel in Transportation and Storage Packages* (Lichtenwalter 1997), has been used in selecting benchmark cases to validate the criticality code system and establishing an upper subcritical limit to be applied when using specific computer code systems and associated analytical methodologies in design applications. This NUREG/CR report references two American Nuclear Society criticality safety standards, ANSI/ANS-8.1 and ANSI/ANS-8.17. They are discussed in Section 6.1.2.3.

The guidance provided in NUREG 1520, *Draft of the Standard Review Plan for the Review of a License Application for Fuel Cycle Facility* (NRC 1999), for both Integrated Safety Analysis and criticality safety analysis, is considered indicative of current NRC thinking on risk informed regulation of spent fuel handling and storage facilities.

### 6.1.2.3 Industry Standards

NRC Regulatory Guide 3.71 provides a comprehensive list of industry standards potentially applicable to preclosure criticality safety. Criticality standards of particular note considered to be applicable to preclosure criticality safety are discussed below.

- ANSI/ANS-8.1-1983 (Reaffirmed in 1988), *American National Standard for Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors*. This standard provides general guidance for preventing criticality accidents in the handling, storage, processing and transporting of certain fissionable materials, specifically U-233, U-235, and Pu-239. It also provides basic criteria and limits for certain simple geometries of fissionable materials. It states requirements for establishing validity and ranges of applicability of any calculational method used in criticality safety analysis.

- ANSI/ANS-8.17-1984 (Reaffirmed in 1997), *Criticality Safety Criteria for the Handling, Storage, and Transportation of LWR Fuel Outside Reactors*. This standard provides guidance for preventing criticality accidents during the handling, storage, and transportation of reactor fuel. ANSI/ANS-8.17-1984 is intended to supplement ANSI/ANS-8.1-1983 with additional guidance specific to reactor fuel handling. This standard allows for neutron absorbers to be used for criticality control, and credit to be taken for burnup through reactivity measurements or through analysis and verification of exposure history. It also provides criteria for establishing criticality safety, although no specific safety margin is recommended.

- ANSI/ANS-8.19-1996, *Administrative Practices for Nuclear Criticality Safety*. This standard provides general guidance for implementation of an effective nuclear criticality

safety program at operating facilities where fissionable materials are handled. An effective nuclear criticality safety program requires cooperation among management, supervision, and criticality safety staff and relies upon conformance with operating procedures by employees.

### 6.1.3 Criticality Design Approach

The guidance from NUREG-1520 (NRC 1999) and American National Standards Institute criticality safety standards will be demonstrated through a comprehensive set of detailed calculations and safety analyses performed consistent with proven state-of-the-art methodology.

The design approach is described in detail as follows:

- The entire facility is reviewed to identify locations where fissionable material may exist.

- Each potential workstation or area that may contain fissionable material is analyzed, to determine the physical characteristics and configurations present and risks involved in operations.

- The operations are reviewed to identify potential controlled parameters (see Section 6.1.4) and select a preferred means of criticality control, which includes identifying one or more controlled parameters. The general guidance used for preclosure waste form storage and handling at the MGR is to employ, where practicable, reliance on equipment design that uses passive-engineered controls rather than on administrative controls. Where possible, geometry control will be designed into the facility. If necessary, fixed neutron absorbers will be provided. Where geometry control alone, including any fixed absorbers, is not possible, limits on fissionable material mass or other reliable and verifiable reactivity control methods, such as minimum fuel burnup requirements, will be established.

- As appropriate, analyses will employ conservative neutron moderation and reflection assumptions to conservatively bound any possible reflector environment, including consideration of the possible presence of materials such as bodies of personnel, oils, water, concrete, lead, or other metals such that they will not cause an increase in reactivity beyond limits.

- A modern comprehensive criticality analysis methodology validation will be performed consistent with NUREG/CR-6361 (Lichtenwalter et al. 1997). The range of situations to be evaluated for the MGR is large. Fissionable material forms to be evaluated include a range of commercial SNF designs and burnups, and other defense waste forms. Benchmark experiments shall be selected for each specific design application in order to establish appropriate values for method bias and uncertainty to be applied in the final calculation of system reactivity. An administrative safety margin of 0.05 $\Delta k$ is currently applied as a design acceptance criterion in criticality calculations for all waste forms; administrative safety margins shall be justified as required by ANSI/ANS-8.1.

- An Integrated Safety Analysis is performed to provide a comprehensive systematic review of facility hazards, including criticality, to confirm the adequacy of the selected

means of criticality control in each area. Adherence to criticality safety principles must be demonstrated, such as the "double contingency principle" stipulated by ANSI/ ANS-8.1, which states that "process designs shall incorporate sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident can occur." Compliance with the double contingency can be demonstrated by identifying specific process conditions on which reliance is placed. However, the Integrated Safety Analysis shall also demonstrate sufficient redundancy and diversity in MGR site and design features such that a preclosure criticality event is quantitatively determined to be beyond design basis, or "not credible." Common mode failures are appropriately accounted for in the Integrated Safety Analysis.

### 6.1.4 Means of Criticality Control

Several means of criticality control are available. Controlled parameters available for criticality control of activities involving significant quantities of fissionable materials include the following:

- Geometry
- Mass
- Density
- Isotopics
- Reflection
- Moderation
- Interaction
- Neutron Absorber (e.g., boron)
- Volume

Controlled parameters and feasible techniques for controlling them are established in such a manner as to minimize the risks from an inadvertent criticality. Geometry control constitutes the preferred controlled parameter with fixed neutron absorbers employed as necessary. This preference is consistent with the general guidance applied in preclosure waste form storage and handling design at the MGR, which is to employ, where practicable, reliance on equipment design that uses passive-engineered controls rather than on administrative controls. Techniques for criticality control, listed in order of hierarchical preference, are further identified and described as follows:

A. Passive-Engineered Controls–Controls that employ fixed design features or devices to preclude inadvertent criticality in operations. No human intervention is required except maintenance and inspection.

B. Active-Engineered Controls–Controls that utilize active hardware to sense conditions and automatically place a system in a safe condition. Actuation and operation of these controls require no human intervention.

C. Augmented Administrative Controls–Controls that rely on human judgement, training and actions for implementation, but employ active warning devices (audible and

visual) that prompt specific human actions to occur before the process can proceed to augment the implementation of the controls.

D. Simple Administrative Controls–Controls that rely solely on human judgement, training, and actions for implementation.

The preclosure criticality safety strategy also relies on a defense-in-depth approach. The first aspect of defense-in-depth involves taking advantage of the many natural features of the site and engineered features of the MGR surface facilities, emplacement drifts and waste package designs that contribute to reducing risks associated with criticality at the MGR during preclosure operations. A second aspect of defense-in-depth is the systematic use of conservative assumptions and parameters in criticality safety calculations supporting the design of preclosure facility features, including waste packages, canister staging racks for dry storage, and fuel assembly baskets and the basket staging rack in the fuel assembly staging pool. Examples of how defense-in-depth is provided in the MGR criticality design approach are provided as follows:

- The selection of a site where surface facilities can be placed above the flood plane significantly reduces criticality risks associated with dry handling and loading operations performed at the MGR. Although dry handling facilities can be demonstrated to accommodate water intrusion by conservatively assuming the presence of pure water in criticality safety calculations, eliminating water intrusion mechanisms, such as external flooding, significantly reduces the risk of criticality associated with these areas of the MGR surface facilities.

- In order for a criticality event to occur during operations involving the waste package, multiple changes in process conditions (e.g., waste package breach, water intrusion and retention, removal of neutron absorbers inside waste package) must occur. The multiple liner barriers, neutron-absorbing materials in the basket, and the iron in the basket materials, contribute to reduce risk and protect against the occurrence of a criticality event involving a waste package. The use of two separate waste package shell materials and the welded closure system prevents entry of water into the waste package during preclosure.

Many of the natural site features and engineered features incorporated into the MGR surface facilities, emplacement drifts, and the waste package designs will tend to reduce the probability of a preclosure criticality event. Many of these defense-in-depth features shall be identified and credited in the Integrated Safety Analysis when demonstrating that preclosure criticality constitutes a BDBE.

### 6.1.5 Quality Assurance and Safety Classification Philosophy

The classification of items important to safety and waste isolation at the MGR is based on three different levels of quality defined in QAP-2-3, Rev. 10, "Classification of Permanent Items." QL-1 is used to describe high safety significant or waste isolation significant items that have a direct impact on safety. QL-2 is used to describe low safety significant or waste isolation significant items that could have limited or indirect impact on safety. QL-3 is used to describe

those items that do not meet definitions of QL-1 or QL-2, but may provide public or worker safety benefits. SSCs identified as providing direct criticality control functions are designated as QL-1, which provides for the highest level of Quality Control Program oversight. Criticality control related SSCs shall be designed and procured to ensure performance consistent with facility design basis documentation, including environmental qualification criteria (i.e., operating environment characteristics such as temperature, incident radiation, or response to seismic events). Items identified as QL-1 due to criticality control functional requirements shall be reviewed by criticality safety engineers for input on design and procurement criteria, and incorporated into the criticality safety training program.

### 6.1.6 Criticality Safety Management Measures

In addition to a conservative design approach, facility management measures are implemented to minimize criticality related risks during MGR preclosure operations. Facility management and workers are trained to recognize the importance of procedure adherence and the interdependence of the facility design and procedural controls in ensuring operational safety.

### 6.1.7 Functions and Responsibilities

Functions and responsibilities related to criticality safety management measures are identified as follows:

- Management and Supervision–Accepts responsibility for the safety of operations, is knowledgeable in aspects of criticality safety relevant to project activities, establishes a criticality safety policy, assigns authority and responsibility to the line organization for criticality safety, and promotes and authorizes criticality safety training for the line organization.

- Engineers–Provide technical guidance and perform evaluations. Analyses are performed to demonstrate the sub-criticality of fissionable material storage and handling processes under normal and credible abnormal conditions. Document the interdependence between design of the facility and procedural controls and limits. Maintain familiarity with standards, guides, and codes; with operations requiring criticality safety controls; and with reports of violations and other deficiencies for improvements in safety practices and procedures.

- Workers–Perform work in accordance with approved procedures. Maintain criticality safety training and understand technical bases of limits and controls. Promptly report deviations from procedures or unforeseen changes in process conditions affecting criticality safety, and assist with assessments to ensure that procedures are followed.

- Procedures–Organized for convenient use by operators and free of extraneous material. Procedures should include limits and controls significant to criticality safety during specific operations as well as provide the bases for these limits and controls. Procedures should be reviewed and revised as improvements become desirable, reviewed by engineers and supervision before use, and supplemented by checklists or flow sheets. Fire-fighting instructions regarding fissionable materials should be provided in the

procedures. These instructions should include explicit guidance as to whether water can be used to fight the fire.

- Materials Control–Control the movement of fissionable materials. Ensure that fissionable material is labeled as required by procedural controls; methods are established to monitor the presence and effectiveness of credited neutron absorbers; and control of fissionable material is maintained to ensure sub-criticality under normal and credible accident conditions.

- Audits and Inspections–Operations shall be reviewed to confirm that procedures are being followed and that process conditions have not been altered so as to affect nuclear criticality safety bases.

- Emergency Planning–Prepare and maintain approved emergency response procedures; designate personnel assembly points and evacuation routes; train personnel and any contract workers in emergency response; exercise the plan; pre-arrange for medical care and treatment of injured and exposed personnel; and provide provisions for assessing exposure and contamination of personnel at assembly areas.

- Posting Requirements–Appropriate material labeling and area postings shall be maintained specifying material identification and limits on selected parameters that are subject to procedural control.

## 6.2 SITE FACTORS AFFECTING CRITICALITY CONTROL AND SAFETY PROGRAMS

The Yucca Mountain site characteristics are reviewed for compatibility with the criticality design approach and the proposed criticality safety program described in this report. The range of site environmental conditions anticipated as Category 1 design events (i.e., normal conditions and moderately frequent design events with frequencies of occurrence as low as 1E-2 events/year) shall be accommodated by the criticality design and safety programs. The criticality design and safety program shall also consider external hazards (e.g., floods, high winds, and earthquakes) associated with the Yucca Mountain site as initiating events such that criticality safety can be ensured by design and through implementation of the criticality safety program. As discussed in Section 6.2.2, natural site factors and engineered design features shall be demonstrated to collectively reduce the probability of a preclosure criticality event to less than credible levels.

### 6.2.1 Site Characteristics

Facility design features providing criticality control functions shall be designed and procured to ensure performance under the full range of environmental conditions anticipated over the preclosure operational lifetime. Yucca Mountain site-specific environmental characteristics shall be reviewed to demonstrate the adequacy of the preclosure facilities criticality safety design and operations approach. Items relied upon for criticality control shall be designed, procured and maintained consistent with equipment qualification requirements written to encompass anticipated environmental conditions of operation, including extreme conditions for

characteristics such as temperature, insolation, and humidity. In addition, the criticality control design approach shall minimize reliance on active systems, such as pool water cooling systems, in order to minimize the potential impact of environmental conditions on facility operations.

### 6.2.2 Site-Related External Hazards

Externally initiated events that could impact criticality control design features or otherwise invalidate analysis assumptions are reviewed to support site recommendation. Items relied upon for criticality control shall be designed to withstand design basis external events associated with the Yucca Mountain site. For example, criticality control features designed to provide geometry control are designed and procured to ensure geometry is maintained in the event of design basis seismic events. Items required to protect criticality control features from external hazards, such as tornado missiles, shall also be treated and identified as important to safety in order to ensure design, procurement, construction and maintenance activities are conducted in a manner commensurate with the item's importance to safety.

## 6.3 CONCLUSIONS

The criticality safety strategy for the MGR incorporates both permanent design features and a criticality safety program to ensure criticality safety. Items relied upon to provide preclosure criticality control functions have been identified and site characteristics and external hazards associated with the Yucca Mountain site have been reviewed. The ability to design, procure, install and maintain those criticality control related SSCs in a manner that ensures reliability under the full range of design conditions applicable to the MGR site will be addressed. In addition, the ability to implement the criticality safety program at the Yucca Mountain Site has also been addressed.

## 6.4 REFERENCES

### 6.4.1 Documents Cited

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Lichtenwalter, J.J.; Bowman, S.M.; DeHart, M.D.; and Hopper, C.M. 1997. *Criticality Benchmark Guide for Light Water Reactor Fuel in Transportation and Storage Packages.* NUREG/CR-6361. Oak Ridge, Tennessee: Oak Ridge National Laboratory. TIC: 233099.

NRC (U.S. Nuclear Regulatory Commission) 1999. *Draft of the Standard Review Plan for the Review of a License Application for Fuel Cycle Facility.* NUREG-1520. Washington, D.C.: U.S. Nuclear Regulatory Commission. TIC: 245198.

### 6.4.2 Codes, Standards, Regulations, and Procedures

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# 7. RADIATION PROTECTION

Preclosure radiological protection general design criteria are provided for the MGR at a high level in the *Revised Interim Guidance Pending Issuance of New U. S. Nuclear Regulatory Commission (NRC) Regulations (Revision 01, July 22, 1999), For Yucca Mountain, Nevada* (Dyer 1999), Section 111. This section states that the repository operations area shall be designed to maintain radiation doses, levels, and concentrations of radioactive material in air in restricted areas within the limits specified in 10 CFR Part 20. 10 CFR Part 20, *Standards for Protection Against Radiation*, states that the licensees should make every reasonable effort to maintain exposures to radiation as far below the limits specified in Part 20 as is reasonably achievable. ALARA means making every reasonable effort to maintain exposures to radiation as far below the established dose limits in 10 CFR Part 20 as is practical taking into account the state of technology, and the economics of improvements in relation to benefits to worker safety, public health and safety, and other societal and socioeconomic considerations.

In response to radiation protection regulations, a comprehensive MGR operational radiation protection program will be developed for use during the operational phase of the MGR, to ensure compliance with 10 CFR Part 20. This MGR operational radiation protection program, including the Design ALARA plan (to be developed) will be submitted as part of the License Application, and will be revised as necessary upon receipt of the NRC operating license for the MGR, to ensure compliance with 10 CFR Part 20 and any applicable license conditions.

This section provides a methodology for implementation of Design ALARA considerations. The application of the ALARA design philosophy is incorporated into the MGR preclosure facilities to satisfy the regulatory requirements. This includes descriptions of:

- Design ALARA requirements and guidelines
- General work processes performed during the design phase
- Design ALARA functions and products.

## 7.1 GENERAL DESIGN ALARA PROGRAM DESCRIPTION

### 7.1.1 Design ALARA General Process

Radiation protection in the design phase is to be implemented in a multi-faceted manner. Design engineers are responsible for evaluation and implementation of the radiation protection features into the design to ensure that public and occupational doses are ALARA. This Design ALARA program is provided to support these efforts. The basic elements of a sound Design ALARA program are:

- Policy
- Management commitments to the program
- Engineering procedures
- ALARA training

- Radiation Protection Design ALARA specialists oversight
- Design ALARA analysis
- Specification of radiation equipment, instrumentation, and facilities
- Periodic assessments
- Preliminary technical specifications for implementation of the radiation protection limits and controls for operations.

Engineering procedures are developed to implement ALARA processes into the facility design. Design engineers are trained to the procedures and in the areas of basic radiation physics, Design ALARA measures and major plant radiation sources. Design ALARA engineers provide guidance to design personnel through design product reviews. Design ALARA personnel perform occupational dose assessments of preliminary designs to focus on systems and operations that produce significant potential occupational exposures. Additional emphasis is placed on high dose activities in a graded approach to evaluate design options. An ALARA review committee can be utilized to focus attention and resolve interfacing problems. The design is optimized to meet ALARA objectives. Lastly, administrative controls are defined that rely on operations for assurance of meeting radiation protection requirements. These controls are identified and carried forward in the design and licensing documentation. Final system design is evaluated against performance criteria to ensure compliance. Performance criteria can be in the form of cumulative occupational dose, individual dose, and person-rem savings.

Fixed radiation monitoring equipment is needed to satisfy operational radiation protection criteria. This equipment is identified during the design phase to provide the necessary radiation protection information during plant operations. Basic equipment includes area radiation monitoring for areas of access that have the potential for changing radiological conditions. Also included are ventilation system process and effluent monitors and/or samplers to ensure compliance with offsite dose criteria. Additional portable monitoring equipment is used to supplement fixed monitoring systems. Locations for area monitors and/or samplers are based on dose assessment uncertainties and the potential for unplanned exposures. Locations of effluent monitors are for potential release points from the facility. Emphasis will be placed on minimizing the number of release points to minimize monitoring cost and compliance complexity.

Examples of operating controls that are identified during the design phase to ensure plant ALARA safety criteria are also identified. This typically includes such controls as locked access controls to restricted areas, automated monitoring and sampling prior to discharge of effluents, alarm indications and automatic release termination, and operational interlocks. Radiation protection facilities, including change rooms, portable equipment storage, counting equipment, and adequate office space for radiation protection personnel, are specified by design to support ALARA operations.

### 7.1.2 Design Management Policy

The Design ALARA Program describes a management policy on radiation protection and the policy will include management commitment to:

- Ensure that each supervisor implements his or her responsibilities to integrate appropriate radiation protection controls into design activities.
- Ensure that each individual responsible for the design of the facility understands and accepts the responsibility to follow Design ALARA processes.
- Understand regulatory requirements, radiation dose limits, and design and operational controls that minimize occupational radiation exposures.
- Maintain a comprehensive radiation protection program during the design phase to support satisfying ALARA goals for expected radiation exposure to workers during the operational phase.

The MGR preclosure facilities design ALARA policy will commit to a process that ensures compliance with the intent of Revised Interim Guidance Section 111 (Dyer 1999) and 10 CFR 20.1101.

### 7.1.3 Program Functions and Responsibilities

The responsibility and authority for implementing the Design ALARA program will be assigned to an individual (or committee) with organizational freedom to ensure its development and implementation. The following summarizes the basic responsibilities of a design ALARA program:

The management staff is responsible for:

- Ensuring the implementation of the design ALARA program policy
- Conducting periodic program reviews
- Providing budget and resources to perform the work
- Supporting design ALARA decisions
- Supporting establishment of a Quality Assurance program that identifies ALARA objectives
- Establishing the definition of the classification of the system SSCs as important to radiation protection safety
- Supporting development of design ALARA implementing engineering procedures.

Personnel qualified in Design ALARA (called Design ALARA personnel for the remainder of this section) are responsible for the development and administration of the overall Design ALARA program. These personnel specialize in the areas of radiation protection design and operations, such as:

- Coordinating the development of design ALARA implementing procedures
- Coordinating ALARA training for the design groups
- Reviewing relevant design documents and provide assessments and feedback
- Supporting or establishing a graded approach to ALARA design
- Supporting classification of SSCs classification for radiation protection
- Providing radiation protection design criteria support including dose goals and apportionment among the design elements
- Ensuring a consistent level of performance
- Preparing preliminary and final dose assessments
- Preparing Design ALARA guidelines
- Developing design ALARA implementation mechanisms, i.e., cost-benefit analyses, design optimization methods, etc.
- Supporting radiation monitoring requirements
- Preparing radiation zone drawings.

The individual design organizations are responsible for:

- Incorporating ALARA design criteria into engineering procedures
- Participating in the ALARA training program
- Incorporating ALARA principles and policy guidelines into the facility design
- Providing draft design documents to the Design ALARA personnel for feedback
- Reviewing ALARA policy and guidelines
- Incorporating Design ALARA personnel feedback on ALARA design features
- Supporting the definition of the classification of the system SSCs as important to safety
- Implementing ALARA design
- Documenting the ALARA evaluations and results
- Supporting the design of the radiation monitoring program
- Developing auditable records of incorporation of ALARA into the design.

## 7.2 DESIGN ALARA PRODUCTS

### 7.2.1 Training

MGR management and affected groups will be committed to a design of a facility that can maintain radiation doses as far below the limits specified in 10 CFR Part 20 as is reasonably achievable. Training is a key component to this commitment. Design personnel will be provided with training and guidance for minimizing the potential for radiation exposure as part of their design responsibility. Personnel qualified in Design ALARA will have lead responsibilities in developing training programs and guides that ensure compliance with 10 CFR Part 20 and utilize the applicable guidance in NRC Regulatory Guide 8.8. Regulatory Guide 8.8 provides information relevant to attaining goals and objectives for planning, designing and constructing a facility to meet the regulatory requirement that doses to personnel during operation will be ALARA. Doses during maintenance, abnormal occurrences and accident events and recovery actions are included within the scope of design ALARA training and guidance.

MGR Design ALARA programs will include both a review of the fundamental concepts employed in facility radiation control, and consideration of proven methods and new technology which may be applicable to facility design and modifications. A formal training program is developed and administered to employees requiring training on an as needed basis with basic training provided for new hires and transferred employees as needed. The training program will emphasize design ALARA issues based on operating experience as appropriate.

### 7.2.2 ALARA Design

An ALARA design procedure will be developed and used for the design of the facilities and the Design ALARA program. The following ALARA topics are to be considered.

*Documentation of the DOE and M&O MGR management commitments to ALARA during the design phase.* Basic elements of this commitment are the appropriation of qualified resources to fulfill the functions of the program and support for the development and implementation of procedures, goals, and policies.

*References to, or citation of, engineering procedures in support of implementation of ALARA into the design.* Additional procedures that may be developed include a procedure for the overall implementation of ALARA into the design process, in addition to calculations and analysis, Quality Assurance classifications, a dose assessment procedure and a cost-benefit procedure, and other design implementing procedures,, specifications, and drawings.

*Guidelines for design personnel to minimize the potential for radiation exposure as part of the design responsibility.* Typical items may be ALARA goals, ALARA training, design product reviews, dose assessments, and the cost of radiation dose for cost-benefit analysis.

*Preliminary radiation zoning information to determine exposures associated with each facility and each work activity.* For areas of high radiation, special cost-effective design features will be considered to reduce the time spent in the areas and reduce the source of exposure. Facility design features that will support the Design ALARA program include laboratories, counting

rooms, decontamination stations, equipment storage, and means for access control. Since work performed in radiation areas has the potential for occupational radiation exposure, significant manual operations performed in radiation areas will be identified and analyzed. Emphasis will be placed on reducing doses using facility design as opposed to operating limits and controls. Evaluations will be performed to determine the effectiveness of remote operations and the relocation of SSCs to areas outside of the radiation areas. Doses during maintenance, abnormal occurrences, accident events, and recovery actions will be included within the scope of design ALARA training and guidance.

*Dose assessment examples for guidance.* The dose assessment is a compilation of the dose rate and access information to generate an estimate of the dose involved in the operation and maintenance of SSCs. See section 7.2.3 for more information.

*Description of the bases for performing a cost-benefit analysis.* Performance of an adequate cost-benefit analysis to ensure optimization of dose reduction for each design area and activity will be discussed.

### 7.2.3 Dose Assessment

Design dose assessments are performed to aid in understanding the radiation field and work to be performed. The dose assessment combines estimated dose rates in the work areas, the time to perform this work, the number of personnel performing the work, the type of personnel performing the work and the annual frequency of the work to be performed. This provides the expected exposure to the individual in the work group and the total group exposure. Both of these values can then be evaluated to determine design ALARA effectiveness. NRC Regulatory Guide 8.19 provides a basic outline of the dose assessment method.

The design group uses this information to evaluate where changes to the design may be made to reduce dose in a cost-effective manner. As design progresses the dose assessment may be modified to incorporate design changes and incorporate better data on labor, layout and equipment requirements.

Dose assessments will be performed for work in radiation areas including normal operations, Category 1 events as defined in Section 2 of Dyer (1999), and abnormal situation recovery actions. These dose assessments are also useful in evaluation of decommissioning options.

### 7.2.4 Cost-Benefit Analysis

The purpose of a cost-benefit analysis is to assess the following:

- Benefits of possible radiation exposure reduction techniques
- Cost to implement each proposed design feature.

A cost of personnel exposure can be defined to support this decision. Current industry practice is to establish a value of "dollars per person-rem" saved as an indicator of ALARA benefit. For example, the NRC adopted the value of $2000 per person-rem subject to present worth considerations (NRC 1995). That is, if the cost of one design option over another or the modification to a baseline design costs less than this value, then it is recommended. Costs

greater than the threshold would recommend against making the change. A method for developing a cost-benefit analysis is given in *Determining Effectiveness of ALARA Design and Operational Features* (Hall et al. 1979) and "*Reassessment of NRC's Dollar Per Person-Rem Conversion Factor Policy*" (NRC 1995).

Many ALARA design changes will also enhance system performance in a way that there is no cost versus benefit analysis required. ALARA evaluations are documented to demonstrate compliance with 10 CFR Part 20.

### 7.2.5 Design Reviews

Since the design of a facility is the incorporation of many different criteria from many design groups; potentially affected groups review design products. Design concepts and features represented in descriptions, drawings, and specifications will reflect radiation protection considerations of the facility, operations, maintenance, processing of radioactive wastes, effluent control and decontamination. Operations include normal, Category 1 and recovery actions. Radiation protection aspects of decommissioning for the preclosure design will be factored into planning, designing, construction, and modification activities.

The MGR design organization is made aware of potential SSCs that are important to safety for public and occupational exposure by such methods as System Description Documents and the classification analysis of SSCs. Regular feedback between the design groups and Design ALARA staff will also bring out the importance to ALARA design features.

### 7.2.6 Facility and Equipment Design Features

Radiation sources within the MGR facilities will differ with respect to location, intensity, and characteristics. The magnitude of the dose rates that result from these sources is dependent on many factors, including the facility and equipment design, layout, mode and length of operation, and radiation source strength and characteristics. The following ALARA program features or functions can be found in NRC Regulatory Guide 8.8, Section C.2:

- Access Control of Radiation Areas
- Radiation Shields and Geometry
- Process Instrumentation and Controls
- Control of Airborne Contaminants and Gaseous Radiation Sources
- Isolation and Decontamination
- Radiation Monitoring Systems.

An example for determining the effectiveness of the design and operational features of NRC Regulatory Guide 8.8, would be to utilize the report, *Determining Effectiveness of ALARA Design and Operational Features* (Hall et al. 1979). That report presents and demonstrates a method developed by United Nuclear Industries, Inc., for determining whether occupational radiation exposures in nuclear power reactors are ALARA. The methods discussed in the report for determining occupational radiation exposures could be helpful when applied to the design of SSCs in the MGR preclosure facilities.

The design of radiation protection facilities, instrumentation, and equipment will be consistent with size and quantities of equipment needed for normal facility operations and abnormal situation that may require supplemental workers and extensive work in high radiation areas. MGR preclosure facility design features will reflect the following considerations that are detailed in NRC Regulatory Guide 8.8, Section C.4:

- Counting Room(s)–A low-radiation background counting room is needed to perform routine analyses on facility samples containing radioactive material collected from air, water, surfaces, and other sources.

- Portable Instruments–Portable instruments are needed for measuring dose rates and radiation characteristics.

- Personnel Protective Equipment–Personnel protective equipment will include anticontamination clothing and equipment (e.g., per requirements of ANSI Z88.2) and respiratory equipment (e.g., per NRC Regulatory Guide 8.15).

- Support Facilities–Design features of support facilities will include consideration of portable calibration areas, personnel decontamination areas, change rooms, sufficient office space, etc., as detailed in NRC Regulatory Guide 8.8, Section C.4.e.

## 7.3 REFERENCES

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### 7.3.2 Codes, Standards, Regulations, and Procedures

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# 8. FIRE PROTECTION

This section describes the MGR preclosure fire safety strategy which considers Yucca Mountain site characteristics, and natural and human-induced events and processes and provides an overview of the proposed MGR preclosure fire safety strategy that is relevant to a site suitability determination as follows:

- Major regulatory and industry guidelines and design requirements for fire protection
- Fire hazards analyses performed during the design phase to define fire protection requirements and support of an Integrated Safety Analysis
- General and MGR-specific fire protection design criteria for supporting performance goals of Section 111 of Dyer (1999)
- Description of the proposed MGR fire safety and protection program for achieving radiological and non-radiological safety goals.

The preclosure fire safety strategy addresses potential impacts that Yucca Mountain site characteristics (i.e., geology, hydrology, climatology, and meteorology), including severe natural and human-induced events, could have on design and procurement of SSCs required for fire protection.

Fire protection analysis is an integral part of the design process for the MGR and is updated as the design evolves. Examples of design support through radiological and fire hazards analyses performed to date are described.

## 8.1 DESCRIPTION OF THE MGR PRECLOSURE FIRE SAFETY STRATEGY

The fire safety strategy for preclosure operations has two principal safety objectives: (1) radiological safety of the public and workers, and (2) non-radiological personnel safety and health. The first objective is addressed as part of the Integrated Safety Analysis required by Dyer (1999) and by implementing applicable fire-safety guidance provided by the NRC for other nuclear facilities. The second objective is addressed by adopting applicable codes and standards for fire protection programs. Compliance with regulatory requirements and adherence to fire protection codes and standards will ensure that fires are prevented or mitigated before they pose a hazard involving a release of radioactive material.

### 8.1.1 MGR Facility

There are six basic SNF handling operations identified for the MGR that may affect preclosure fire safety.

A. Operations in the carrier preparation system
B. Operations in the carrier/cask handling system
C. Operations in the ATS
D. Operations in the CTS
E. Operations in the DCHS
F. Emplacement of waste packages and subsequent storage in Yucca Mountain.

In addition, there will be concurrent construction (development) of subsurface facilities during part of the emplacement operations. Provisions will be made, also, for operations associated with waste package retrieval and remediation.

Operations A—E and waste package remediation are carried out in the surface facilities and Operation F and waste package retrieval are carried out primarily in the subsurface facilities. The WTB, which is part of the surface facilities, will contain combustible material that could initiate a fire that could potentially release radioactive material during a fire if not prevented by design. Fire hazards analyses will be developed for the respective facilities and a fire protection program will be developed that meets the regulations and standards for surface and subsurface operations and fire protection programs. The fire protection program will be included as part of the Integrated Safety Analysis for the MGR. The fire protection program will address issues of personnel and facility safety in addition to potential radiological consequences associated with potential fires.

The potential for fire-initiated radiological hazards are essentially eliminated for Operational Areas A and B because the incoming transport casks are licensed in accordance with 10 CFR Part 71 that requires the casks to withstand a severe fire environment without loss of containment. Similarly, the fire-initiated radiological hazards are essentially eliminated for Operational Areas E and F because the waste packages will be designed to withstand the fire environment specified in 10 CFR Part 71.

The design of the surface facility will ensure that an adequate supply of fire suppressant materials is available to mitigate the worst-case fire scenario. For example, an assessment will include a list of water based automatic suppression systems and their maximum demands and will establish facility firewater requirements including capacity, pressure, and duration requirements. A water storage tank having sufficient capacity will be maintained with the same water sources that will support repository construction.

### 8.1.2 Regulatory Requirements and Industry Codes and Standards

#### 8.1.2.1 Regulatory Requirements

It is anticipated that 10 CFR Part 63 (currently in draft form) (64 FR 8640) will be the governing regulation for the MGR. In anticipation of the issuance of 10 CFR Part 63, DOE has identified interim guidance (Dyer 1999) based on the draft of 10 CFR Part 63. This guidance is risk-informed and performance-based. As such, it does not include prescriptive design criteria for fire protection features. Instead, fire protection measures associated with radiological safety are to be evaluated as part of the Integrated Safety Analysis that will be included as part of the License Application. A primary goal of the Integrated Safety Analysis is to ensure that comprehensive hazards and accident sequence analyses are performed to demonstrate that the performance requirements of Section 111 of Dyer (1999) are met, including potential fire-initiated events. As described in the proposed regulations in Section 112 of Dyer (1999):

> *The integrated safety analysis of the geologic repository operations area shall include: (e) an analysis of the performance of the major design structures, systems and components, both surface and subsurface, to identify those that are important to safety, including identification and description of controls that are*

*relied on to limit or prevent potential design basis events or mitigate their consequences, and including identification of measures taken to ensure the availability of identified safety systems.*

Nevertheless, the MGR design and concept of operations are cognizant of the prescriptive design and operational requirements and precedents for fire protection that have been established for other nuclear facilities, as described in Subsection 8.1.2.2. Those requirements will be applied, as deemed appropriate, to the MGR. The MGR Compliance Program identifies NRC regulatory guidance, and NRC-endorsed industry codes and standards that are potentially applicable to each system, including those for fire protection. As appropriate, design criteria for fire protection identified in the compliance program guidance packages for quality-affecting systems are incorporated into the System Description Documents for the respective systems. The application of these regulatory precedents and industry codes and standards will serve, in large measure, to prevent the initiation and propagation of fires.

#### 8.1.2.2 Regulatory Guides and NUREG Reports

NRC Standard Review Plans for various facilities and several NRC Regulatory Guides were used as guidance to develop the preclosure fire safety strategy.

Recent NRC guidance applicable to fuel storage facilities licensed under 10 CFR Part 72 are considered to be generally applicable to the design and operation of surface facilities, which includes both wet and dry assembly and canister handling systems. System Description Documents for MGR systems have incorporated applicable NRC guidance for 10 CFR Part 72 facilities. Fire protection guidance applicable to dry storage systems is provided in Draft NUREG-1567, *Standard Review Plan for Spent Fuel Dry Storage Facilities* (NRC 2000). This Standard Review Plan incorporates by reference material from other NRC and DOE documents that include:

- NRC Regulatory Guide 3.48, *Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation or Monitored Retrievable Storage Installation (Dry Storage)*
- NRC Regulatory Guide 3.38, *General Fire Protection Guide for Fuel Reprocessing Plants*
- NUREG-0800, Section 9.5.1, Fire Protection Program (NRC 1987)
- DOE Order 420.1, *Facility Safety*
- DOE G-420.1/B-0, *Implementation Guide for Use with DOE Orders 420.1 and 440.1 Fire Safety Program.*

For the MGR, fires are to be considered in the context of the Integrated Safety Analysis. Applicable guidance for integrating the fires analysis into the Integrated Safety Analysis is derived from NUREG-1520 (NRC 1999c), NUREG-1701 (NRC 1999a), and NUREG-1702 (NRC 1999b).

#### 8.1.2.3 Industry Standards

Numerous industry standards are considered to be applicable to the fire safety and protection program, primarily from the National Fire Protection Association and from the American National Standards Institute/American Nuclear Society. Standards endorsed by the NRC in standard review plans and regulatory guides will be followed, as deemed applicable to the MGR operations areas.

### 8.1.3 Quality Assurance and Safety Classification Philosophy

The classification of items important to safety and waste isolation at the MGR is based on three levels of quality defined in QAP-2-3, "Classification of Permanent Items." The definitions of the quality levels were presented previously in Section 4.4.1. SSCs associated with fire detection and suppression systems are classified as QL-2 if they provide fire protection functions for SSCs designated as QL-1. Otherwise, SSCs that provide fire protection for workers and facilities are classified as conventional quality.

Fire protection SSCs classified as QL-2 will be designed and procured to ensure performance consistent with facility design basis documentation, including environmental qualification criteria (i.e., operating environment characteristics such as temperature, humidity, incident radiation, or conditions following a DBE, including seismic events). The fire protection functional requirements will be reviewed by fire protection engineers to develop design and procurement criteria.

### 8.1.4 Fire Safety and Protection Program

A comprehensive fire safety and protection program will be developed to minimize the fire-related risks for MGR preclosure operations. The fire safety and protection program will demonstrate to the NRC the adequacy of the following areas of fire protection:

A. Fire Hazards Analysis (FHA): a systematic analysis of the fire hazards, identification of specific areas involving QL-1 SSCs, development of design basis fire scenarios,. evaluation of anticipated consequences, and determination of the adequacy of facility fire safety

B. Fire Protection Program: addresses design and operational features such as construction features; passive fire-rated barriers; process and operational features; fire detection and alarm systems; fire suppression systems and equipment; supply of water and other materials for fire suppression and back-up capability; design-basis documents; inspection, maintenance, and testing of fire protection features and systems; and requirements for manual fire-fighting capability

C. Organization and Conduct of Operations: establishes and maintains the organization and management, training and qualifications, fire prevention, engineering review of design changes, Quality Assurance, and documentation and record keeping.

The following subsections describe the content of each area.

#### 8.1.4.1 Fire Hazards Analysis

An FHA will be performed to document specific fire hazards, fire protection features proposed to control those hazards, and the overall adequacy of facility fire safety. The FHA consists of a systematic analysis of the fire hazards, an identification of specific areas involving QL-1 SSCs, the development of design-basis fire scenarios, an evaluation of anticipated consequences, and a determination of the adequacy of facility fire safety. The FHA will be included in the Integrated Safety Analysis.

Guidance for performing an FHA has been adapted from DOE G-420.1/B-0, *Implementation Guide for Use with DOE Orders 420.1 and 440.1 Fire Safety Program*, but addresses the specific needs for the MGR and the performance requirements of Section 111 of Dyer (1999). The following describes the contents of the FHA.

Although a detailed FHA can be performed only at an advanced state of design evolution, a preliminary FHA will be performed for the MGR early in the design phase to ensure incorporation of an acceptable level of protection in the evolving design. The FHA will be performed under the direction of a qualified fire protection engineer, with support from chemical, electrical, mechanical, criticality-safety, radiation protection, and systems engineers, and other cognitive staff from the respective surface and subsurface facility engineering staffs, as needed.

The FHA will contain, but not be limited to, a conservative assessment of the following items and safety issues:

- Descriptions of construction (type); fire hazards; fire protection features; and operations (or concept of operations)
- Potential for a radiation or toxic incident from a fire
- Impact of natural hazards (earthquake, flood, or wind) on fire safety
- Protection of QL-1 SSCs
- Life safety considerations
- Emergency planning
- Fire department/brigade response
- Maximum possible fire loss
- Security and safeguards considerations related to fire protection
- Exposure fire potential and the potential for fire spread between two fire areas.

The FHA will be organized according to the individual fire areas that comprise operational areas of the respective surface and subsurface facilities. A fire area is a location bounded by fire-rated construction, having a minimum fire resistance rating of 2 hours. The FHA through fire loading analysis and fire modeling (if necessary) will document that the fire ratings are appropriate for each fire area boundary. The FHA will contain an inventory of QL-1 SSCs that are susceptible to fire damage within each sub-area. Loss of systems such as ventilation, cooling, or electrical power that could cause failures elsewhere in the facility will be evaluated. When the facility design is sufficiently advanced, the FHA will also consider the improper operation of equipment due to spurious signals induced by fire damage and other potential interactions such as the effects of combustion products, manual fire-fighting efforts, and the activation of automatic fire suppression systems.

If potential radiological events are identified, the consequence analyses may need to produce fire-related parameters (temperatures, pressures, and air velocities) for evaluating radioactive material dispersion through the facility air distribution system as a result of fire. The radiological consequences will then be determined as part of the Integrated Safety Analysis.

The quantity and associated hazards of flammable and combustible material expected to be found within the fire area will be factored into the analyses. Consideration will also be given to the presence of transient combustibles associated with maintenance activities and storage. Average combustible loading, by itself, will not be used to estimate fire area fire severity. As a minimum, for each designated fire area, the following fire hazards will be evaluated for potential fire severity and consequent damage:

- Fire load from solid combustible materials (both quantity and configuration);
- Flammable and combustible liquids and gases used in the open processes within the fire area (quantities or flow rates);
- Process chemicals and materials (both quantity and location) that could present a toxic or radiological hazard, or that could significantly affect health or the quality of the environment through a release as a result of a fire emergency; and
- Potential ignition sources.

As appropriate, fire hazards associated with HEPA filters will be included in the analysis.

The FHA will support an assessment of the facility requirements for fire suppressant materials (water and non-water) including capacity, pressure, and duration requirements. The assessment will include a list of water- and non-water-based automatic suppression systems and their maximum demands, interior hose stream requirements, and exterior hydrant requirements. The FHA will also note where water should not be used as a fire suppressant because of potential criticality concerns. This assessment will support the design and layout of the facility fire water system layout, e.g., the locations and characteristics of pumps, lines, tanks, and sectionalizing valves. A water storage tank having sufficient capacity will be maintained with the same water sources that will be used during repository construction.

For each designated fire area that involves a QL-1 SSC, the FHA will provide input to the Integrated Safety Analysis regarding the postulated accident sequences caused or aggravated by fire. Either quantitative or qualitative methods may be used. Where quantitative analytical methods are used, input data and assumptions are documented. The FHA will define those fire protection systems and procedures that provide reasonable assurance that the defined consequences of an accident sequence will not occur or will be successfully mitigated. The proposed coverage of fire detection and suppression systems will be discussed for each fire area.

The FHA will be supported by a separate evaluation that demonstrates that the active portion(s) of the fire protection system that are classified as QL-2 has sufficient reliability and capacity to ensure that the likelihood of fire-initiated sequences are consistent with the frequency categorization of DBEs used in the Integrated Safety Analysis. The evaluation may be quantitative or qualitative. For example, a quantitative approach might employ a fault-tree model that includes redundant trains or components, human-errors, and common-cause failures to evaluate the probability that the system provide the required fire-suppression or detection capability. A qualitative approach might assume the consequences of a single, worst-case automatic fire-protection system malfunction during a fire and, if redundant automatic fire protection systems are provided in the area, only the system that causes the most vulnerable condition is assumed to fail.

Passive fire protection features, such as blank fire-rated walls or continuous fire-rated cable wraps are assumed to remain viable in accordance with their fire endurance rating to the extent that they are properly constructed and maintained.

#### 8.1.4.2 Fire Protection Program

The purpose of the Fire Protection Program (FPP) is to provide assurance, through a defense-in-depth approach, that a fire will not significantly increase the risk of radioactive releases to the environment in accordance with the performance requirements of Section 111 of Dyer (1999). Based on risk-informed insights from the Integrated Safety Analysis and FHA, that will be updated as design information is developed, the FPP will:

- Provide defense-in-depth that achieves balance among prevention, detection, containment, and suppression of fires
- Establish the fire protection policy for the protection of QL-1 SSCs and the procedures, equipment, and personnel required to implement the program at the site
- Comprise the fire detection and extinguishing systems and equipment, administrative controls and procedures, and trained personnel
- Ensure that the portions of the fire protection system classified as QL-2 have sufficient reliability and capacity (e.g., demonstrate that a single active failure will not prevent the system from protecting a QL-1 SSC).

As noted in Section 8.1.2.2, portions of NUREG-0800 (NRC 1987, Section 9.5.1) and NRC Regulatory Guide 3.38 may be applicable to the MGR contingent on the design of the installation and associated fire hazards. As appropriate, the national codes and standards cited in these NRC

guidance documents, in particular, applicable National Fire Protection Association codes and standards will be applied to the MGR.

The documentation of the FPP and the FHA (in support of the Integrated Safety Analysis) will provide sufficient material for the NRC to conduct a safety evaluation. The FPP description will include:

- Managerial and personnel positions that will be responsible for the design, and conduct of operations for implementation of the fire protection program and associated quality assurance
- Demonstration that the fire protection program is able to prevent or minimize radioactive releases to the environment from fire hazards in the facility
- Discussion of facility arrangement, buildings, and structural and compartmentation features which affect the methods used for fire protection, fire control, and control of hazards for the protection of SSCs, including consideration of criticality safety
- Demonstration that design criteria and bases for the detection and suppression systems for smoke, heat and flame control are in accordance with NRC guidelines and provide adequate protection for SSCs
- Requirements, as appropriate, for technical specifications (i.e., limiting conditions for operation and surveillance requirements) for fire protection systems.

The requirements of Dyer (1999) are performance based wherein principal design requirements are to be derived, in part, from the Integrated Safety Analysis. Nevertheless, consideration will be given to design principles from NRC precedents for other facilities and industry codes and standards (see Section 8.1.2). In particular, using guidance from NUREG-1567 (NRC 2000), which incorporates by reference various regulatory guides, standard review plans, industry standards, and DOE Orders. The principal design criteria of the MGR are expected to include requirements that address the following general issues:

- QL-1 SSCs and QL-2 fire protection systems will be designed and located so that they can continue to perform their safety functions effectively under credible fire and explosion exposure conditions.
- Non-combustible and heat resistant materials will be used wherever practical throughout the MGR particularly in locations vital to the maintenance of safety control functions.
- Detection, alarm, and suppression systems will have capability and sufficient capacity to minimize the adverse effects of fires and explosions on QL-1 SSCs.
- The design will provide protection against adverse effects that might result from the operation or failure of the fire suppression systems.
- The design of the control room or control areas will permit occupancy and actions to be taken to provide safe control under abnormal or accident conditions.

### 8.1.4.3 Organization and Conduct of Operations

Organizational charts and functional descriptions will define responsibilities for the FPP. The functional areas listed below will be specifically addressed in the FPP, although some positions and responsibilities may be combined as appropriate:

A. Management and Supervision

1. Upper level management responsible for the FPP.

2. Onsite or offsite management position(s) directly responsible for formulating, implementing, and periodically assessing the effectiveness of the FPP including fire drills and training conducted by the fire brigade and plant personnel; and reporting results with recommendations for improvements or corrective actions.

3. Onsite management position responsible for the overall administration of MGR operations and emergency plans which include the FPP and which provide a single point of control and contact for contingencies.

B. Operational Supervisors and Workers

1. Personnel responsible for periodic inspections to minimize the amount of combustibles; determine the effectiveness of housekeeping practices; and ensures the prompt and effective corrective actions are taken to correct conditions adverse to fire protection and preclude their recurrence.

2. Personnel responsible for fire-fighting training for operations personnel and the plant's fire brigade; design and selection of equipment; periodic inspection and testing of fire protection systems and equipment and determining the acceptability of the systems under test; conducting and evaluating fire drills to evaluate how well training objectives have been met.

3. Personnel responsible for review and evaluation of proposed work activities to identify potential transient fire loads.

4. Personnel responsible for indoctrination of MGR and contractor personnel in administrative procedures which implement the fire protection program, and the emergency procedures relative to fire protection; instruction of personnel on the proper handling of leaks or spills of flammable materials that are related to fire protection.

5. Position responsible for fire protection quality assurance.

6. Positions that are part of the MGR's fire brigade.

C. Administrative Controls

Administrative controls will be used to maintain the performance of the fire protection system and personnel. These controls will establish procedures to:

1. Prohibit the bulk storage of combustible materials in or near items important to safety (QL-1, -2, and –3)

2. Govern the handling and use of ordinary combustible materials, combustible and flammable gases and liquids, HEPA and charcoal filters, dry ion exchange resins, or other combustible supplies

3. Govern the handling of and limit transient fire loads in buildings containing items relied on for safety during all phases of operation, and especially during maintenance or modification operations; and control the removal from the area of waste, debris, scrap, oil spills, or other combustibles immediately upon completion of a work activity

4. Govern the use of ignition sources by use of a hot work permit system to control welding, flame cutting, brazing, or soldering operations

5. Govern leak testing

6. Conduct periodic housekeeping inspections

7. Control disarming of fire detection or fire suppression systems; and establish fire watches in areas where systems are disarmed

8. Test and maintain the fire protection equipment and emergency lighting and communication

9. Control actions to be taken by an individual discovering a fire

10. Control actions to be taken by control room operator in response to a fire alarm or notification; e.g., when to call for fire brigade assistance

11. Define strategies (supported by pre-fire plans) for fighting fires in areas presenting a hazard to QL-1 SSCs, of radiological exposure to personnel, or release of radioactivity

12. Establish a tracking program regarding emplacement and modification of fire barrier penetration seals that are defined in the Integrated Safety Analysis or FHA as items relied on for safety.

## 8.2 SITE FACTORS AFFECTING FIRE SAFETY AND PROTECTION PROGRAM

The Yucca Mountain site characteristics are reviewed for compatibility with the fire safety and protection program approach described in this report. The design bases for the MGR will provide reasonable assurance that no credible fire-initiated sequences can result in radioactive releases that exceed the performance requirements of Section 111 of Dyer (1999) for either Category 1 or Category 2 DBEs. The fire safety and protection program will consider external hazards (e.g., range fires, lightning, and earthquakes) associated with the Yucca Mountain site as potential initiating or exacerbating events such that fire safety can be ensured by design and

through implementation of the fire safety program. Further, the fire safety and protection program will ensure that the MGR site will have an adequate makeup supply and storage capacity for the fire suppression systems of the surface and subsurface facilities.

### 8.2.1 Site Characteristics

Facility design features providing fire protection functions will be designed and procured to ensure performance under the full range of environmental conditions anticipated over the preclosure operational lifetime. Yucca Mountain site-specific environmental characteristics that are anticipated to occur over the preclosure operational lifetime are described in Section 3. Fire protection SSCs that are classified as QL-2 will satisfy equipment qualification requirements written to encompass anticipated environmental conditions of operation, including extreme conditions for characteristics such as temperature, solar insolation, and humidity. Yucca Mountain site-specific environmental characteristics will be reviewed to demonstrate the adequacy of the preclosure facilities design and operations approach.

### 8.2.2 Site-Related External Hazards

External hazards associated with the Yucca Mountain site are identified through hazards analyses as described in Sections 5 and 8.3. Facility design features providing fire protection functions will be designed to withstand design basis external events associated with the Yucca Mountain site. For example, fire protection features will provide protection against credible range fires as appropriate to prevent any threat to QL-1 SSCs. Further, fire protection systems that protect or interact with QL-1 SSCs will withstand a design basis earthquake, lightning, design basis winds and tornadoes, rainfall, and flooding to the extent deemed necessary from the Integrated Safety Analysis and FHA.

## 8.3 PRELIMINARY FIRE HAZARDS ANALYSIS OF THE MGR OPERATIONS

The fires safety strategy will be applied to the MGR as the design evolves. At present, the principles of the strategy have been applied, in part, to the MGR design that supported the Viability Assessment. Preliminary radiological and fire hazards analyses were performed on the level of design information that was available. The preliminary radiological hazards analyses were tasked with identifying potential accidents that could result in a release of radioactivity to the public or to exposure of workers, including those initiated by internal or external fires. FHAs, performed for the respective surface and subsurface MGR operations areas (CRWMS M&O 1999a, 1999b, and 1999c) have shown that most credible fires are low to moderate size. No credible scenarios have been identified that can result in any credible release of airborne radioactivity or in abnormal exposure of workers.

The following subsections provide more details on the radiological and fire hazards analyses.

### 8.3.1 Internal and External Hazards Analyses

Potential radiological accidents were identified in *Monitored Geologic Repository Internal Hazards Analysis* (CRWMS M&O 2000a) and *MGR External Events Hazards Analysis* (or EHA) (CRWMS M&O 2000b). Fire scenarios initiated both within the facility (internal events) and outside of the facility (external events) were addressed qualitatively to identify potential

fire-initiated events involving radioactive waste forms. The IHA and EHA considered concepts of operations and preliminary layout drawings to identify where conventional fire hazards such as combustible material, electrical equipment, and vehicle fuels might interact with a radioactive waste form. In addition, the IHA identified two instances where heatup and ignition of zirconium alloy cladding might occur.

The IHA (CRWMS M&O 2000a) systematically addressed potential internal hazards and associated events that could lead to radioactive consequences to the public or facility workers. A generic checklist of potential events that included "fire" and "explosion/implosion" was applied to each operational area of the MGR. The IHA provided an initial screening to identify whether or not a fire, explosion, or implosion could can occur and potential ways for a fire to interact with a radioactive waste form. The IHA thereby identified areas where more detailed fire hazards analyses are needed to further screen out the fire hazards or to ensure that fire prevention/mitigation features are provided in the design and operation of the MGR. Table 8-1 summarizes the IHA conclusions on locations where waste forms are potentially susceptible to fire or explosion hazards. The EHA (CRWMS M&O 2000b) similarly applied generic checklists of potential external events that might threaten a waste form. Range fires were so identified.

Table 8-1. Internal Hazards Analysis Identification of Locations Where Potential Fires and Explosions Might Lead to Radiological Consequences

| Functional Area | Potential Fire Hazards That Might Lead to Radiological Consequences | Potential Explosion Hazard |
|---|---|---|
| Waste Receipt and Carrier/Cask Transport | Diesel fuel | None |
| Carrier Preparation | Diesel fuel | None |
| Carrier Bay | Diesel fuel | None |
| Canister Transfer | None | None |
| Assembly Transfer | Zirconium alloy cladding fire after loss of pool water; Zirconium alloy cladding fire due to SNF overheating in assembly dryer | None |
| Disposal Container Handling/Remediation | Disposal Container and cladding burnthrough during welding – radiation release; zirconium alloy fire | None |
| Subsurface Transport/Emplacement/ Monitoring | Electrical fire associated with locomotives and power supplies; transporter car; emplacement gantry and power supplies | None |
| Site-Generated Waste Treatment/Liquid LLW | None identified | None |
| Site-Generated Waste Treatment/Solid LLW | Fire in combustible LLW | None |

The potential radiological fire hazards identified to date will be subject to re-examination as the MGR design details become available, and as the Integrated Safety Analysis progresses. Meanwhile, preliminary FHAs for the surface and subsurface analyses have defined the essential fire protection features that will prevent or mitigate significant fire hazards. The preliminary FHAs, taken in combination with the IHA and consideration of the fire resistant designs of transport casks and waste packages, demonstrate that design features that prevent and/or mitigate the effects of fires will effectively eliminate any significant releases of radioactivity due to

fire-initiated scenarios. Furthermore, design and layout of fire suppression systems will be reviewed to ensure that no potential criticality events can occur in the WHB.

The following subsections summarize the fires hazards analyses.

### 8.3.2 Fire Hazards Analysis for Surface Facilities

Preliminary FHAs have been performed for the MGR surface facilities: *Waste Handling Building Fire Hazards Technical Report* (CRWMS M&O 1999a) and *Waste Treatment and Carrier Prep Buildings Fire Hazards Technical Report* (CRWMS M&O 1999b). The objective of both reports was to establish the requirements for fire protection systems to provide a proper level of personnel safety and property protection. The FHAs provide an evaluation of general arrangement drawings (floor plans and elevations) to determine the fire-protection system classification that considers protection of QL-1 SSCs. The FHAs provide preliminary information on how the WHB, CPB and WTB may be protected by mitigating the effects of fires. The results form the basis for design criteria in the System Description Documents for the respective systems.

To conduct the FHA, the rooms of the surface facility were divided into a series of fire areas of dissimilar functions. A fire area is defined as an area bounded by a two-hour rated construction. The respective fire areas are defined by applying several criteria: area usage (occupancy classification); radiation hazard; level of fire hazard; accessibility; type of building construction; number of occupants; and maximum possible fire loss limitations.

The level of fire hazard (fire load) and/or explosion hazards in each fire area is used to evaluate the type of building construction, egress requirements, and fire suppression systems based on applicable codes and standards. Fire load ratings of light, moderate, and severe express the pounds/square foot of combustible material or energy potential expressed in British thermal units per square foot. In a typical building, the fire load includes combustible contents, interior finish, floor finish, and structural elements. The FHA accounted for the HVAC confinement requirements for the respective fire areas which are classified as primary, secondary, tertiary, or none, based on the level of potential airborne radioactivity expected during normal operations and anticipated occurrences.

The respective fire loads were further classified as Light, Ordinary Hazard, and Extra Hazard per the National Fire Protection Association automatic sprinkler standard. Fire protection systems other than wet-pipe sprinklers may be used as the design evolves where it may be desirable to use a non-water suppression agent (e.g., should an area be shown to have a potential criticality scenario as well as a fire hazard).

Table 8-2 summarizes the results of the preliminary FHAs for the conceptual surface facilities. No explosion hazards in the WHB were identified. Laboratory gases are assumed to be non-flammable. A detailed review of explosion hazards will be performed when processes and materials have been developed.

It is concluded that neither the WHB, CPB, or WTB appear to present any fire hazards that cannot be mitigated satisfactorily. With one exception, areas are rated low or moderate,

including the WHB primary areas (see Table 8-2) where the waste handling operations are performed.

### 8.3.3 Fire Hazards Analysis for Subsurface Facilities

The *Subsurface Fire Hazards Technical Report* (CRWMS M&O 1999c) identifies design and operations features, fire and explosion hazards for both development and emplacement phases of subsurface operations.

The Subsurface FHA discusses two distinct fire areas: the Development Fire Area, and the Emplacement Fire Area. Fire barriers between the development and emplacement operations and their respective ventilation systems will provide isolation of one area from the other in the event of a fire. For both the development and emplacement sides, continuing evaluation is given to the life-safety issues of ventilation and egress routes. For the Emplacement Fire Area, special attention will be given to the potential effects of available fire suppression agents on nuclear safety and the needs for manual intervention (fire brigade).

The subsurface FHA has identified several fire hazard prevention and mitigation features to meet the NRC regulations and DOE orders, which provide the bases for design criteria in the System Description Documents. Some of the design requirements are listed below:

A. Development Fire Area Prevention/Mitigation Requirements

1. Ventilation barriers between development and emplacement to comply with a minimum fire resistance rating of two hours (to be re-examined as the design evolves for need of an increased fire resistance rating)
2. Automatic fire suppression systems onboard tunnel boring machines, maintenance rail car, conveyor system
3. Fire-retardant hydraulic fluids, fire-retardant cables, fire-resistant conveyor belting; electrical distribution equipment in cabinets
4. Batteries having minimum hydrogen offgassing
5. Fire protection system to withstand earthquakes.

Table 8-2. Summary of Fire Hazards Analysis for Surface Facilities

| Functional Area | Operations System | Area Description | Fire Hazard |
|---|---|---|---|
| WHB Primary | Carrier/Cask Handling<br>Assembly Transfer<br>Canister Transfer<br>Disposal Container Handling<br>Waste Package Remediation | HVAC confinement: primary or secondary. Occupancy: WHB primary areas –never; secondary areas – never/ intermittently.<br>Fire hazard: electrical wiring, cable and motors.<br>All penetrations require fire seals. | Low to Moderate |
| WHB Primary Support | Operating Galleries<br>Equipment Transfer Corridors<br>Contaminated Equipment Rooms<br>LLW Collection and Packaging<br>Maintenance Equipment Rooms<br>Weld Material Storage<br>Maintenance Shop<br>Forklift Staging and Servicing<br>Waste Handling Operations Center | HVAC confinement: tertiary or none.<br>Occupancy: normally occupied; some may contain radioactive contaminated materials.<br>Fire hazard: ordinary combustible material used for operation/maintenance.<br>All penetrations from areas of greater hazard require fire rated penetrations. | Moderate |
| WHB Pool Support | Pool Treatment Equipment Room | HVAC confinement: tertiary.<br>Occupancy: normally occupied. May contain radioactive contamination.<br>Fire hazard: ordinary combustible material used for operation/maintenance.<br>All penetrations from areas of greater hazard will require fire rated penetrations. | Low to Moderate |
| WHB Facility Support | Radiation Protection, Security, Operations, Administration, Building Circulation | HVAC confinement: tertiary or none.<br>Occupancy: Normally occupied.<br>Fire hazard: ordinary combustible material used for construction, office furnishings/supplies, or operation/maintenance.<br>All penetrations from areas of greater hazard will require fire rated penetrations. | Moderate |

Table 8-2. Summary of Fire Hazards Analysis for Surface Facilities (Continued)

| Functional Area | Operations System | Area Description | Fire Hazard |
|---|---|---|---|
| WHB HVAC Equipment | HVAC Equipment Room | HVAC confinement: tertiary.<br>Occupancy: Normally unoccupied; may contain contaminated equipment.<br>Fire hazard: ordinary combustible material used for operation/maintenance; HEPA filters located here.<br>All penetrations into area require fire rated penetrations. | Low to Moderate |
| WHB Misc. Building Support | Fire Protection, Electrical Equipment, Communications Equipment | HVAC confinement: none.<br>Occupancy: Normally occupied.<br>Fire hazard: electrical wiring, switchgear, cable, and electronic devices.<br>Fire protection alarm system monitors & displays give status of fire alarms and suppression devices in the facility.<br>All penetrations from areas of greater hazard will require fire rated penetrations. | Moderate to High |
| WHB Structure | WHB Structure | Noncombustible structure as defined by Uniform Building Code.<br>Withstands Design Basis external events: earthquake, extreme wind/tornado winds, and potential tornado missiles.<br>Provide adequate fire protection of WHB operations from fires external to the WHB. | Low |
| WTB Processes | Solid Waste Processing<br>Non-recyclable LLW Processing<br>Recyclable Liquid LLW Processing | HVAC confinement: none.<br>Occupancy: Normally occupied.<br>Fire hazard: combustible solid wastes (contaminated); ordinary combustibles used in operations/maintenance. | Low to Moderate |
| WTB Operations | Offices<br>Health Physics<br>Staging<br>Electrical Rooms<br>HVAC Equipment Room | HVAC confinement: none.<br>Occupancy: Normally occupied.<br>Fire hazard: combustibles used in operations/maintenance; electrical motors, cabling, cabinets. | Low to Moderate |

Table 8-2. Summary of Fire Hazards Analysis for Surface Facilities (Continued)

| Functional Area | Operations System | Area Description | Fire Hazard |
|---|---|---|---|
| WTB Structure | Concrete/steel Structure | | Non-combustible |
| CPB Structure | Concrete/steel Structure | | Non-combustible |
| CPB Operational Area | Carrier Preparation Area | HVAC confinement: none.<br>Occupancy: Normally occupied.<br>Fire hazard: ordinary combustibles used in operations/maintenance; may be diesel fuel in prime mover. | Moderate |

B. Emplacement Fire Area Prevention/Mitigation Requirements

1. Redundant, automatic fire extinguishing systems onboard transport locomotives, waste package transporter, emplacement gantry and inspection gantry (primarily for fires initiated in electrical or electronic systems)

2. Redundant, automatic fire extinguishing systems onboard diesel-powered waste retrieval equipment

3. Fire-resistant cable insulation; electrical distribution equipment in cabinets and in alcoves

4. Fire protection system to withstand earthquakes.

In both the Development Fire Area and the Emplacement Fire Area, there will be provisions for personnel safety including self-rescuers, refuge chambers, and smoke removal.

The hazards identified and their proposed mitigation are considered preliminary and scoping but provides a reasonable basis for supporting existing design requirements and to develop new requirements for fire protection systems.

### 8.3.4 Design/Operational Features for Prevention/Mitigation of Fire-Initiated Radiological Events

Features of the MGR operations and facility design that prevent or mitigate the effects of the potential fire-initiated radiological hazard are described in the following paragraphs.

Transport casks entering the MGR are designed to withstand the severe transportation fire environment specified by 10 CFR 71.73, *Hypothetical Accident Conditions*. Therefore, a radiologically significant design basis fire for the carrier preparation area and the carrier bay would have to exceed the size and duration of such fires. The FHA for the CPB shows that the fire level is moderate and, therefore, there is no credible means by which a fire in the CPB or carrier bay of the WHB could cause a breach of transport cask and a release of radioactivity.

Similarly, waste packages will be designed to withstand the same fire environment as transportation casks. The FHAs show that only low to moderate fire hazards exist in the primary functional areas of the WHB and the subsurface facilities, so it is unlikely that any credible fire in the WHB will approach the severity of a design basis fire for a transport cask. Therefore, after completion of the final seal weld, a fire-induced breach of a waste package is not credible at any point in the waste stream beyond the welding station inside the WHB.

Elsewhere in the WHB, bare SNF assemblies and sealed HLW canisters are handled. These operations are performed within the robust, non-combustible confinement structure provided by the WHB. The FHA shows that the fire hazard level is low to moderate for these operations areas. A design basis fire for these areas has to have temperature and duration sufficient to cause a breach of SNF cladding or HLW canister. It is unlikely that fires of sufficient severity can occur. Even if a release of radioactivity occurs, the radioactivity would be confined by the robust structure of the WHB and the confinement provided by the HVAC system. Further, the

separation provided by the primary, secondary, and tertiary confinement zones of the HVAC will prevent or retard propagation of fires between WHB operational areas. Therefore, even if a low to moderate fire should occur, it is judged to be an extremely unlikely scenario for a significant release of radioactivity to the public. As design details evolve, a quantification of the potential fires in each operational area will be performed to ensure that there is no credible mechanism for a release of radioactivity that would exceed the interim guidance dose limits (i.e., Section 111 of Dyer 1999). The structure of the WHB is classified as QL-1 so portions of the fire protection system will be classified as QL-2 as required to ensure the integrity of the QL-1 structure.

The IHA identified ignition of zirconium alloy cladding as a potential fire source in the SNF pool, the assembly dryer in the ATS, and the welding station in the DCHS. The zirconium alloy fire in the pool area was postulated to occur after an event (other than a fire) that causes loss of pool water. However, for the MGR, the pool water is provided primarily for biological shielding and not for SNF cooling, so the zirconium alloy kindling temperature is unlikely to be reached. Should a radioactive release be initiated, confinement will be provided by the HVAC. When the design of the pool system and the thermal loading of the SNF are better known, the credibility and severity of a zirconium alloy fire will be determined. An event sequence involving a loss of pool water, zirconium alloy fire, and a radioactive release is expected to be incredible. If necessary, fire prevention and accident mitigation features will be defined.

The zirconium alloy fire in the ATS assembly dryer was postulated to occur after an uncontrolled heatup of SNF in the assembly dryer. However, limit switches, operator controls and administrative controls are expected to ensure that the zirconium alloy kindling temperature is unlikely to be reached. Should a radioactive release be initiated, confinement will be provided by the HVAC. When the design of the assembly dryer system and the thermal loading of the SNF are better known, the credibility and severity of a zirconium alloy fire will be determined. An event sequence involving heatup of SNF in the ATS dryer, zirconium alloy fire, and a radioactive release is expected to be incredible. If necessary, fire prevention and accident mitigation features will be defined.

The zirconium alloy fire in the welding station is postulated to occur as a result of a burnthrough of the waste package such that the SNF inside becomes overheated and ignites. Due to the current level of design detail, the likelihood of this event is not known. The confinement provided by the WHB structure and HVAC are not likely to be threatened by such a fire. When the design of the welding system controls and thermal characteristics are better known, the credibility and severity of a zirconium alloy fire will be determined. An event sequence involving a welding burnthrough, zirconium alloy fire, and a radioactive release is expected to be incredible. If necessary, fire prevention and mitigation features will be defined.

The WHB will be constructed of non-combustible materials and will be designed to withstand the effects of the most severe fire or explosion that can credibly occur outside the building. This will be determined later as part of the design basis fire for the WHB. The radiological EHA has identified range fires as one issue to be examined as the design matures. Other sources of fires outside the WHB, and perhaps the most probable, are diesel fuel and gasoline that is expected to be used in vehicles bringing waste transport casks onto the site, in vehicles used onsite for normal operations and personnel transport, and any onsite storage tanks of such fuel.

The IHA identified a potential radioactive release by a fire in the solid waste processing area of the WTB. The FHA for the WTB shows that the fire level is low to moderate. The potential radiological source term from burning of solid LLW is small. Therefore, the consequences to the public or workers from such fires are expected to be well within the dose limits established for Category 1 DBEs. Therefore, standard industrial-grade provisions for prevention and mitigation of the fires should suffice. As the design and characteristics of the solid LLW are better defined, analyses will be performed to verify this conclusion.

The FHA for the subsurface operations area (CRWMS M&O 1999c) addressed the emplacement operations and the concurrent construction operations as being in separate fire areas (i.e., separated by a fire barrier having at least a two-hour rating). No combustible fuels are to be used in normal subsurface transport, emplacement, performance verification, or maintenance. The primary source of credible fires is electrical equipment and cabling. Diesel-powered vehicles for certain recovery, rescue, or waste-package retrieval operations have not been ruled out. Since the waste packages will be designed to withstand the intensity and duration of the design basis fire for transport casks, it is unlikely that any credible fire in the subsurface operations can pose a threat to the integrity of the waste package.

## 8.4 CONCLUSIONS

The fire safety strategy for the MGR relies on both permanent design features and a fire safety and protection program. SSCs providing preclosure fire protection functions will be identified and site characteristics and external hazards associated with the Yucca Mountain site have been described. The fire hazards analyses will support the Integrated Safety Analysis that addresses radiological safety but will also support the fire protection program for non-radiological safety. Although DOE's interim guidance for the proposed 10 CFR Part 63 (64 FR 8640) is performance-based and non-prescriptive, the fire protection program for the MGR will follow, to the extent deemed applicable, industry codes and standards, NRC guidance, and DOE guidance for fire protection and safety at other fuel-cycle facilities.

## 8.5 REFERENCES

### 8.5.1 Documents Cited

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### 8.5.2 Codes, Standards, Regulations, and Procedures

64 FR 8640. Disposal of High-Level Radioactive Wastes in a Proposed Geologic Repository at Yucca Mountain, Nevada. Proposed rule 10 CFR 63. Readily available.

10 CFR (Code of Federal Regulations) 71. Energy: Packaging and Transportation of Radioactive Material. Readily available.

10 CFR 72. Energy: Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste. Readily available.

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# 9. RADIOACTIVE AND HAZARDOUS WASTE MANAGEMENT

Secondary radioactive waste streams will be generated during the processes associated with receiving and packaging of commercial SNF, DOE SNF, and DOE HLW for disposal at the MGR. These secondary waste streams will be primarily LLW, as defined in Table 9-1.

The Site Generated Radiological Waste Handling System will handle radioactive waste products that are generated at the geologic repository operations area. The waste will be collected, treated where required, packaged for shipment, and shipped to a disposal site. Waste streams include LLW in solid and liquid forms, hazardous waste, as well as mixed waste that contains hazardous and radioactive constituents. The Site-Generated Radiological Waste Handling System will have equipment located in the WHB as well as the WTB. The SSCs that support the collection, segregation, and disposal of LLW and site-generated hazardous, non-hazardous, and sanitary waste disposal will be located in the MGR WTB. The activities to be conducted within the WTB include dry waste sorting, drum waste compaction, and liquid waste cleaning (CRWMS M&O 1999). However, there will be no installed hardware for processing mixed waste. The types of wastes expected to be generated at the MGR are summarized in Table 9-1 (DOE 1999, p. 4-75).

Table 9-1. Waste Types

| Waste Type | Waste Description |
|---|---|
| Industrial Wastewater | Liquid wastes from industrial processes that do not include sanitary sewage. Repository industrial wastewater would include water used for dust suppression and process water from building HVAC systems. |
| Sanitary Sewage | Domestic wastewater from toilets, sinks, showers, kitchens, and floor drains from restrooms, change rooms, and food preparation and storage areas. |
| Sanitary and Industrial Solid Waste | Solid waste that is neither hazardous nor radioactive. Sanitary waste streams include paper, glass, and discarded office material. State of Nevada waste regulations identify this waste as *household waste.* |
| Low-Level Radioactive Waste | Radioactive waste that is not classified as HLW, transuranic waste, byproduct material containing uranium or thorium from processed ore, or naturally occurring radioactive material. The repository LLW would include such wastes as personal protective clothing, air filters, solids from the liquid LLW treatment processes, radiological control and survey waste, and possibly used dual-purpose canisters. |

Table 9-1. Waste Types (Continued)

| Waste Type | Waste Description |
|---|---|
| Hazardous Waste | Waste designated as hazardous by the Environmental Protection Agency or State of Nevada regulations. Hazardous waste, defined under the Resource Conservation and Recovery Act, is waste that poses a potential hazard to human health or the environment when improperly treated, stored, or disposed of. Hazardous wastes appear on special Environmental Protection Agency lists or possesses at least one of the following characteristics: ignitability, corrosivity, toxicity, or reactivity. The use of hazardous materials at the MGR will be controlled and minimized. Hazardous waste streams from the repository could include certain used rags and wipes contaminated with solvents. |
| Low-Level Mixed Waste | Hazardous waste that also exhibits characteristics of LLW. Low level mixed waste could include solvents or other chemicals used in decontamination activities. Low level mixed waste is not anticipated to be produced as a result of normal repository operations due primarily to proper selection and segregation of materials used at the repository, as well as minimization of the use of hazardous materials. |

## 9.1 ONSITE WASTE SOURCES

LLW will be generated primarily in the WHB, and lesser quantities of this waste will be generated from activities within the WTB. Due to the possibility of receiving a surface-contaminated SNF/HLW shipment, the possibility exists for the generation of minor quantities of LLW in the CPB. LLW streams will be primarily liquids and solids. Minor releases of particulates will be processed through the WHB and WTB HEPA filtration systems prior to discharge into the atmosphere.

Radioactive isotopes anticipated in the LLW and mixed waste at the MGR will consist of mixed fission products and activation products associated with the handling of commercial SNF. The term "waste" as used in this section refers to those wastes generated during MGR operations, and does not include SNF or other HLW stored or otherwise handled at the MGR.

### 9.1.1 Gaseous Wastes

Gaseous wastes will not be generated at the MGR; however, airborne radioactive contamination can be generated in the WHB and potentially, the WTB. Potential sources of airborne contamination include:

- Aerosols of surface contamination from the exterior of transportation casks or from the exterior of DPCs or disposal containers
- Cask, canister, or waste package leakage as a result of a failed seal

- Aerosolized particulates from SFAs (fuel cruds) located in the storage racks, dryers, transfer machines, as well as other parts of the ATS

- Gases escaping from leaking spent fuel rods located in various parts of the ATS (including radionuclides soluble in the pool water)

- Particulates or aerosols released in the waste treatment area in the WTB.

Airborne radioactive contamination levels in the WHB and WTB ventilation exhausts are expected to be less than the limits listed in Table 2 of Appendix B to 10 CFR Part 20. Continuous HEPA filtration of facility exhausts will be provided, as described in Section 9.5.

### 9.1.2 Liquid Wastes

#### 9.1.2.1 Low-Level Radioactive Waste Water

Liquid LLW is comprised of fluids that are contaminated with radioactive materials. It will be generated by decontamination and maintenance activities as well as by other operations performed in a Radiologically Controlled Area (RCA). Reduction of waste volume is a primary objective in the treatment of this waste and treated LLW water will be recycled to the extent practical. The surface facilities will segregate aqueous wastes at their source of generation into recyclable and non-recyclable waste streams. The non-recyclable stream may contain detergents or other non-hazardous cleaning agents; it will be collected, solidified, and packaged for shipment offsite. The recyclable stream will be treated to recycle a large portion of the water while the remaining concentrated waste will be packaged for shipment; this will greatly reduce the volume of waste requiring disposal. The various activities that will generate these two waste streams include (CRWMS M&O 1995):

Recyclable liquid waste (aqueous streams suitable for treatment and recycling):

- Floor washdown
- Loaded transportation cask exterior decontamination
- Unloaded transportation cask exterior decontamination
- Waste package washing
- Small equipment/tool decontamination.

Non-recyclable waste (aqueous streams unsuitable for treatment and recycling):

- Floor washdown
- Loaded transportation cask decontamination
- Unloaded transportation cask exterior decontamination
- Small equipment/tool decontamination.

All liquid LLW will be solidified and shipped, as described in Section 9.2. There will be no liquid LLW discharge.

#### 9.1.2.2 Liquid Hazardous Waste

The repository surface facilities will generate a number of streams classified as hazardous waste, such as oils, antifreeze coolant, medical waste, and solvents. Hazardous waste can be generated both in the RCA and the BOP area of the repository. Hazardous waste from the RCA will be handled separately from hazardous waste generated by the BOP area of the repository. The source of waste oils is generally site vehicles and other transport equipment. Medical waste will only be generated in the medical center in the BOP area of the repository (CRWMS M&O 1995).

#### 9.1.2.3 Liquid Low-Level Mixed Waste

Liquid hazardous waste may become liquid low-level mixed waste if it becomes contaminated with radionuclides. It is not anticipated that mixed waste will be produced during waste handling operations, but there will be an allowance made for temporarily staging a small quantity of this waste prior to shipping it offsite. The production of mixed waste will be minimized by administratively controlling the use of hazardous materials in MGR to prevent the inadvertent mixing of hazardous materials with radiological wastes. Even with these administrative controls, small amounts of mixed wastes may be generated. For example, a transport vehicle temporarily in the WHB could leak oil, which could become radiologically contaminated. Low-level mixed waste generation is not anticipated in the BOP area since this is an unrestricted area without any radioactive waste handling or processing operations (CRWMS M&O 1995).

#### 9.1.2.4 Sanitary Wastes

Sanitary wastes to be generated at the MGR include the effluents from facility drinking water fountains, water closets, lavatories, mop sinks, and other similar fixtures. The site water system will supply potable and non-potable water to the surface water distribution systems. Site water originates at the Nevada Test Site wells approximately 3.5 miles southwest of the North Portal. Potable water is provided for drinking, cooling, decontamination, and sanitary uses; non-potable water is provided for construction and fire protection. The water system meets State of Nevada requirements (CRWMS M&O 1999, p. 57).

### 9.1.3 Solid Waste

#### 9.1.3.1 Solid Low-Level Waste

Solid LLW will be generated as a result of SNF and HLW handling operations, decontamination operations, housekeeping activities, and maintenance activities conducted within the RCA. These wastes must be processed for several purposes (CRWMS M&O 1995):

- To segregate wet solids, compactible solids, non-compactible solids and oversized equipment/tools that require mechanical disassembly
- To reduce the volume of compactible solids to the maximum extent possible
- To reduce the mobility of the wastes during prolonged storage.

Solid radioactive waste will be safely accumulated at the point of origin, then sent to the WTB, where it will be treated appropriately and packaged in drums.

Solid LLW will consist of wet solids such as ion exchange resins and filter cartridges, as well as dry active waste such as tools, protective clothing, and plastic bags. Solids will be sorted, volume reduced, and packaged for shipment. Cask decontamination activities produce waste paper and cloth that are classified as compactible, solid LLW.

Metallic, non-compactible, solid LLW will include spent HEPA filter elements discharged from radioactive service, as well as valves, fittings, pipes, bolts and other various metallic scrap classified as non-compactible, solid LLW generated by maintenance operations. These operations will be performed periodically on each of the transportation casks serving the repository surface facilities (CRWMS M&O 1995).

Used (opened and unloaded) dual-purpose canisters will be considered LLW; they will be placed in an overpack suitable for shipping. The used canisters will be packaged for offsite shipment at the WHB and will not be processed in the WTB (CRWMS M&O 2000, Attachment II, Section 1.4.4.3.3).

#### 9.1.3.2 Solid Hazardous Waste

The repository surface facilities may have the potential to produce solid hazardous waste, both in the RCA and BOP areas. Rags, paper, or plastic containing chemicals or solvents are examples of this type of waste. This hazardous waste will not be allowed into the WTB; it will be collected and packaged in drums for proper disposition, in a manner similar to that for liquid hazardous waste (CRWMS M&O 1995).

#### 9.1.3.3 Solid Low-Level Mixed Waste

The repository surface facilities may also have the potential to produce solid low-level mixed waste. Rags or paper containing solvents or chemicals used in decontamination or cleanup activities are examples of this type of waste. It is not anticipated that mixed waste will be produced during waste handling operations, but there will be an allowance made for temporarily staging a small quantity of this waste prior to shipping it offsite (CRWMS M&O 1995).

### 9.1.4 Waste Quantities

Table 9-2 (DOE 1999) lists the estimated total waste quantities for repository activities associated with emplacement and development (years 2010 to 2033). Major waste-generating activities would include the receipt and packaging of SNF and HLW and continued development of the subsurface emplacement areas. The three HLW packaging scenarios under consideration would affect the volumes of LLW and hazardous waste generated at the surface facilities as a result of the differences in handling the SNF and HLW. In addition, waste would be generated in personnel areas such as change rooms, restrooms, and offices. The dual-purpose canister packaging scenario could require the disposal of an additional estimated 44,000 cubic meters (1.6 million cubic feet) of low level radioactive waste (not listed in Table 9-2). DOE could decide to recycle the canisters if doing so would be more protective of the environment and more cost-effective than direct disposal. Recycling would require melting and recasting of the canister metal to enable other uses (DOE 1999, Table 4-40).

Table 9-2. Estimated Waste Quantities from Emplacement and Development Activities (years 2010 to 2033)

| | High thermal load | | | Intermediate thermal load | | | Low thermal load | | |
|---|---|---|---|---|---|---|---|---|---|
| Waste Type | UC[a] | DISP[b] | DPC[c] | UC[a] | DISP[b] | DPC[c] | UC[a] | DISP[b] | DPC[c] |
| Hazardous (cubic meters)[d] | 5,800 | 2,300 | 2,200 | 5,800 | 2,300 | 2,200 | 5,800 | 2,300 | 2,200 |
| Sanitary and industrial solid (cubic meters) | 50,000 | 41,000 | 42,000 | 50,000 | 41,000 | 42,000 | 70,000 | 61,000 | 62,000 |
| Sanitary sewage (million liters)[e] | 1,400 | 1,100 | 1,200 | 1,400 | 1,100 | 1,100 | 1,400 | 1,200 | 1,200 |
| Industrial wastewater (million liters) | 900 | 780 | 780 | 930. | 810 | 810 | 1,400 | 1,300 | 1,300 |
| Low-level radioactive (cubic meters, after treatment) | 67,000 | 18,000 | 26,000 | 67,000 | 18,000 | 26,000 | 67,000 | 18,000 | 26,000 |

Notes: a. UC = uncanistered.

b. DISP = disposable canister.

c. DPC = dual-purpose canister.

d. To convert cubic meters to cubic feet, multiply by 35.314.

e. To convert liters to gallons, multiply by 0.26481.

Monitoring and maintenance activities after the completion of emplacement (years 2034 to 2110) would also generate wastes, but in much smaller quantities. The first few years after the completion of emplacement would generate greater quantities of wastes due to the decontamination and decommissioning of surface nuclear facilities. DOE estimates that as much as 520 cubic meters (18,000 cubic feet) of LLW and as much as 260 cubic meters (9,200 cubic feet) of hazardous waste would be generated from this activity.

Monitoring and maintenance activities over 26 years would generate a maximum of about 9,900 cubic meters (350,000 cubic feet) of sanitary and industrial solid waste and about 230 million liters (60 million gallons) of sanitary sewage. Ongoing monitoring and maintenance activities for 76 years would generate a maximum of about 20,000 cubic meters (710,000 cubic feet) of sanitary and industrial solid waste and about 450 million liters (120 million gallons) of sanitary sewage (DOE 1999).

## 9.2 LIQUID WASTE TREATMENT AND RETENTION

### 9.2.1 Design Objectives

The WTB will house the site-generated radioactive waste handling system, which will collect, prepare, and/or store the site-generated low-level radioactive solid and liquid, and mixed waste for disposal. The site-generated radioactive waste handling system will control the collection of waste and treat it prior to packaging for disposal offsite. It is expected that the radioactivity of the waste will be low enough that no special facility features will be required to meet NRC

radiological safety requirements for shielding and criticality. The WTB will be adjacent to the WHB carrier bay. The facility will house the handling equipment, process tanks, piping, instrumentation, offices, and personnel involved in the collection and processing of liquid and solid waste from the WHB preparation and handling processes. The system will also contain equipment, tanks, and piping for dewatering of spent resin that has been used for purification of the pools in the WHB.

The majority of the WHB liquid waste will be pumped through piping to the WTB process systems. Other waste will arrive in sealed containers via the site transportation system. Recyclable liquid waste will be treated and made available for users. Non-recyclable liquid waste will be packaged for disposal.

The current waste treatment system conceptual design configuration for disposal of liquid LLW includes the following features (CRWMS M&O 1997b):

- Classification and segregation of aqueous LLW streams as recyclable or non-recyclable aqueous waste
- Treatment of aqueous recycle streams via filtration, evaporation, and ion exchange.

### 9.2.2 System and Equipment Description

#### 9.2.2.1 Low-Level Radioactive Waste Water

Equipment associated with the recyclable liquid LLW treatment system includes a collection tank for the recyclable waste, filters, feed and storage tanks, an evaporator, an ion exchange unit, and various pumps and piping. Liquid LLW will be received through piping from the WHB and from the WTB waste process vessels and sumps. The system will perform the waste processing functions required to receive the waste, separate the recyclable from the non-recyclable liquid waste, process and package the waste, treat the recyclable wastewater, and store useable water. The system will segregate and store the useable and non-reusable liquid waste in separate tanks. The system will process the recyclable liquid waste through evaporation, condensation, and ion exchange components. Reusable condensate will be stored in a recycle water tank, from which the liquid will be pumped to facility users (CRWMS M&O 1999).

Equipment associated with the non-recyclable liquid LLW treatment system includes various holding tanks, a pH (hydrogen ion concentration potential) adjustment tank, as well as various pumps and piping. The non-recyclable liquid LLW from the evaporator and aqueous non-recyclable low level waste from the WHB will be treated in this system. The pH (hydrogen ion concentration potential) will be adjusted, and the waste will be immobilized packaged in drums at the drum fill station. This waste stream will be treated on a batch basis, and the waste drums will be transported offsite for disposal (CRWMS M&O 1999).

#### 9.2.2.2 Liquid Hazardous Waste

Segregation of the hazardous waste will be practiced to minimize the potential for generation of low-level mixed waste from cross-contamination of BOP hazardous waste with radionuclides. Following collection and handling in 55-gallon drums, hazardous waste will be shipped to a staging shed; one each in the RCA and BOP areas of the repository, prior to shipment for final

treatment and disposal at a commercial Resource Conservation and Reclamation Act-licensed facility (CRWMS M&O 1995). Hazardous waste will not be allowed into the WTB.

#### 9.2.2.3 Liquid Low-Level Mixed Waste

Liquid low-level mixed waste will be collected at the point of generation in shielded areas isolated from areas handling hazardous waste and LLW. The liquid low-level mixed waste from the RCA area will be segregated into oil-based, water-based, and hydrocarbon-based streams. Following collection, the liquids will be packaged, as received, in drums (i.e., 55-gallon capacity) suitable for handling and storage of this waste. The filled waste drums will then be sealed and loaded onto site vehicles for transfer to the low-level mixed waste transfer point inside the WTB. From this location, the drums will be shipped to an appropriate facility for treatment and disposal (CRWMS M&O 1995).

## 9.3 CONVENTIONAL WATER/SANITARY WATER SYSTEM

### 9.3.1 Design Objectives

The site water system supplies potable and non-potable water to the surface water distribution systems. Water appropriations are presently permitted for 430 acre-feet per year. It is estimated that this level of appropriation will adequately meet the water requirements for all future phases of the repository operations through closure. The existing sanitary waste system is estimated to be adequate in capacity to handle future phases of repository operations. The system consists of a septic tank and a leach field (CRWMS M&O 1999, p. 57).

### 9.3.2 System and Equipment Description

Site water originates at the Nevada Test Site wells, from which it is pumped to a booster station and then to potable and non-potable water tanks on Exile Hill. From here, it is distributed throughout the RCA, the BOP area, and to the subsurface. Equipment associated with this system includes chillers, pumps, hot water boilers, and expansion tanks (part of the HVAC systems). The water system will possess adequate pumping, flow, pressure, and reserve capacity for the water distribution networks that the site water system serves.

Sanitary sewage is collected in the septic tank, and liquid effluent is routed to the leachfield, where it is removed by percolation in the soil. The design standards for the sanitary sewer collection and treatment systems conform to the State of Nevada requirements and regulations (CRWMS M&O 1999). Administrative and design controls will prevent the disposal of radioactive material in the sanitary waste system.

During the operation and monitoring phase, the sanitary sewage disposal system should be able to handle the estimated daily sewage flows and the industrial wastewater facilities should be able to handle the estimated annual wastewater flows.

## 9.4 SOLID WASTES

### 9.4.1 Design Objectives

Solid LLW will be received in a variety of forms, including:

- Resin, slurry, etc. material from the liquid low-level radiological waste system
- Compactible material, including rags, clothing, metal shavings, filters, etc.
- Non-compactible material requiring shredding or disassembly, including major pieces of equipment such as opened DPCs and large mechanical parts.

The solid low-level radiological waste system will receive non-compactible or oversized waste, solid compactible LLW, and spent ion exchange resins from the liquid LLW treatment system.

### 9.4.2 System and Equipment Description

#### 9.4.2.1 Solid Low-Level Waste

The solid LLW system separates compactible and non-compactible wastes, reduces non-compactible waste to compactible form, and compacts the waste for disposal. Large solid waste will be routed to a mechanical disassembly station where operators will dismantle or cut up and separate the large pieces. The undersize and suitable separated material will then be loaded into separate drums. The pieces will again be separated at a waste type sorting station, where large pieces will be routed to a shredder, reducing the waste to compactible form. Shredded and undersized waste will be compacted into drums at an in-drum compactor component. The non-compactible stream from the initial sorting process will be placed in 55-gallon drums.

#### 9.4.2.2 Solid Hazardous Waste

Solid hazardous waste from the BOP facilities will be prepared and handled independently from the hazardous waste generated in the RCA area to prevent cross-contamination of the waste (and generation of mixed waste). Preparation and handling of solid hazardous waste will be performed at the source of generation. If necessary, a portable shearing unit may be used for the reduction of oversized waste material prior to packaging. In addition, an absorbent material may be added to the bottom of the solid hazardous waste container to absorb any free liquid that may be present. Fresh absorbent material and clean storage containers may share storage space with the empty drums designed to contain liquid hazardous waste. Partially filled waste containers will be sequestered in one of several accumulation areas located near the points of waste generation. Once the containers are filled with waste, each container will be sealed closed and transferred by site vehicle to the RCA or BOP staging shed, as appropriate. Storage of the hazardous waste will be limited such that a Resource Conservation and Reclamation Act permit is not required. Following temporary storage, the waste will be transported to a Resource Conservation and Recovery Act licensed treatment and disposal facility (CRWMS M&O 1995).

#### 9.4.2.3 Solid Low-Level Mixed Waste

Solid low-level mixed waste generated in the RCA will be handled in the same manner as liquid low-level mixed waste. Solid materials that are contaminated with hazardous constituents and radioactive nuclides will be collected at the source of generation. They will be allowed to accumulate in storage containers, such as 55-gallon drums. Once the drums are filled with low-level mixed waste, they will be transported by site vehicles to the WTB, where a low-level mixed waste transfer point will be provided to accumulate the drums. The low-level mixed

waste will be transferred to an offsite facility for final treatment and disposal (CRWMS M&O 1995).

## 9.5 OFF-GAS TREATMENT AND VENTILATION

### 9.5.1 Design Objectives

The WTB ventilation system will provide proper environmental conditions for the equipment used in this facility as well as for the health, safety, and comfort of operating personnel. The ventilation system will be designed to confine radioactive and hazardous materials within the waste treatment area as close to the point of origin as practicable and also prevent uncontrolled releases to rooms and areas normally occupied by personnel. The ventilation system exhaust and supply air flows will be adjusted to maintain the facility at a negative pressure with respect to the outside atmosphere, ensuring that air leakage will be into the WTB structure (CRWMS M&O 1995). Airborne contamination will be removed and airflow will be controlled away from penetration barriers to protect personnel from radiation exposure and minimize inadvertent release of radioactive particles to the site boundary. Fire protection, radiation monitoring, and leak detection systems will also be included in the design. The WTB ventilation confinement zones are:

- Primary Confinement Zone: Process enclosures (tanks, drums, and other miscellaneous process items), process off-gas vent systems, and the final exhaust HEPA filters.

- Secondary Confinement Zone: Enclosures/rooms that contain potentially contaminated pieces of process equipment or rooms supporting primary confinement functions. Examples of areas that are classified as secondary confinement zones include the LLW area, the recyclable liquid LLW area, the solid LLW area, the low-level mixed waste interim storage area, the temporary storage area, and the associated HEPA-filtered final exhaust air system.

- Tertiary Confinement Zones: Rooms through which the contaminated material is transferred into the processing areas, and the associated ventilation system (CRWMS M&O 1997a).

The other areas of the facility will normally be clean and provisions will be made to prevent these areas from obtaining a more negative pressure than any adjacent potentially contaminated area (CRWMS M&O 1995).

### 9.5.2 System and Equipment Description

This system will be comprised of three separate and independent subsystems. The first is a HEPA filtered system that mainly will serve the waste treatment area. The second subsystem will serve the shipping and receiving area. The third subsystem will serve the offices and other miscellaneous rooms (CRWMS M&O 1995).

#### 9.5.2.1 Waste Treatment Area HVAC Subsystem

This subsystem is a separate, independent ventilation subsystem designed to operate continuously, maintain design conditions, and provide for contamination confinement. The waste treatment process vents are discharged to the outside environment by a dedicated HEPA unit with two testable stages of HEPA filters and through the stack by the secondary confinement final exhaust fans. The negative pressure in the process vent system is maintained by the process vent blowers. Corrosive vapors, noxious gases or vapors, and flammable (or combustible) gases are not anticipated from these vents. This subsystem is a once-through concept consisting of supply air handling units for room ventilation and filtered exhaust with two testable stages of HEPA filtration, exhaust fans, and a stack. The waste treatment area is classified as a secondary confinement ventilation zone and will be maintained at a negative pressure with respect to the outside atmosphere. This subsystem is provided with backup units to meet redundancy requirements for maintenance only. This subsystem does not have emergency power.

#### 9.5.2.2 Receiving and Shipping Room HVAC Subsystem

This subsystem is designed to operate continuously, maintain design conditions, and ensure proper indoor air quality. This subsystem and associated/independent HVAC equipment room are not classified as confinement zones, and no redundancy or emergency power is required. The subsystem is a recirculation type designed to operate normally with approximately 10 percent outside air, operating once-through as required for removing diesel fumes discharged by truck exhaust. This subsystem consists of a supply air handling unit and a recirculation/exhaust fan. The truck door will be provided with air curtains to prevent excessive inlet of dust or loss of treated air.

#### 9.5.2.3 Offices and Other Miscellaneous Rooms HVAC Subsystem

This subsystem is similar to the Receiving and Shipping Room HVAC Subsystem except that the change rooms will be provided with a single stage of HEPA filtration and the diesel fumes exhaust operating mode will not be required.

## 9.6 RADIOLOGICAL IMPACT OF NORMAL OPERATIONS - SUMMARY

The site-generated radiological waste treatment system will perform the functions required to receive radioactive LLW generated at the waste handling facilities in the RCA, and safely process and package the waste in containers suitable for disposal. The WTB will house the process systems that will segregate liquid and solid LLW streams and package the waste for disposal offsite. Protection of workers and the environment from the maximum expected radiation levels and releases from the process vessels, piping, and material handling components will be provided by appropriate barriers and may include shielding, leak detection, and sump collection components. The process areas will be filtered and vented to the WTB ventilation system and the process, ventilation system, and operating areas will be continuously monitored by the radiological monitoring and alarm system. Radiation protection principals will be incorporated in the design to achieve ALARA exposure levels. Shielding will be provided in those locations where concentrated wastes are accumulated, such as adjacent to evaporators and ion exchange units. Provisions for remote repair will also be employed (CRWMS M&O 1999).

## 9.7 REFERENCES

### 9.7.1 Documents Cited

CRWMS M&O (Civilian Radioactive Waste Management System Management and Operating Contractor) 1995. *Waste Treatment Building Interim Design Study for FY 1995.* BCB000000-01717-5705-00007, REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19960619.0126.

CRWMS M&O 1997a. *Surface Nuclear Facilities HVAC Analysis.* BCBD00000-01717-0200-00013 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19980112.0189.

CRWMS M&O 1997b. *Secondary Waste Treatment Analysis.* BCBD00000-01717-0200-00005 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19971208.0201.

CRWMS M&O 1999. *Repository Surface Design Engineering Files Report.* BCB000000-01717-5705-00009 REV 03. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19990615.0238.

CRWMS M&O 2000. *Engineering Files for Site Recommendation.* TDR-WHS-MD-000001 REV 00. Las Vegas, Nevada: CRWMS M&O. MOL.20000607.0232.

DOE (U.S. Department of Energy) 1999. *Draft Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada.* DOE/EIS-0250D. Summary, Volumes I and II. Washington, D.C.: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ACC: MOL.19990816.0240.

### 9.7.2 Codes, Standards, Regulations, and Procedures

10 CFR (Code of Federal Regulations) 20. Energy: Standards for Protection Against Radiation. Readily Available.