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Yucca Mountain Site Characterization Project

Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation

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DISCLAIMER

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CHANGE HISTORY

Revision Number	Interim Change No.	Effective Date	Description of Change
0	N/A	06/29/00	Initial issue
0	01	8/21/00	Interim change issued to make minor editorial changes to Sections 4.1 and 4.3.1 at the request of DOE for acceptance of document. Also updated Section 2.4.2 to reference revision/change edition of procedure AP-3.11Q that was used to prepare this ICN.
0	02	11/03/00	Interim change issued to include lens of the eye dose calculations. This resulted in changes to the Executive Summary, and Sections 5.3.5.4, 5.3.6.1, 5.3.6.2, and 5.5.1.
0	03	07/10/01	Interim change issued to include a discussion of low-temperature repository operational modes. This interim change resulted in changes in Sections 4 and 5 as well as the addition of Appendix A. Updated Tables 4-1 through 4-3 to reflect most recent Classification Analyses of structures, systems, and components.

ACKNOWLEDGEMENT

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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.

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 potential 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 potential 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 potential 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).
- Shallow Dose Equivalent to Skin = 4E-2 rem.
- The Lens Dose Equivalent was not calculated in previous analyses of Category 2 design basis events. However, compliance with the Lens Dose Equivalent limit is achieved if the sum of the Skin Dose Equivalent (Shallow Dose Equivalent to Skin) and the Total Effective Dose Equivalent does not exceed 15 rem. For the maximum radiological consequence Category 2 design basis event at the Monitored Geologic Repository the sum of these does (= 6E-2 rem) is below the 15 rem limit.

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 collocated 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
- Shallow Dose Equivalent to Skin = 1E-1 rem/year
- Lens Dose Equivalent = sum of the worker Skin dose Equivalent (Shallow Dose Equivalent to Skin) and Total Effective Dose Equivalent = 1 E-1 rem/year (which is less than the 15 rem/year dose limit). Therefore, the calculated worker Lens Dose Equivalent is in compliance with the Lens Dose Equivalent limit

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
χ/Q	atmospheric dispersion factor (seconds/cubic meters)
λ	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 and Interim Changes 1 and 2 were prepared in accordance with the development plan for *Preclosure Safety Assessment* (CRWMS M&O 1999). Interim Change 3 was prepared in accordance with the *Technical Work Plan for: Preclosure Safety Analysis* (CRWMS M&O 2000). The original issue, Interim Change 1, and Interim Change 2 of this document were found to be in compliance with the requirements of this Technical Work Plan. 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.

No software routines, macros, or models as defined by AP-SI.1Q, *Software Management*, are used in this document. The word processing software used is off-the-shelf commercial software (Microsoft Word). Electronic management of information is controlled as identified in the *Technical Work Plan for: Preclosure Safety Analysis* (CRWMS M&O 2000, page 8).

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.

CRWMS M&O 2000. *Technical Work Plan for: Preclosure Safety Analysis*. TWP-MGR-SE-000010 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20001220.0017.

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-SI.1Q, Rev. 3, ICN 1. *Software Management.* Washington, D.C.: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ACC: MOL.20010515.0126.

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

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 potential repository surface facilities, however, flight procedures generally keep aircraft east of the potential 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 potential repository; therefore the military training route is three miles, at the closest, from the potential 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.

This screening analysis was based on a 100-year preclosure period, which is associated with the higher-temperature repository-operating mode. However, this screening process is valid for lower temperature operating modes as well. By expanding the range of thermal operating modes to include lower temperature modes, the preclosure period could be expanded from between 100 years to 325 years (DOE 2001). A discussion of the effect of extending the preclosure operational phase beyond 100 years on the screening of external events is included in Appendix A.

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Regulatory Guide 1.109, Rev. 1. 1977. *Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I.* Washington, D.C.: U.S. Nuclear Regulatory Commission. Readily available.

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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

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).

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. Based on the DBE definitions from Dyer (1999), DBE categories are defined as follows based on a high-temperature mode 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 λ is defined as event frequency. For the high-temperature mode preclosure operational period of 100 years, the threshold frequency for credible DBEs is interpreted to be greater than 1E-06 events per year.

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 24 years. Using this more realistic period would result in a Category 1 cutoff at 4.2E-2 per year and potentially allow more DBEs to be compared with the less restrictive Category 2 dose limits. Assuming a preclosure period of 325 years would lower the Category 1 cutoff to 3.1E-3 per year and lower the Category 2 cutoff to 3.1E-7 per year. However, since the waste handling and emplacement operations will be complete after approximately 24 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
Control and Tracking	Waste Emplacement/Retrieval System	Provide operational information to the Operations Monitoring and Control System; minimize the likelihood of 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 2001d
Disposal Containers	Canistered SNF Disposal Container Defense HLW Disposal Container DOE SNF Disposal Container Non-Fuel Components Disposal Container Uncanistered SNF Disposal Container Naval SNF Disposal Container	Provide containment and criticality control for SNF assemblies.	CRWMS M&O 1999c CRWMS M&O 1999d CRWMS M&O 1999e CRWMS M&O 1999f CRWMS M&O 1999g CRWMS M&O 1999h
Drip Shield	Emplacement Drift System	Provide containment, waste package protection, and heat transfer.	CRWMS M&O 2001f
Locomotives	Waste Emplacement/Retrieval System	Minimize the likelihood of 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 2001d
Modified Waste Package Transporter	Waste Emplacement/Retrieval System	Minimize the likelihood of 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 2001d
Small Canister Staging Racks	Canister Transfer System	Provide criticality control for defense HLW canisters.	CRWMS M&O 1999j

Table 4-1. Quality Level 1 Important to Safety Items (Continued)

SSC	MGR System	Important to Safety Function	Reference
Waste Package Transporter	Waste Emplacement/Retrieval System	Minimize the likelihood of 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 2001d
WHB Structure	WHB System	Provide containment of radioactive materials, radiation shielding, and protection of equipment from internal and external hazards.	CRWMS M&O 1999k

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	Maintain structural integrity in the event of a DBE, e.g., seismic event. Prevent interactions with QL-1 structures, systems, and components.	CRWMS M&O 1999b
	Carrier/Cask Handling System		CRWMS M&O 2001a
	Canister Transfer System		CRWMS M&O 1999j
	Disposal Container Handling System		CRWMS M&O 1999p
	Waste Package Remediation System		CRWMS M&O 2001e
Control and Tracking System	Assembly Transfer System	Minimize the likelihood of drop of assembly transfer basket during transfer of SNF assemblies.	CRWMS M&O 1999b
Control and Tracking System	Carrier/Cask Handling System	Provide operations support necessary for waste handling safety by controlling crane movement during handling of transportation casks.	CRWMS M&O 2001a
Control and Tracking System	Canister Transfer System	Support site-generated radiological waste collection and management functions.	CRWMS M&O 1999j
	Disposal Container Handling System		CRWMS M&O 1999p
	Waste Package Remediation System		CRWMS M&O 2001e
Cooling System	Assembly Transfer System	Collect and manage the site-generated radioactive waste generated in the SNF container cooling process.	CRWMS M&O 1999b
Covered Shuttlecars	Waste Emplacement/Retrieval System	Provide for radioactive particulate confinement.	CRWMS M&O 2001d
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 2001e
	Waste Emplacement/Retrieval System		CRWMS M&O 2001d
Dry Assembly Transfer Machine	Assembly Transfer System	Maintain structural integrity in the event of a DBE, e.g., seismic event. Minimize the likelihood of 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	Minimize the likelihood of breach of waste package in emplacement drift due to rockfall	CRWMS M&O 1999s
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
Invert	Emplacement Drift System	Provide support for mobile equipment in the drifts and for the drip shield and waste package/pallet combination.	CRWMS M&O 2001f
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	Minimize the likelihood of 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
Multi-Purpose Hauler	Waste Emplacement/Retrieval System	Provide for radioactive particulate confinement for breached waste packages.	CRWMS M&O 2001d
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 Primary, Secondary, and Tertiary Confinement Area Ventilation Systems	WHB Ventilation System	Mitigate the consequences of a facility DBE.	CRWMS M&O 2001c
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 2001e
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
Waste Package Emplacement Pallet	Emplacement Drift System	Prevent the waste package from shifting and impacting the drip shield.	CRWMS M&O 2001f
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	Provide 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
Bottom Lift Transporter	Waste Emplacement/Retrieval System	Permit remote operations and increased distance between facility operators and radiation sources; limits onsite worker doses from planned recovery operations.	CRWMS M&O 2001d
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
Control and Tracking System	Carrier Preparation Building Materials Handling System	Provide operations support necessary for waste handling safety; this system operates the crane, which permits remote operations and increased distance between facility operators and radiation sources.	CRWMS M&O 2001b
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
Emplacement Drift Forklift	Waste Emplacement/Retrieval System	Permit remote operations and increased distance between facility operators and radiation sources; limits onsite worker doses from planned recovery operations.	CRWMS M&O 2001d
Emplacement Drift Gantry Carrier	Waste Emplacement/Retrieval System	Permit remote operations and increased distance between facility operators and radiation sources; limits onsite worker doses from planned recovery operations.	CRWMS M&O 2001d
Emplacement Drift Restoration Locomotive	Waste Emplacement/Retrieval System	Permit remote operations and increased distance between facility operators and radiation sources; limits onsite worker doses from planned recovery operations.	CRWMS M&O 2001d
Emplacement Gantry	Waste Emplacement/Retrieval System	Permit remote operations and increased distance between facility operators and radiation sources; limits onsite worker doses from planned recovery operations.	CRWMS M&O 2001d
Extendable Conveyor	Waste Emplacement/Retrieval System	Permit remote operations and increased distance between facility operators and radiation sources; limits onsite worker doses from planned recovery operations.	CRWMS M&O 2001d
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
Load-Haul-Dump Loader	Waste Emplacement/Retrieval System	Permit remote operations and increased distance between facility operators and radiation sources; limits onsite worker doses from	CRWMS M&O 2001d

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

- 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 (based on the repository higher-temperature operating mode 100-year preclosure period)	"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	Less than once every 100 years, but	"(a) Other human-induced event sequences that have at

(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 this section and depicted in Table 5-3. The use of a 100-year preclosure period to screen internal events that could impact the surface facilities is bounding since surface-handling operations will be complete after approximately 24 years. The waste forms in the surface facility will not be vulnerable once the waste package emplacement operations are complete. The ventilation, monitoring, and performance confirmation activities to occur in the subsurface facility after emplacement is complete, but prior to permanent closure, are not expected to result in the breach of a waste package.

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).

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)	Shipping Cask

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

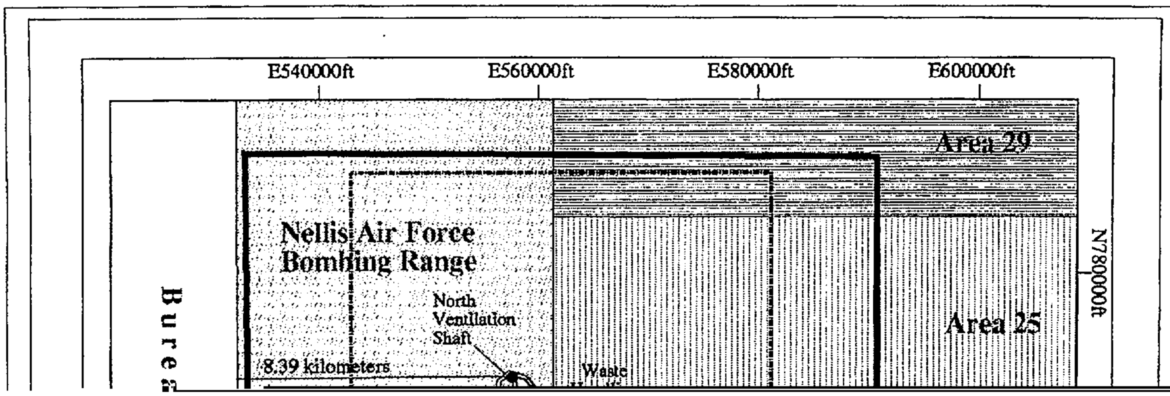
5.3.5.3 Atmospheric Dispersion Factors

Atmospheric dispersion factors (χ/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 χ/Q at the distances evaluated. The chronic χ/Q is an annual average, or best estimate, χ/Q. Category 2 event releases are modeled as "acute" releases and use the conservative, maximum sector 99.5 percentile acute (0.5 percent exceedance) χ/Q values for the distances evaluated. The selection of the maximum sector 99.5 percent χ/Q value was based on being larger than the 95 percent overall site χ/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 χ/Qs due to the increased dispersion associated with elevated releases.



E540000ft
E560000ft
E580000ft
E600000ft
Area 29
Nellis Air Force
Bombing Range
North
Ventilation
Shaft
8.39 kilometers
Waste
Area 25
N780000ft

- 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
- As per the NUREG-1567 (NRC 2000) Lens Dose Equivalent limit, the sum of the worker dose Skin Dose Equivalent (Shallow Dose Equivalent to Skin) and Total Effective Dose Equivalent (= 1E-1 rem) does not exceed the 15 rem limit.

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 100 meter distance used to calculate the dose to a noninvolved or co-located 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

The 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 associated with the higher repository temperature operating mode, 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). The BDBEs are discussed in this analysis for completeness.

The BDBEs considered in this section are internal event sequences that the MGR has specifically addressed in the preliminary design process 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

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

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 (565ºC)
	Cladding Failure in the ATS Dryer	ATS Dryer	None – analysis expected to demonstrate there is insufficient heat output to ignite cladding (565ºC) 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
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
		Subsurface	

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 Δk 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.

- 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 provides 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,

Regulatory Guide 8.8, Rev. 3. 1978. *Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be as Low as is Reasonably Achievable.* Washington, D.C.: U.S. Nuclear Regulatory Commission. Readily available.

Regulatory Guide 8.15, Rev. 1. 1999. *Acceptable Programs for Respiratory Protection.* Washington, D.C.: U.S. Nuclear Regulatory Commission. Readily available.

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8. FIRE PROTECTION

This section describes the MGR preclosure fire safety strategy which considers Yucca Mountain

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 backup 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

- 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

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

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 Assembly Transfer Canister Transfer Disposal Container Handling Waste Package Remediation	HVAC confinement: primary or secondary. Occupancy: WHB primary areas –never; secondary areas – never/ intermittently. Fire hazard: electrical wiring, cable and motors. All penetrations require fire seals.	Low to Moderate
WHB Primary Support	Operating Galleries Equipment Transfer Corridors Contaminated Equipment Rooms LLW Collection and Packaging Maintenance Equipment Rooms Weld Material Storage Maintenance Shop Forklift Staging and Servicing Waste Handling Operations Center	HVAC confinement: tertiary or none. Occupancy: normally occupied; some may contain radioactive contaminated materials. Fire hazard: ordinary combustible material used for operation/maintenance. All penetrations from areas of greater hazard require fire rated penetrations.	Moderate
WHB Pool Support	Pool Treatment Equipment Room	HVAC confinement: tertiary. Occupancy: normally occupied. May contain radioactive contamination. Fire hazard: ordinary combustible material used for operation/maintenance. 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. Occupancy: Normally occupied. Fire hazard: ordinary combustible material used for construction, office furnishings/supplies, or operation/maintenance. 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. Occupancy: Normally unoccupied; may contain contaminated equipment. Fire hazard: ordinary combustible material used for operation/maintenance; HEPA filters located here. All penetrations into area require fire rated penetrations.	Low to Moderate
WHB Misc. Building Support	Fire Protection, Electrical Equipment, Communications Equipment	HVAC confinement: none. Occupancy: Normally occupied. Fire hazard: electrical wiring, switchgear, cable, and electronic devices. Fire protection alarm system monitors & displays give status of fire alarms and suppression devices in the facility. 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. Withstands Design Basis external events: earthquake, extreme wind/tornado winds, and potential tornado missiles. Provide adequate fire protection of WHB operations from fires external to the WHB.	Low
WTB Processes	Solid Waste Processing Non-recyclable LLW Processing Recyclable Liquid LLW Processing	HVAC confinement: none. Occupancy: Normally occupied. Fire hazard: combustible solid wastes (contaminated); ordinary combustibles used in operations/maintenance.	Low to Moderate
WTB Operations	Offices Health Physics Staging Electrical Rooms HVAC Equipment Room	HVAC confinement: none. Occupancy: Normally occupied. 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. Occupancy: Normally occupied. 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

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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.

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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.

APPENDIX A

LOWER-TEMPERATURE OPERATIONAL PARAMETERS AND DESIGN BASIS EVENT SELECTION

APPENDIX A

LOWER TEMPERATURE OPERATIONAL PARAMETERS AND DESIGN BASIS EVENT SELECTION

In addition to the higher repository temperature operational mode and a corresponding preclosure operating period of 100 years, lower temperature operational modes (and corresponding longer preclosure operating periods) have been considered as part of the repository design process. This Appendix describes the impacts of lower-temperature operating modes and longer preclosure operating periods on the evaluation of design basis events.

A1. DESIGN FLEXIBILITY

The repository design documented in the *Engineering Files for Site Recommendation* (CRWMS M&O 2000a) is a flexible design that can be operated over a range of thermal conditions. Flexibility in the design allows the evolutionary process of the design to continue. Because the specific thermal criteria (e.g., waste package and drift wall temperature limits) that will be imposed on future design enhancements have not been selected, the repository design must be sufficiently flexible to allow operation under a wide range of potential thermal conditions. An assessment of repository performance over a range of thermal conditions will support the selection of the specific thermal criteria upon which future design enhancements will be based. The key aspects of design flexibility are (1) the ability of the repository design to support a range of construction approaches; (2) the capability to dispose of a wide range of waste container sizes; (3) the ability to support a range of thermal operating modes; and (4) the ability to continue to enhance the design to best achieve performance-related benefits identified through ongoing analyses (DOE 2001).

The current design of the potential repository can be operated to support a range of thermal operating modes. For example, the potential repository could be operated in a mode that keeps temperatures below the boiling point of water (96°C [205°F] at the repository elevation); or the potential repository could be operated such that the host rock reaches a temperature that is above the boiling point of water. Drift wall and waste package temperatures and relative humidity can be managed by altering several operational features of the design: (1) varying the thermal load to the repository by managing the thermal output of the waste packages; (2) managing the period and rate of drift ventilation prior to repository closure; and (3) varying the distance between waste packages in emplacement drifts. Other parameters, such as postemplacement natural ventilation, could also be used to reduce long-term repository temperatures. Altering design features, such as emplacement drift spacing, could also be used in conjunction with variations in operational parameters to achieve a lower-temperature repository environment (DOE 2001).

Table A-1 provides a comparison of operational parameters to produce several examples of lower-temperature repository operational modes. These examples are discussed in the following subsections.

Table A-1. Comparison of Estimates of Operational Parameters for Example Lower-Temperature Operating Modes (DOE 2001)

Parameters	Higher Temperature Operating Mode	Example Scenarios 1: Increased Waste Package Spacing and Extended Ventilation	2: De-Rated or Smaller Waste Packages	3: Increased Spacing and Duration of Forced Ventilation	4: Extended Surface Aging with Forced Ventilation	5: Extended Natural Ventilation
Variable Parameters						
Waste package spacing (m)	0.1	2	0.1	6	2	0.1
Maximum waste package thermal loading	11.8 kW	11.8 kW	<11.8 kW	11.8 kW	<11.8 kW	11.8 kW
Linear thermal loading objective (kW/m) at emplacement	1.45	1	1	0.7	0.5	1.45
Years of forced ventilation after start of the emplacement	50	75	75	125	125	75
Years of natural ventilation after forced ventilation period	0	250	250	0	0	>300
Dependent Parameters						
Size of pressurized water reactor waste packages	21 PWR	21 PWR	<21 PWR	21 PWR	21 PWR	21 PWR
Total excavated drift length (km)	~60	-80	~90	~130	~80	~60
Required emplacement area (acres)	~1,150	-1,600	~1,800	~2,500	~1,600	~1,150
Average waste package maximum temperature (ºC)	>96	<85	<85	<85	<85	<96

NOTE: PWR = pressurized water reactor.

A1.1 LOWER WASTE PACKAGE TEMPERATURE ACHIEVED THROUGH EXTENDED VENTILATION AND MINIMAL INCREASE IN DISPOSAL AREA

By extending the time during which loaded emplacement drifts are ventilated, the repository could be operated at lower temperatures with minimal increase in the disposal area. This subsection describes two example lower-temperature operating mode scenarios that are likely to satisfy the following three objectives:

1. Maintain average waste package surface temperatures below 85°C (185°F)
2. Ensure that 70,000 MTHM of waste fits within the upper block
3. Close and seal the repository within approximately 300 years.

Example Scenarios 1 and 2 could achieve a lower-temperature operating mode through extended forced and natural ventilation. Example Scenario 2 also includes a de-rated or a smaller waste package. The de-rated option involves placing fewer assemblies in a waste package than for which it is rated (i.e., placing 12 spent fuel assemblies in a waste package with a capacity for 21 spent fuel assemblies).

Example Scenario 1: Increased Waste Package Spacing and Extended Ventilation–In this example, loaded waste packages would be emplaced an average of about 2 m (6.6 ft) apart to create a 1 kW/m drift thermal load at emplacement. The drift-to-drift spacing would remain at 81 m (266 ft). For the first 75 years after the start of emplacement, fans would actively ventilate the drifts with an airflow rate of 15 m^3/s per drift. Because of the time required for emplacement, the drifts loaded last would be actively ventilated for 50 years. The repository would be allowed to ventilate naturally for 250 years. Other operational parameters would be unchanged.

Waste package and drift wall temperature responses vary with time such that:

1. During the forced ventilation period (0 to 75 years), the heating effect of the waste is steadily reduced until the fans are turned off in year 75.

2. From that point, natural ventilation flow continues to remove heat; however, the reduced flow rate (3 m^3/s) takes several years to turn the temperature response to a downward trend (approximately years 70 to 100).

3. The natural ventilation flow rate gradually decreases over time as the waste decays due to the fact that the heat from the waste is the main driver in inducing convective currents in airflow.

4. The temperature rises in years 100 to approximately 130 due to the abrupt reduction in the natural ventilation flow rate. After year 130, the reduced flow rate starts reducing temperatures until year 300, when the repository would be sealed and closed.

5. At that point, the lack of ventilation would force a steep rise in the drift temperatures until the capacity of the host rock to transfer heat away from the waste packages starts to have a regulating effect on drift temperatures and establishes a very slow downward trend in thermal response over a period of 1,000 years or longer.

Note that in the more natural situation of a steady decrease in the natural ventilation flow rate, the calculated thermal response would actually show a steady decrease in drift temperatures across most of the natural ventilation period.

Some of the implications of this scenario are: (1) the flexibility to readily adjust to a higher-temperature operating mode in drifts loaded later by moving waste packages closer together; (2) a requirement for additional drift excavation to accommodate more widely spaced waste packages; (3) increased complexities in projecting the thermal-hydrologic response of the repository because the widely spaced waste packages would act more like point heat sources within drifts; and (4) the programmatic uncertainty associated with the protracted period of natural ventilation (DOE 2001).

Example Scenario 2: De-Rated or Smaller Waste Packages–In this scenario, the thermal output of the waste packages is reduced by limiting waste package loading. This can be achieved by limiting the number of spent nuclear fuel assemblies to less than the waste package design capacity (de-rating) or replacing the large waste packages (e.g., those containing 21 pressurized water reactor fuel assemblies) with smaller waste packages (e.g., those that would contain 12 pressurized water reactor fuel assemblies) that have a lower thermal output.

Waste packages would be placed end-to-end within the drifts to create a 1 kW/m linear thermal load at emplacement. Other operational parameters of this scenario are identical to those in Scenario 1.

The primary difference between Scenario 2 and Scenario 1 is a potential reduction in the complexity of modeling the thermal-hydrologic response of the repository because the thermal loading would more closely resemble a line load. A second difference is the increase in the number of waste packages required, which could result in an increase in the total excavated drift length and the total area required for emplacement.

A1.2 LOWER WASTE PACKAGE TEMPERATURE ACHIEVED THROUGH INCREASED DISPOSAL AREA AND LIMITED NATURAL VENTILATION PERIOD

Lower-temperature operating goals can also be achieved with a limited increase in the forced ventilation period by increasing the area used for emplacement. The three objectives for this set of examples are:

1. Maintain average waste package surface temperatures below 85°C (185°F)
2. Ensure that 70,000 MTHM of waste fits within the upper and lower blocks
3. Close and seal the repository within approximately 125 years.

Example Scenarios 3 and 4 could achieve lower temperatures through extended forced ventilation. Example Scenario 4 also includes extended surface aging of spent nuclear fuel (DOE 2001).

Example Scenario 3: Increased Spacing and Duration of Forced Ventilation–Lower temperatures could be achieved with a limited increase in the duration of the preclosure period by emplacing waste packages an average of approximately 6 m (20 ft) apart. This spacing would create a drift thermal load at emplacement of approximately 0.7 kW/m. The drift-to-drift spacing would remain at 81 m (266 ft). The loaded drifts would be actively ventilated for 125 years from the start of waste emplacement; the drifts that are loaded in the last year of emplacement operations would receive 100 years of forced ventilation. The implications of this approach include: (1) a preclosure period comparable to the higher-temperature operating mode preclosure period of approximately 100 years; (2) an increase in waste package spacing which would result in point-thermal loading of drifts and may give rise to increased complexities in modeling the thermal-hydrologic response of the host rock; (3) increases in the total excavated drift length and the total area required for emplacement; and (4) requirements for additional years of forced ventilation, maintenance, and other site and operations support (DOE 2001).

Example Scenario 4: Extended Surface Aging with Forced Ventilation–In this example, surface aging of the hotter portion of the commercial spent nuclear fuel inventory, combined with the spacing of waste packages to approximately 2 m (6.6 ft) apart within the drifts, reduces the linear thermal load to approximately 0.5 kW/m at emplacement. Surface aging of the hottest wastes would extend the total emplacement period from approximately 25 years to 50 years. However, initiation of repository operations would not be delayed because the cooler commercial spent nuclear fuel, along with the generally cooler DOE waste forms, could be emplaced immediately while the hotter commercial spent nuclear fuel cools through aging. To meet the goal of a maximum waste package surface temperature of 85°C (185°F), forced drift ventilation would continue for approximately 125 years from the start of waste emplacement, with the last drifts loaded receiving 75 years of forced ventilation. At the end of the operating period, the potential repository would be closed and sealed, with no provision for extended natural ventilation. The implications of this scenario include: (1) the ability to accommodate the waste packages with drift-to-drift spacing of 81 m (266 ft) in the areas currently characterized for a repository; (2) a preclosure period comparable to that of the higher-temperature operating mode; (3) a smaller increase in total drift length and disposal area than in Scenario 3; (4) an increase in the spacing of waste packages, which may result in thermal point-loading that introduces additional complexities in modeling the thermal-hydrologic response of the potential repository; and (5) a longer emplacement period and additional fuel handling activities, which could increase the preclosure safety risk (DOE 2001).

A1.3 LOWER ROCK TEMPERATURE AND IN-DRIFT RELATIVE HUMIDITY THROUGH INDEFINITE NATURAL (PASSIVE) VENTILATION

A third approach for meeting the goals of lower-temperature operating modes is to keep the temperature of the host rock below the boiling point of water and maintain in-drift relative humidity below 50 percent by incorporating passive natural ventilation into repository operations. A potential repository that maintains a low relative humidity is possible because of the natural characteristics of the Yucca Mountain site, including the arid environment and thick unsaturated zone.

An example lower-temperature scenario using this approach is framed around the following objectives:

1. Maintain in-drift relative humidity below 50 percent
2. Maintain the host rock temperature below the boiling point of water
3. Ensure that 70,000 MTHM of waste fits within the upper block (DOE 2001).

Example Scenario 5: Extended Natural Ventilation–A lower-temperature dry repository can be created by increasing the duration of forced ventilation to approximately 75 years after the start of emplacement (50 years after the last drift is loaded) followed by an indefinite period of natural ventilation (DOE 2001). However, a performance requirement for the potential repository directs that the repository design shall allow the repository to remain open for up to 300 years following final waste emplacement (Curry 2001, Section 5.1.1.1) (expected to last approximately 24 years). Therefore, because this Example Scenario has no determinable closure date, it will not be considered in this Appendix.

A2. PRECLOSURE RADIOLOGICAL SAFETY CONSIDERATIONS ASSOCIATED WITH THE LOWER-TEMPERATURE REPOSITORY OPERATIONAL MODES

The primary preclosure radiological safety considerations for the example lower-temperature repository operating modes involve such changes as increases in the total excavated drift length, increases in the length of the forced and natural ventilation periods (which increase the repository preclosure period), modification of the subsurface ventilation system to accommodate and facilitate natural ventilation, changes with waste package numbers/sizes, and the addition of a surface aging pad.

A2.1 PRECLOSURE SAFETY CONSIDERATIONS

Specific preclosure safety considerations associated with each example lower-temperature repository operating mode are described in the following paragraphs.

Example Scenario 1: Increased Waste Package Spacing and Extended Ventilation–In this example, the total excavated drift length and required emplacement area would be increased to accommodate increased waste package spacing. Increasing the emplacement area may impact the likelihood of a rockfall striking a waste package. Forced ventilation will occur for the first 75 years after the start of emplacement, followed by 250 years of natural ventilation. Increasing the preclosure time period may impact the selection of design basis events. Other operational parameters would be unchanged.

Example Scenario 2: De-Rated or Smaller Waste Packages–In this scenario, waste packages would be loaded at a decreased capacity or else a greater number of smaller waste packages is proposed than was proposed for the higher-temperature repository mode design or the other example designs for lower repository temperature operating modes. Because the waste packages could be smaller in size, more would be required to emplace the same amount of spent nuclear fuel. An increase in the number of waste packages may impact the likelihood of occurrence of design basis events. As with Example Scenario 1, forced ventilation would occur for the first 75 years after the start of emplacement, followed by 250 years of natural ventilation. Increasing the preclosure time period may impact the selection of design basis events. In addition, an increase in the number of waste packages required would result in an increase in the total excavated drift length and the total area required for emplacement. Increasing the emplacement area may impact the likelihood of a rockfall striking a waste package.

Example Scenario 3: Increased Spacing and Duration of Forced Ventilation–Waste packages would be placed an average of approximately 6 m (20 ft) apart, requiring a larger total excavated drift length and a larger required emplacement area than those required in the other example lower temperature operating scenarios as well as the higher-temperature operating scenario. Increasing the emplacement area may impact the likelihood of a rockfall striking a waste package. The loaded drifts would be actively ventilated for 125 years from the start of waste emplacement; there would be no natural ventilation period following the active ventilation period. Increasing the preclosure time period may impact the selection of design basis events.

Example Scenario 4: Extended Surface Aging with Forced Ventilation–In this example, surface aging of the hotter portion of the commercial spent nuclear fuel inventory would take place on a pad located on the surface, thereby increasing the size of the total area of the surface facilities. The spacing of waste packages in the repository would be increased, thereby requiring an increase in the total excavated drift length and the total area required for emplacement (however, smaller than those in Example 3). Increasing the emplacement area may impact the likelihood of a rockfall striking a waste package. Surface aging of the hottest spent nuclear fuel would extend the total emplacement period from approximately 25 years to 50. Forced drift ventilation would continue for approximately 125 years from the start of waste emplacement; at the end of the operating period, the repository would be closed and sealed with no provision for extended natural ventilation. Increasing the preclosure time period and the additional fuel handling activities may impact the selection of design basis events.

Example Scenario 5: Extended Natural Ventilation–Not considered, as discussed in Section A.1.3.

Other changes inherent in these example lower-temperature operating modes have little or no impact on preclosure radiological safety. These changes include additional spacing between the emplaced waste packages and the increased use of the subsurface ventilation system in removing decay heat from the repository.

A2.2 PRECLOSURE SAFETY EVALUATION METHOD

The safety evaluation method used to evaluate the preclosure suitability of the Yucca Mountain site is not affected by consideration of the example lower-temperature operating modes and, therefore, is identical to the method described in Section 5.

Proposed 10 CFR 963.13 (64 CFR 67054) requires that the preclosure safety evaluation consider (1) a preliminary description of the site characteristics, the surface facilities, and the underground operating facilities [10 CFR 963.13(b)(1)]; (2) a preliminary description of the design bases for the operating facilities and associated limits on operations [10 CFR 963.13(b)(2)]; (3) a preliminary description of potential hazards, event sequences, and consequences [10 CFR 963.13(b)(3)]; and (4) a preliminary description of the structures, systems, components, equipment, and operator actions intended to mitigate or prevent accidents [10 CFR 963.13(b)(4)]. These preliminary descriptions for the repository design in the higher-temperature operating mode are contained in Section 4. The following Sections of this Appendix describe the changes in those preliminary descriptions when the example lower-temperature operating modes are considered.

A3. FACILITY DESIGN BASES AND LIMITS ON OPERATIONS

One example of a lower-temperature repository operating mode (Example Scenario 4) included the addition of a surface aging pad for spent nuclear fuel. Besides this change, the site characteristics and surface facilities are not significantly affected by consideration of the lower-temperature repository operating modes described in Section A1 of this Appendix. As discussed in Section A2 of this Appendix, the facility changes that could potentially affect preclosure safety are an increase in the number of emplacement drifts excavated, modifications of the subsurface ventilation system to accommodate and facilitate natural ventilation, an increase in the number of smaller waste packages, and the possible addition of the surface spent nuclear fuel aging area. Other aspects of the site characteristics and facility descriptions are bounded by the evaluation in Section 5.

Consideration of the example lower-temperature repository operating modes described in Section A1 of this Appendix introduces minor changes to the facility design bases. These changes are due to an extended preclosure period, modifications to the subsurface ventilation system to accommodate and facilitate natural ventilation, and the installation of the spent nuclear fuel aging pad.

As with the higher-temperature repository operating mode, the lower-temperature repository operating modes rely on the waste packages to provide containment of radioactive material during the preclosure period. The emplacement handling system, the pallet, and the ground support system are additional preventive features that ensure that there are no credible events that can compromise the waste package containment integrity. In the same manner, the transportation cask (or other cask/canister) to be potentially used to store the spent nuclear fuel on the surface aging pad would also rely on the chosen cask/canister to provide containment during the aging period.

The natural ventilation system is a passive system that requires minimal periodic inspection and monitoring to ensure that it performs its intended function. One threat to an operating mode employing natural ventilation is the full or partial collapse of underground openings or ventilation shafts. Although a rockfall-induced blockage of a natural ventilation pathway is not expected to compromise the thermal performance of a lower-temperature operating mode, additional design features could be incorporated to prevent or mitigate such an event (a discussion of the rockfall event is provided in Section A4.1 of this Appendix).

Additional equipment would likely be used during the latter portion of the natural ventilation and monitoring phase to accommodate limited remote monitoring of the subsurface facilities. Additional limits on operations might be also imposed to ensure that thermal performance objectives are met. However, none of these additions are likely to affect the health and safety of workers or the public.

A4. PRELIMINARY DESCRIPTION OF HAZARDS, EVENT SEQUENCES, AND CONSEQUENCES

Extending the preclosure operational period to 325 years could have an effect on the categorization process for design basis events, as described in Section 5. The potential impact on the categorization process results from the definitions of event categories presented 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): the definitions depend on the probability of an event occurring "before permanent closure." For example, Category 2 design basis events are defined as "other natural and man-induced events that have at least one chance in 10,000 of occurring before permanent closure of the geologic repository." Based on a preclosure period of 325 years, Category 2 events could be interpreted as those having an annual frequency of occurrence of at least 3.1E-07 per year (i.e., 1/10,000÷325). In this case, beyond design basis (non-credible) events would include those natural and human-induced event sequences with an annual frequency less than 3.1E-07 per year.

Likewise, Category 1 design basis events are defined as "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" (Dyer 1999). Thus, the lower cutoff frequency for Category 1 design basis events would become 3.1E-03 per year (i.e., 1/325).

In Section 5, the impact of a 325-year preclosure period on the design basis event frequency thresholds was addressed. The reasoned conclusion was that a 100-year preclosure period is expected to bound the design basis events since the waste handling and emplacement operations will be completed after approximately 24 years (DOE 2001 Section 2.3.4.5). The 100-year preclosure period evaluated in Section 5 is expected to be conservative for internal (i.e., human-induced) design basis events, regardless of the preclosure period, because the waste forms are not vulnerable to an internal event that would cause a radionuclide release after the waste handling and emplacement operations are completed. For example, in the case of a 325 year preclosure period, the waste handling and emplacement operations would still be completed after approximately 24 years. The period beyond emplacement, but prior to permanent closure, would include monitoring and performance confirmation activities, which are not expected to result in internal design basis events that would lead to a release of radionuclide material.

Several of the example lower-temperature repository operational modes include changes to the repository design, including increasing the total excavated drift length (lower-temperature operating mode Examples Scenarios 1 through 4). Several of the design basis events evaluated in Section 5 involved transporter-related events whose frequency could increase due to the increase in size of the required emplacement area (e.g., transporter collisions at normal operating speeds; bed plate rolls out of waste package transporter; transporter derails without tip over, but with waste package restraint failure; transporter derails with tip over; transporter door closes on waste package; operation of emplacement gantry causes waste package collision). However, there is no release of radiological material associated with these events due to the protection provided by the waste package and the transporter. No new bounding design basis events are expected to result from changes to the size of the emplacement area or the total excavated drift length.

Lower-temperature operating mode Example 2 included an increase in the number of waste packages (switching to smaller/de-rated waste packages). No new bounding design basis events are expected to result from this change. However, the frequency of several events involving the handling of waste packages would increase due to the increase in the number of required handlings of these smaller or de-rated waste packages. These events include such Category 2 events as the unsealed disposal container collision, unsealed disposal container drop and slapdown, and a handling equipment drop onto an unsealed disposal container. The frequency of other events associated with the transport and handling of these smaller or de-rated waste packages may increase as well, including waste package drops, falling objects striking waste packages, waste package slapdowns, waste package collisions, and transporter-related events. However, although the frequency of these events may increase, there is no radiological material release associated with these events. In each case, the release is prevented by the waste package (or, in some cases other structures, systems, or components in addition to the waste package, such as lifting systems or the waste package transporter), as described in Section 5. No new bounding design basis events will result from the use of an increase in quantity of (smaller/de-rated) waste packages.

A review of the *MGR External Events Hazards Analysis* (CRWMS M&O 2000b) indicates that no additional external events or natural phenomena would be expected due to the flexible repository operational modes. The external design basis events evaluated in Section 5 are appropriate for a preclosure period of 100 years or 325 years. The selection of external design basis events (e.g., loss of offsite power) and natural phenomena (e.g., earthquakes) evaluated in Section 5 are not expected to be impacted by a 325-year preclosure period. Under the NRC's precedence of deterministic licensing for natural phenomena, structures, systems, and components important to radiological safety will be designed to withstand such design basis events.

The *MGR External Events Hazards Analysis* employed a generic checklist of 53 categories of natural phenomena and man-made hazards. The initial list is considered comprehensive and applicable to the potential repository site for any operational period. Thus, no additional initial events are postulated for the increased preclosure periods discussed in Section A1. However, the results of systematic screening of events could be affected by the operational time and, for some events, by the physical size (i.e., area or "footprint") of the respective surface and subsurface facilities.

Each of the 53 event categories was screened for applicability to the MGR preclosure operational period using a five-step process. The first step was a deterministic screening to eliminate events that cannot exist at the Yucca Mountain site. For example, events involving coastal phenomena were eliminated. The second step screened out long-term phenomena that require thousands of years for perceptible changes to take place and are, therefore, not of concern as an initiating event during the preclosure period, irrespective of its duration of 100 to 300 years. For events that passed the first two screening criteria, the third step considered whether or not radiological consequences of an event could be significant during the preclosure period. Although this step resulted in a few events being screened out, this step was found to be applicable or indeterminate in most cases, so events were subjected to the fourth step.

In the fourth step, a postulated initiating event was screened out if its estimated frequency was less than 1.0E-06 per year and, thereby, defined as a Beyond Design Basis Event. As noted in Section 5.2 of the *MGR External Events Hazards Analysis*, the analysis also considered the effect of an alternative preclosure period on the order of 300 years and concluded that the factor of three would not change the results of the frequency screening. As discussed in Section 5.1.1.3.2, no new credible external events are addressed as a consequence of longer preclosure periods extending to 325 years.

Some of the event screening, however, assumed no significant change in the land usage in the vicinity of the potential site. Thus, an event involving a pipeline carrying hazardous material near the surface facilities was screened out because there are no industrial activities identified in the area at the present time. This situation could change over a preclosure period ranging from 100 years to 325 years. Further, the evaluation of frequencies of events such as potential aircraft hazards at the site may change over such a period. However, the risk of such enterprises would be evaluated in periodic updates of the *Integrated Safety Analysis* of the preclosure operations as a condition imposed by the potential license.

Design options for the potential repository as part of the example lower-temperature repository modes (including an expanded emplacement area and a potentially larger surface area due to the addition of a surface aging area) require a reexamination of two of the external design basis events evaluated in Section 5: the rockfall event and the aircraft crash event. The rockfall and aircraft crash events are evaluated with consideration given to the example lower-temperature repository operating modes in Sections A4.1 and A4.2 of this Appendix, respectively.

The potential exists for the addition of several design basis events associated with the placement of spent nuclear fuel on the surface as part of the extended surface aging scenario (lower-temperature repository mode Example 4). The design for this potential facility is conceptual at this time. It is expected that spent nuclear fuel stored in shipping casks or other casks/canisters on a pad would be exposed to the same natural phenomena as the surface facilities (DOE 2001). In addition, there may be several design basis events associated with placing, moving, and removing these casks/canisters. These events are not expected to result in a radioactive material release since the spent nuclear fuel will be stored in casks designed to withstand these potential events. As with the waste packages in the subsurface facility, these casks/canisters would be designed and relied upon to provide containment of the spent nuclear fuel during the aging time required during this preclosure period.

A4.1 ROCKFALL HAZARD

Several of the example lower-temperature repository operational modes included an increase of the total excavated drift length (lower-temperature operating mode Examples 1 through 4). An increase in the size of the emplacement area has the potential to increase the frequency of a rockfall. Rockfall has the potential to damage a waste package as well as the potential to have a deleterious effect on subsurface ventilation.

One threat to lower-temperature repository operating modes that use natural ventilation is the full or partial collapse of underground openings. The severity of the impact would depend on the design of the opening and the location and size of the rockfall. Rockfall in an emplacement drift could create higher resistance in the affected flow path, causing the airflow to rebalance – less air would flow in the damaged drift and more would flow in undamaged drifts. If the fall occurred in a main drift, emplacement drifts that received or exhausted air through that pathway could experience reduced airflow. Although rockfall may reduce natural ventilation flow, it is unlikely that it would significantly restrict it because a natural fall of material typically leaves pathways through which air can find its way (i.e., fallen rocks leave behind holes that become alternative flow paths).

An analysis of rockfall onto the waste package for rock masses greater than or equal to the design basis 6-MT rock was evaluated in *Preclosure Design Basis Events Related to Waste Packages* (CRWMS M&O 2000c, Section 6.3.2.1.5.1) and found to be beyond design basis (non-credible) based on (1) a 100-year preclosure period; (2) a seismic event recurrence frequency of 10,000 years (used to predict the number of key blocks greater than 6 MT for a given rock unit); (3) redundant ground supports (at least one steel set and one rock bolt); and (4) a ground control inspection and repair program with a 3-year inspection interval. This evaluation also addressed the impact of a longer preclosure period, up to 300 years, on the probability of a ground control failure and the potential for a greater than 6-MT rockfall to be a credible event. The report concluded that the event could be considered beyond design basis (non-credible), even with a 300-year preclosure period. This conclusion was based on the assumption that an inspection and repair program would be maintained throughout the 300-year preclosure period.

This event was recently reexamined in *Update to Waste Package DBE Rockfall Analysis* (BSC 2001) with consideration given to the lower-temperature repository operational modes and preclosure periods extending to 325 years. Several of the lower-temperature repository operating mode examples provided in Section A1 involve expanding the total excavated drift length, thereby increasing the number of potential key blocks that may be encountered in the repository. In the *Update to Waste Package DBE Rockfall Analysis* it was assumed that key blocks greater than or equal to 10 MTs are sufficient to cause a waste package breach. This analysis also assumed a common-cause failure of the steel sets supporting the key block due to the fact that the number of steel sets required to support a key block is unknown.

The *Update to Waste Package DBE Rockfall Analysis* concluded that the preclosure rockfall event scenario is estimated to be incredible (beyond design basis) for those preclosure scenario time periods that are less than 325 years (the total preclosure time period for lower-temperature repository operating mode Example Scenarios 3 and 4). The other two example scenarios considered in this Appendix (lower temperature repository operating mode Example Scenarios 1 and 2) were borderline credible due to the longer length of their total preclosure periods (325 years). At this time, the final design of the potential repository is not complete. However, potential design optimizations that would make this event non-credible (beyond design basis) include:

- Early placement of drip shields: the placement of the drip shields at the end of the forced ventilation period will most likely enhance natural ventilation cooling of waste packages and will also preclude the consequences of rockfall during the remaining preclosure period.

- Use of more rock bolts near the key blocks: if more than one rock bolt is to be placed in each key block, credit can be taken for the necessity of multiple rock bolt failures to occur (to cause a rockfall) in a manner similar to that used for steel sets in the analysis of this event in the *Update to Waste Package DBE Rockfall Analysis*.

- Waste package placement strategy: this policy would restrict placement of waste packages in drift areas where these key blocks are located. Even if the key block detection method was only 70 percent accurate, the reduction in the estimated frequency of a rock falling onto at least one waste package would be sufficient enough for this frequency to be lower than the credibility threshold for all lower-temperature repository operating mode example preclosure time periods. It should be noted that this policy is compatible with the current potential repository layout, which includes a 10 percent unexcavated contingency to allow for unexpected circumstances such as inadequate ground conditions (BSC 2001).

Thus, with consideration given to any or all of these design optimizations in the final potential repository design, the rockfall event is judged to be non-credible (beyond design basis) for the potential repository higher- and lower-temperature operating modes described in Section A1.

A4.2 AIRCRAFT CRASH HAZARD

As stated in Section 5.1, the *MGR External Events Hazards Analysis* identifies the basis for screening external events, including the aircraft crash event. The results from the *MGR Aircraft Crash Frequency Analysis* (CRWMS M&O 1999) were close to, but below, the screening criterion presented in Section 5.1.1.3.2.D. The aircraft crash event was determined to be a non-credible (beyond design basis) event. Therefore, this event was screened out in the *MGR External Events Hazards Analysis.*

The inputs required for the analysis of this event will change as more flight information is available and the facility design evolves. One factor involved in the calculation of the aircraft crash event frequency is dependent on the surface facility size; this parameter is expected to change as the potential repository design evolves. Example Scenario 4 includes an increase in the size of the surface area with the potential addition of a pad to be used as part of a lower-temperature repository operating mode that utilizes extended surface-aging of spent nuclear fuel. This design option could affect the analysis of the aircraft-crash hazard. Conservative assumptions made in the *MGR Aircaft Crash Frequency Analysis* allow variations in the input parameters without impacting its conclusion.

These conservative assumptions include:

- Total area sizes were used for the waste handling building and the effective area of the waste treatment building, the carrier preparation building, and the transportation cask parking areas. Total area sizes were used rather than limiting the footprint to areas with sufficient radioactive material that, if released, would exceed boundary dose limits.

- A preclosure period of 100 years was assumed rather than the approximately 24-year period for the surface facility and emplacement operations. This period is used in translating the regulatory probability limit of 1 in 10,000 to a frequency (per year) limit and the shorter duration will result in a higher allowable frequency limit. The balance of the preclosure period that could potentially extend to over 300 years involves radioactive material located underground and not exposed to the aircraft crash event. Any spent nuclear fuel cooled on the surface, as discussed in Section A1 of this Appendix, as part of lower repository temperature operational mode Example Scenario 4, would be emplaced during a 50-year period (DOE 2001), which is also less than the 100-year preclosure period assumed in the analysis.

- The analysis included the over-flights traversing the entire Nevada Test Site (NTS) in the aircraft counts. Monitoring of aircraft over the NTS since the completion of the *MGR Aircraft Crash Frequency Analysis* indicates that only 10 percent of the flights used in the analysis are within 3.5 miles of the potential surface facility.

- The bounding case in the *MGR Aircraft Crash Frequency Analysis* assumed a crash rate for small Air Force attack and fighter aircraft that was higher than the specific crash rate for multi-engine aircraft. Input from Nellis Air Force Base indicates that multi-engine aircraft will dominate the aircraft types anticipated over the NTS. Decreased crash rates for specific aircraft types were not considered in the *MGR Aircraft Crash Frequency Analysis.*

- The *MGR Aircraft Crash Frequency Analysis* took no credit for the pilot's ability to divert the aircraft away from surface facilities. The NRC has allowed taking credit for the pilot's ability to avoid surface facilities in 90 percent of the crashes, as reported in *Safety Evaluation Report Concerning the Private Fuel Storage Facility*, (NRC 2000).

Potential surface facility designs continue to evolve. Given the conservative assumptions used in determining the aircraft crash frequency, the conclusions of the *MGR Aircraft Crash Frequency Analysis* are considered reasonable, and the aircraft crash event is judged to be a non-credible (beyond design basis) event for all of the potential repository temperature operational modes. This conclusion will be reconfirmed prior to License Application when additional flight information is available and the design of the potential repository is more mature.

A4.3 DESIGN BASIS EVENT DOSE CONSEQUENCES

The Category 1 and Category 2 design basis event dose consequences for onsite workers and offsite members of the public presented in Section 5 will not be significantly impacted by the selection of repository temperature operating mode and, therefore, are considered bounding. If more waste packages are handled (as with the de-rated or smaller waste packages described as part of Example 2 of the lower-temperature repository operating modes), the worker dose could potentially increase. Normal operation worker dose exposures will be controlled by the facility as low as is reasonably achievable (ALARA) program, as discussed in Section 7. Therefore, it is expected that the dose results presented in Section 5 will bound such changes.

A5. PRELIMINARY DESCRIPTION OF STRUCTURES, SYSTEMS, COMPONENTS, EQUIPMENT, AND OPERATOR ACTIONS INTENDED TO MITIGATE OR PREVENT ACCIDENTS

Consideration of the example lower-temperature repository operating modes described in Section A1 of this Appendix has no impact on the evaluation of the structures, systems, components, equipment, and operator actions intended to mitigate or prevent accidents presented in Section 5.

The ground support system and subsurface ventilation system (as modified to accommodate and facilitate natural ventilation) play an important role in achieving repository preclosure performance objectives. The ground support system in the emplacement drifts, the inverts, dripshields, the waste emplacement system, and the waste packages are important to safety structures, systems, and components. During the active ventilation phase with full human access to the repository, it is assumed that there would be sufficient time (approximately three weeks [DOE 2001]) to repair potential failures in the ground support or subsurface ventilation systems before they significantly affect thermal performance. During the natural ventilation phase, it is assumed that air monitoring and the capability to respond to off-normal events will be sufficient to repair failed systems before they significantly affect performance. The classification of the subsurface ventilation system will be reexamined once the potential repository design is finalized to examine the impact of preclosure safety on the classification of this system.

The preclosure suitability criteria outlined in proposed 10 CFR 963.14 include: (1) the ability to contain radioactive materials and limit releases [10 CFR 963.14(a)]; (2) the ability to implement control and emergency systems to limit exposure to radiation [10 CFR 963.14(b)]; (3) the ability to maintain a system and components that perform functions important to safety [10 CFR 963.14(c)]; and (4) the ability to preserve the option to retrieve wastes during the preclosure period [10 CFR 963.14 (d)]. This section describes the impacts on the evaluation of the preclosure suitability criteria when the example lower-temperature repository operating modes are considered.

A5.1 CONTAINMENT OF RADIOACTIVE MATERIAL AND LIMITING RELEASES

Containment of waste during emplacement operations and until permanent closure of the repository is provided by the waste packages. If a pad is constructed to provide for extended surface aging (lower-temperature repository operating mode Example Scenario 4), the cask/canister utilized for the aging process would provide containment during preclosure operation of the pad.

Prevention features of the transporter, rail system, emplacement handling equipment, pallet, and ground support ensure that no credible events can occur that are beyond the design basis of the waste package for containment integrity.

The natural ventilation system and extended operational period associated with the example lower-temperature operating modes outlined in Section A1 of this Appendix have no impact on the ability of the facility to contain radioactive material and limit releases during the preclosure period.

A5.2 IMPLEMENTATION OF CONTROL AND EMERGENCY SYSTEMS

During the period of active, forced ventilation of the subsurface facilities, the control and emergency systems relied on to limit radiation exposures to workers and members of the public are identical to those required for the higher thermal temperature repository operational mode. During the period of natural ventilation of the subsurface facilities, measurements at the intake and exhaust shafts (e.g., airflow, temperature, humidity, radiation) and institutional controls (e.g., onsite or remote security) would be maintained for 325 years (DOE 2001). There are no credible internal or external design basis events postulated for this period that would necessitate emergency systems to protect onsite workers or members of the public from a radiological release.

A5.3 MAINTAINING A SYSTEM AND COMPONENTS THAT PERFORM THEIR INTENDED SAFETY FUNCTION

The natural ventilation system and extended operational period associated with the example lower-temperature repository operating modes have no impact on the ability of the facility structures, systems, and components that are important to safety to perform their intended safety functions. After final emplacement operations, neither the subsurface ventilation system nor the modifications to accommodate and facilitate natural ventilation are expected to be important to preclosure safety.

A5.4 PRESERVING THE OPTION TO RETRIEVE WASTES

Under normal conditions, waste package retrieval uses the same equipment and facilities as emplacement operations, but in the reverse order. Alternative waste package retrieval equipment has been identified for off-normal conditions. Various scenarios of off-normal retrieval have been studied, and conceptual use of such equipment has been demonstrated (DOE 2001, Section 2.3.4.6). The ability to retrieve wastes is preserved with the example lower-temperature repository operating modes described in Section A1 of this Appendix.

A6. CONCLUSIONS

The lower-temperature repository operating mode examples described in Section A1 of this Appendix do not introduce any new bounding radiological hazards or design basis event sequences that were not considered in Section 5.

The lower-temperature repository operating mode scenario involving extended surface aging (Example 4) may introduce new design basis events associated with placing, moving, or retrieval of the casks/canisters involved. However, none of these events are expected to bound the design basis events previously analyzed in Section 5.

The rockfall and aircraft crash external events were reevaluated with consideration given to design changes that may occur as part of the lower-temperature operating mode examples. The design changes to the potential repository are not expected to change the conclusions presented in Section 5 concerning the impacts of these two hazards. The analysis of the effect of the lower-temperature repository operating modes on the rockfall hazard (BSC 2001) concluded that rockfall is non-credible for preclosure scenarios with time periods that are less than 325 years (Example Scenarios 3 and 4) and, therefore, slightly credible for Example Scenarios 1 and 2. However, based on potential design optimizations that could be included in the final design of the potential repository (the potential repository design is not complete at this time), this event is judged to be non-credible (beyond design basis). The conclusions previously achieved concerning the aircraft crash hazard are considered reasonable. This hazard is considered to be a non-credible (beyond design basis) event for the potential repository higher- and lower-temperature operating modes. As the proposed surface facility design is finalized and more information become available, the conclusions concerning those events will be confirmed.

A7. REFERENCES

A7.1 DOCUMENTS CITED

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A7.2 CODES, STANDARDS, REGULATIONS, AND PROCEDURES

64 FR 67054. Office of Civilian Radioactive Waste Management; General Guidelines for the Recommendation of Sites for Nuclear Waste Repositories; Yucca Mountain Site Suitability Guidelines. Proposed rule 10 CFR 963. Readily available.