DRAFT DISCLAIMER

This contractor document was prepared for the U.S. Department of Energy (DOE), but has not undergone programmatic, policy, or publication review, and is provided for information only. The document provides preliminary information that may change based on new information or analysis, and is not intended for publication or wide distribution; it is a lower level contractor document that may or may not directly contribute to a published DOE report. Although this document has undergone technical reviews at the contractor organization, it has not undergone a DOE policy review. Therefore, the views and opinions of authors expressed do not necessarily state or reflect those of the DOE. However, in the interest of the rapid transfer of information, we are providing this document for your information, per your request.

MOL.19991217.0512

OFFICE OF CIVILIAN RADIOACTIVE WASTE MANAGEMENT SYSTEM DESCRIPTION DOCUMENT VOLUME I COVER SHEET			1. QA: QA Page: 1 of 60
2. SDD Title Uncanistered Spent Nuclear Fuel Disposal Container System Description Document			
3. Document Identifier (Including Rev. No. and Change No., if applicable) SDD-UDC-SE-000001 REV 00			
	Printed Name	Signature	Date
4. System Engineer	N. E. Pettit	[signature]	11/30/99
5. Checker	G. C. Johnson ~~J. M. Hartsell~~ GCJ 11/30/99	[signature]	11/30/99
6. Responsible Manager	R. M. Dulin	[signature]	11/30/99

7. Remarks:

The following TBD/TBV are contained in this document:

TBD-232; ~~TBD-233;~~ NEP 11/30/99 TBD-234; TBD-235; TBD-241; TBD-276; TBD-3755; TBD-3764; TBD-3765

TBV-094; TBV-096; TBV-238; TBV-241; TBV-245; TBV-250; TBV-273; TBV-455; TBV-3865

Revision 00 of this System Description Document consists of Volume I, only. Revision 00 of this document supersedes BBA000000-01717-1705-00004 REV 01. Volume II will be presented in a future revision.

Enclosure 4

OFFICE OF CIVILIAN RADIOACTIVE WASTE MANAGEMENT SYSTEM DESCRIPTION DOCUMENT REVISION HISTORY	Page: 2 of 60
1. SDD Title Uncanistered Spent Nuclear Fuel Disposal Container System Description Document	
2. Document Identifier (Including Rev. No. and Change No., if applicable) SDD-UDC-SE-000001 REV 00	
3. Revision	4. Description of Revision
00	Initial Issue This document is a complete revision of the superseded BBA000000-01717-1705-00004 REV 01. The document incorporates changes to the "Monitored Geologic Repository Requirements Document," including switching traceability to the "Revised Interim Guidance Pending Issuance of New U.S. Nuclear Regulatory Commission (NRC) Regulations (Revision 01, July 22, 1999), for Yucca Mountain, Nevada." This revision incorporates the "Classification of the MGR Uncanistered Spent Nuclear Fuel Disposal Container System." This revision incorporates the revision to the "MGR Compliance Program Guidance Package for the Uncanistered Spent Nuclear Fuel (SNF) Disposal Container." Changes have been included for the system to comply with management direction put into effect via the "Monitored Geologic Repository Project Description Document."

CONTENTS

TABLES

SUMMARY

The Uncanistered Spent Nuclear Fuel (SNF) Disposal Container System supports the confinement and isolation of waste within the Engineered Barrier System of the Monitored Geologic Repository (MGR). Disposal containers are loaded with intact uncanistered assemblies and/or individually canistered SNF assemblies and sealed in the surface waste handling facilities, transferred to the underground through the access drifts, and emplaced in emplacement drifts. The Uncanistered SNF Disposal Container provides long-term confinement of the commercial SNF placed inside, and withstands the loading, transfer, emplacement, and retrieval loads and environments.

The Uncanistered SNF Disposal Container System provides containment of waste for a designated period of time, and limits radionuclide release. The disposal container maintains the waste in a designated configuration, withstands maximum handling and rockfall loads, limits the individual SNF assembly temperatures after emplacement, limits the introduction of moderator into the disposal container during the criticality control period, resists corrosion in the expected handling and repository environments, and provides containment of waste in the event of an accident.

Multiple boiling water reactor (BWR) and pressurized water reactor (PWR) disposal container designs are needed to accommodate the expected range of spent fuel assemblies and provide long-term confinement of the commercial SNF. The disposal container will include outer and inner cylinder walls and outer and inner cylinder lids, and an internal metallic basket structure. Exterior labels will provide a means by which to identify the disposal container and its contents.

The two metal cylinders, in combination with the cladding, Emplacement Drift System, drip shield, and natural barrier, will support the design philosophy of defense-in-depth. The use of materials with different properties prevents a single mode failure from breaching the waste package. The inner cylinder and inner cylinder lids will be constructed of stainless steel and the outer cylinder and outer cylinder lid will be made of high-nickel alloy. The basket will assist criticality control, provide structural support, and improve heat transfer.

The Uncanistered SNF Disposal Container System interfaces with the emplacement drift environment and the internal waste by transferring heat from the SNF to the external environment and by protecting the SNF assemblies and their contents from damage/degradation by the external environment. The system also interfaces with the SNF by limiting access of moderator and oxidizing agents to the SNF. A loaded and sealed disposal container (waste package) interfaces with the Emplacement Drift System's emplacement drift waste package supports. The disposal container interfaces with the Assembly Transfer System, Waste Emplacement/Retrieval System, Disposal Container Handling System, and Waste Package Remediation System during loading, handling, transfer, emplacement, and retrieval of the disposal container/waste package.

QUALITY ASSURANCE

The quality assurance program applies to the development of this document. The "SDD Development/Maintenance (Q SDDs) (WP# 16012126M5)" activity evaluation has determined the development of this document to be subject to DOE/RW-0333P, "Quality Assurance Requirements and Description" requirements. This document was developed in accordance with AP-3.11Q, "Technical Reports."

1. FUNCTIONS AND DESIGN CRITERIA

The functions and design criteria for the Uncanistered Spent Nuclear Fuel Disposal Container System are identified in the following sections. Throughout this document, the term "disposal container" is used to indicate the Uncanistered Spent Nuclear Fuel Disposal Container System and the suite of individual disposal containers designed for uncanistered SNF. The system architecture and classification are provided in Appendix B.

The term "disposal container" means the container cylinders and any integral structures (spacers, lifting features, absorbent materials, etc.). The term "waste package" means a disposal container that is loaded with a waste form, sealed by the designed methods, and is tested and accepted.

To address the term "breach" in a quantified manner, threshold limits for failure per American Society of Mechanical Engineers (ASME) code are to be used. Throughout this document when the term "breach" is referred to in a function or criterion, the following apply: During normal handling operations, breach has occurred, analytically, when Subsection NB 3200 limits of stress intensity for the stress categories are exceeded. For accident (design basis event) conditions, breach has occurred, analytically, when 0.9 of the ultimate tensile strength is exceeded.

1.1 SYSTEM FUNCTIONS

1.1.1 The disposal container/waste package contains intact uncanistered and individually canistered SNF within its boundary until it is breached.

1.1.2 The waste package restricts the transport of radionuclides to the outside of the disposal container's boundary after it is breached.

1.1.3 The disposal container/waste package provides criticality control during and after it is loaded with waste.

1.1.4 The waste package accommodates the thermal loading strategy for the repository.

1.1.5 The disposal container/waste package provides identification of individual disposal containers and their contents.

1.1.6 The disposal container/waste package provides safety for personnel, equipment, and the environment.

1.1.7 The disposal container/waste package prevents adverse reactions involving the waste form.

1.1.8 The disposal container/waste package withstands loading, handling, sealing, transfer, emplacement, and retrieval loads.

1.1.9 The waste package withstands the emplacement drift environment for the period of interest.

1.1.10 The disposal container/waste package provides conditions needed to maintain the physical and chemical stability of the waste form.

1.1.11 The waste package minimizes mobilization of radionuclides.

1.1.12 The waste package allows heat transfer between the waste form and the environment external to the waste package.

1.1.13 The disposal container/waste package accommodates handling, sealing, loading, emplacement, and retrieval operations.

1.1.14 The disposal container/waste package outer surface facilitates decontamination.

1.2 SYSTEM DESIGN CRITERIA

This section presents the design criteria for the system. Each criterion in this section has a corresponding Criterion Basis Statement in Appendix A of Volume I that describes the need for the criterion as well as a basis for the performance parameters imposed by the criterion. Each criterion in this section also contains bracketed traces indicating traceability, as applicable, to the functions (F) in Section 1.1, the "Monitored Geologic Repository Requirements Document" (MGR RD) (as modified by input transmittal "Preliminary Draft Requirements from the Monitored Geologic Repository Requirements Document" [TBV-3855]), and "Revised Interim Guidance Pending Issuance of New U.S. Nuclear Regulatory Commission (NRC) Regulations (Revision 01, July 22, 1999), for Yucca Mountain, Nevada." In anticipation of the interim guidance being promulgated as a Code of Federal Regulations, it will be referred to as "10 CFR 63" in this system description document. For the applicable version of the codes, standards, and regulatory documents imposed on the design of this system, refer to Appendix E.

1.2.1 System Performance Criteria

1.2.1.1 The disposal container shall accommodate intact fuel assemblies from the assembly classes identified in Tables I-1 and I-2 (TBV-455). Tables I-1 and I-2 also identify parameters (size, weight, and inventory) that may be used in design (TBV-455). The reactors that correspond to the multiple reactor classes (such as GE BWR 2,3 assembly class) are listed in Appendix F for reference (TBV-455). TBV-455 is placed on this criterion because that TBV is placed on the reference documents, in accordance with "Deficiency Report VAMO-98-D-132."

Table I-1. BWR Fuel

Assembly Class	Length* (in.)	Channeled Width* (in.)	Unchanneled Width** (in.)	Assembly Weight (lb)	Non-Fuel Component (NFC) Weight (lb)	Total Weight (lb)	Number of Stored Assemblies 12/31/94	Projected Number of Assemblies 12/31/2040
Big Rock Point	84.8	6.81	6.52	465	101	566	421	0.6×10^3
Humboldt Bay	96	4.8	4.67	276	23	299	390	0.4×10^3
LaCrosse	103.5	5.91	5.62	386	69	455	333***	0.3×10^3 (333***)
Dresden 1	135.7	4.57	4.28	328	30	358	892	0.9×10^3
GE BWR 2, 3	173	5.61	5.36	619	80	699	18,813	3.5×10^4
GE BWR 4-6	177.8	5.61	5.36	588	80	668	39,295	1.3×10^5

* Dimensions are post-irradiation. Widths include fuel channels, but make no allowance for channel spacer buttons and attachment clips.
** Dimensions are post-irradiation.
*** Stainless steel clad assemblies.
Note: For definition of acronyms, symbols and units, see Appendix C.

Table I-2. PWR Fuel

Assembly Class	Length w/o NFC* (in.)	Length w/ NFC* (in.)	Width* (in.)	Assembly Weight (lb)	NFC Weight (lb)	Total Weight (lb)	Number of Stored Assemblies 12/31/94	Projected Number of Assemblies 12/31/2040
Yankee Rowe	112.9	112.9	7.61	797	N/A	797	533 (76**)	0.7×10^3 (76**)
San Onofre 1	138.4	139.9	7.76	1,247	107	1,354	665 (665**)	1.0×10^3 (822**)
Haddam Neck	138.4	139.9	8.5	1,255	166	1,421	892 (888**)	1.5×10^3 (945**)
Indian Point 1	139.1	139.1	6.27	437	N/A	437	160 (160**)	0.2×10^3 (160**)
Fort Calhoun	147.7	158.5	8.12	1,220	67	1,287	570	1.1×10^3
Palisades	148.9	148.9	8.31	1,360	N/A	1,360	793	1.5×10^3
CE 14x14	158.8	169.6	8.11	1,270	77	1,347	4565	9.8×10^3
St. Lucie	159.7	170.6	8.13	1,300	66	1,366	544	1.9×10^3
WE 15x15	161.4	166.9	8.42	1,472	165	1,637	7,490	1.5×10^4
WE 14x14	161.4	166.3	7.76	1,302	130	1,432	4,093	7.8×10^3
WE 17x17	161.4	168.8	8.42	1,482	180	1,662	15,295	5.9×10^4
B&W 17x17	167.4	173.5	8.54	1,505	149	1,654	4	3.1×10^3
B&W 15x15	167.4	173.5	8.54	1,515	165	1,680	5,435	1.0×10^4
CE 16x16	178.6	190.8	8.14	1,430	72	1,502	2,340	8.1×10^3
CE System 80	180	194.8	8.16	1,430	N/A	1,430	1,132	8.1×10^3
South Texas	201.1	201.1	8.4	1,720	200	1,920	424	3.0×10^3

* Dimensions are post-irradiation.
** Number of stainless steel clad assemblies. Remainder are zircaloy alloy clad.
Note: For definition of acronyms, symbols and units, see Appendix C.

[F 1.1.1][MGR RD 3.2.B]

1.2.1.2 The disposal container shall accommodate sealed, disposable, single-element SNF canisters (canisters containing non-intact fuel that will not be opened and repackaged in the surface facilities) that are capable of fitting without forcing (when lowered vertically) into a three-dimensional square rectangular cavity with a cross-sectional width of 9.00 in. (22.9 cm) (TBV-238) by 9.00 in. (22.9 cm) (TBV-238) and a length of 201.1 in. (510.8 cm) (TBV-238). The canister will not exceed a weight of (TBD-3765).

[F 1.1.1][MGR RD 3.2.B]

1.2.1.3 The disposal container/waste package shall be designed, in conjunction with the Emplacement Drift System and the natural barrier, such that the expected annual dose to the average member of the critical group shall not exceed 25 mrem total effective dose equivalent at any time during the first 10,000 years after permanent closure, as a result of radioactive materials released from the geologic repository.

[F 1.1.1, 1.1.9][MGR RD 3.1.C][10 CFR 63.113(b)]

1.2.1.4 The disposal container shall consist of two cylinders; an inner cylinder that is stainless steel (alloy 316) with a nominal thickness of 5 cm, and an outer cylinder that is alloy 22 material with a nominal thickness of 2 cm.

[F 1.1.1, 1.1.2][MGR RD 3.1.C][10 CFR 63.113(a)]

1.2.1.5 The waste package shall be designed to achieve a reliability of (TBD-3755) percent during the first 10,000 years after emplacement in an emplacement drift.

[F 1.1.1][MGR RD 3.1.C][10 CFR 63.113(b), 63.114(d), 63.114(e)]

1.2.1.6 The waste package shall maintain SNF zircaloy cladding temperature below 350 degrees C (662 degrees F) (TBV-241) under normal conditions, and below 570 degrees C (1,058 degrees F) (TBV-245) for short-term exposure to fire, as specified by Criterion 1.2.2.1.11.

[F 1.1.4, 1.1.11, 1.1.12][MGR RD 3.1.C][10 CFR 63.111(a)(2), 63.111(b)(2), 63.112(e)(8), 63.113(b)]

1.2.1.7 The disposal container/waste package shall prevent the breach of single-element canisters and any new breaches to the cladding of uncanistered fuel assemblies during normal handling operations.

[F 1.1.10][MGR RD 3.1.C, 3.1.G][10 CFR 63.111(a)(2), 63.113(b)]

1.2.1.8 The disposal container/waste package shall be designed to support/allow retrieval up to 300 years after the start of emplacement operations.

[F 1.1.8, 1.1.13][MGR RD 3.1.C, 3.2.H][10 CFR 63.111(e)(1)]

1.2.1.9 Combined concentrations of O_2, H_2, H_2O, CO_2, and CO within a waste package shall not exceed 0.25 percent of internal volume (TBV-094), prior to waste package breach.

[F 1.1.7, 1.1.10, 1.1.11][MGR RD 3.1.C][10 CFR 63.111(e)(1), 63.113(b)]

1.2.1.10 The disposal container/waste package, excluding the labels, shall have an external surface finish Roughness Average of 250 μin (6.36 μm) or less.

[F 1.1.14][MGR RD 3.1.C][10 CFR 63.112(e)(2)]

1.2.1.11 The disposal container/waste package shall have all external surfaces (surfaces exposed to the external environment after closing and sealing a disposal container) accessible for visual inspection and decontamination (e.g., no blind holes).

[F 1.1.14][MGR RD 3.1.C][10 CFR 63.112(e)(2)]

1.2.1.12 The disposal container/waste package shall have a label (or other means of identification) with a unique waste package identifier.

[F 1.1.5][MGR RD 3.1.B, 3.1.C, 3.3.K][10 CFR 63.112(e)(2), 63.78]

1.2.1.13 All labels (or other means of identification) applied to the waste package shall not impair the integrity of the waste package.

[F 1.1.5][MGR RD 3.1.C][10 CFR 63.113(b)]

1.2.1.14 All information contained on all labels (or other means of identification) applied to the disposal container/waste package shall be legible or read by remote means until permanent closure of the repository.

[F 1.1.5][MGR RD 3.1.C, 3.1.D, 3.3.K][10 CFR 63.112(e)(2), 63.78]

1.2.1.15 The disposal container shall accommodate the use of filler material (TBV-250), such as iron shot or depleted uranium, added to the interior of the disposal container.

[MGR RD 3.3.N]

1.2.1.16 Lifting features of the disposal container/waste package shall be designed for three times the maximum weight of the loaded and sealed disposal container without generating a combined shear stress or maximum tensile stress in excess of the corresponding minimum tensile yield strength of the materials of construction.

[F 1.1.8, 1.1.13][MGR RD 3.1.C, 3.1.G][10 CFR 63.111(a)(2), 63.111(b)(2), 63.112(e)(2), 63.112(e)(8)]

1.2.1.17 Lifting features of the disposal container/waste package shall be designed for five times the weight of the waste package without exceeding the ultimate tensile strength of the materials.

[F 1.1.8, 1.1.13][MGR RD 3.1.C, 3.1.G][10 CFR 63.111(a)(2), 63.111(b)(2), 63.112(e)(2), 63.112(e)(8)]

1.2.1.18 The waste package shall withstand transfer, emplacement, and retrieval operations without breaching.

[F 1.1.1, 1.1.8][MGR RD 3.1.C, 3.1.G][10 CFR 63.111(a)(2), 63.111(b)(2), 63.112(e)(8)]

1.2.1.19 The disposal container/waste package shall be constructed of non-combustible and heat resistant materials only.

[F 1.1.7, 1.1.10][MGR RD 3.1.C][10 CFR 63.111(a)(2), 63.111(b)(2), 63.112(e)(8)]

1.2.1.20 Disposal container/waste package materials shall exclude the use of explosive or pyrophoric materials.

[F 1.1.2, 1.1.11][MGR RD 3.1.C][10 CFR 63.111(a)(2), 63.111(b)(2), 63.112(e)(8)]

1.2.1.21 Disposal container/waste package materials shall exclude the use of free liquids.

[F 1.1.7, 1.1.10]

1.2.2 Safety Criteria

1.2.2.1 Nuclear Safety Criteria

1.2.2.1.1 During the preclosure period, the waste package shall be designed to withstand (while in a horizontal orientation) a 13 MT (28,665 lb) (TBV-245) rock (spherical geometry assumed) falling 3.1 m (10.2 ft) (TBV-245) onto the side of the waste package without breaching. (TBV-245)

[F 1.1.1, 1.1.2, 1.1.6][MGR RD 3.1.C, 3.1.G][10 CFR 63.111(a)(2), 63.111(b)(2), 63.112(e)(8)]

1.2.2.1.2 During the preclosure period, the disposal container/waste package shall be designed to withstand (while in a vertical orientation) a 2.3 MT (5,100 lb) (TBV-245) spherical object falling 2 m (6.6 ft) (TBV-245) onto the end of the disposal container without breaching. (TBV-245)

[F 1.1.1, 1.1.2, 1.1.6][MGR RD 3.1.C, 3.1.G][10 CFR 63.111(a)(2), 63.111(b)(2), 63.112(e)(8)]

1.2.2.1.3 During the preclosure period, the disposal container/waste package, shall be designed to withstand (while in a vertical orientation) a drop from a height of 2 m (6.6 ft) (TBV-245) onto a flat, unyielding surface without breaching. (TBV-245)

[F 1.1.1, 1.1.2, 1.1.6][MGR RD 3.1.C, 3.1.G][10 CFR 63.111(a)(2), 63.111(b)(2), 63.112(e)(8)]

1.2.2.1.4 During the preclosure period, the disposal container/waste package, shall be designed to withstand (while in a horizontal orientation) a drop from a height of 2.4 m (7.9 ft) (TBV-245) onto a flat, unyielding surface without breaching. (TBV-245)

[F 1.1.1, 1.1.2, 1.1.6][MGR RD 3.1.C, 3.1.G][10 CFR 63.111(a)(2), 63.111(b)(2), 63.112(e)(8)]

1.2.2.1.5 During the preclosure period, the waste package shall be designed to withstand (while in a horizontal orientation) the greater stress resulting from a drop of 1.9 m (6.2 ft) (TBV-245) onto a steel support in an emplacement drift, or a drop of 2.4 m (7.9 ft) (TBV-245) onto a concrete pier, without breaching by puncture. (TBV-245)

[F 1.1.1, 1.1.2, 1.1.6][MGR RD 3.1.C, 3.1.G][10 CFR 63.111(a)(2), 63.111(b)(2), 63.112(e)(8)]

1.2.2.1.6 During the preclosure period, the waste package shall be designed to withstand a tip over from a vertical position with slap down onto a flat, unyielding surface without breaching. (TBV-245)

[F 1.1.1, 1.1.2, 1.1.6][MGR RD 3.1.C, 3.1.G][10 CFR 63.111(a)(2), 63.111(b)(2), 63.112(e)(8)]

1.2.2.1.7 The waste package shall be designed to withstand a Design Basis Earthquake of Frequency Category 2. Both vibratory ground motion and fault displacement of the Frequency Category 2 Design Basis Earthquake must be considered, taking credit as appropriate for interfacing systems that alter or mitigate the effect of the design basis earthquake on the waste package. The surface environment floor response spectra for the system are (TBD-241). The subsurface parameters of the design basis earthquake are defined in Tables I-3 through I-6. (TBD-241, TBV-273)

Table I-3. Parameters for the Vibratory Ground Motion Design Basis Earthquake–Subsurface Environment–Repository Interface (Underground) Design Spectral Accelerations for Design Earthquake Scaled to 5–10 Hz Frequency Range

Response Frequency (Hz)	Horizontal Motion Frequency Category 2 (10,000 Year Recurrence)	Vertical Motion Frequency Category 2 (10,000 Year Recurrence)
0.3	0.0720 g (TBV)	0.0378 g (TBV)
0.5	0.125 g (TBV)	0.0688 g (TBV)
1	0.206 g (TBV)	0.130 g (TBV)
2	0.458 g (TBV)	0.180 g (TBV)
5	0.717 g (TBV)	0.435 g (TBV)
10	0.765 g (TBV)	0.620 g (TBV)
20	0.681 g (TBV)	0.613 g (TBV)
100	0.391 g (TBV)	0.288 g (TBV)

Table I-4. Parameters for the Vibratory Ground Motion Design Basis Earthquake–Subsurface Environment–Repository Interface (Underground) Design Spectral Accelerations for Design Earthquake Scaled to 1–2 Hz Frequency Range

Response Frequency (Hz)	Horizontal Motion Frequency Category 2 (10,000 Year Recurrence)	Vertical Motion Frequency Category 2 (10,000 Year Recurrence)
0.3	0.186 g (TBV)	0.101 g (TBV)
0.5	0.252 g (TBV)	0.149 g (TBV)
1	0.286 g (TBV)	0.206 g (TBV)
2	0.465 g (TBV)	0.212 g (TBV)
5	0.471 g (TBV)	0.309 g (TBV)
10	0.374 g (TBV)	0.295 g (TBV)
20	0.302 g (TBV)	0.244 g (TBV)
100	0.231 g (TBV)	0.156 g (TBV)

Table I-5. Parameters for the Vibratory Ground Motion Design Basis Earthquake–Subsurface Environment–Repository Interface (Underground) Design Peak Velocity (cm/sec) for Design Earthquake Scaled to 5–10 Hz and 1–2 Hz Frequency Ranges

Design Earthquake Frequency (Hz)	Horizontal Motion	Vertical Motion
	Frequency Category 2 (10,000 Year Recurrence)	Frequency Category 2 (10,000 Year Recurrence)
5 – 10	33.63 (TBV)	17.10 (TBV)
1 – 2	41.84 (TBV)	22.38 (TBV)

Table I-6. Parameters for the Ground Displacement Design Basis Earthquake–To Be Applied as Appropriate to Both Surface and Subsurface Facilities

Ground Displacement Design Basis Earthquake	Fault Displacement		Comment
	Surface	Subsurface	
Frequency Category 2 (100,000 Year Recurrence)	(TBD) cm	Less than 1 cm (TBV)	Considered insignificant with respect to repository design except for block-bounding faults: Bow Ridge 12 cm (TBV) Solitario Canyon 30 cm (TBV)

Note: For definition of acronyms, symbols and units, see Appendix C.

[F 1.1.1, 1.1.2, 1.1.6][MGR RD 3.1.C, 3.1.G][10 CFR 63.111(a)(2), 63.111(b)(2), 63.112(e)(8)]

1.2.2.1.8 During the preclosure period, the waste package shall be designed to withstand the impact of a 0.5 kg (1.1 lb) (TBV-245) missile (modeled as a 1 cm diameter, 5 cm long valve stem) travelling at 5.7 m per second (18.7 ft/sec) (TBV-245) without breaching. (TBV-245)

[F 1.1.1, 1.1.2, 1.1.6][MGR RD 3.1.C, 3.1.G][10 CFR 63.111(a)(2), 63.111(b)(2), 63.112(e)(8)]

1.2.2.1.9 During the preclosure period, the waste package shall be designed to withstand, without breaching, the maximum impact resulting from a transporter runaway, derailment, and impact at a speed of 63 km/hr (39 mi/hr) (TBV-245), taking credit as appropriate for interfacing systems that prevent or mitigate the impact on the waste package. (TBV-245)

[F 1.1.1, 1.1.6][MGR RD 3.1.C, 3.1.G][10 CFR 63.111(a)(2), 63.111(b)(2), 63.112(e)(8)]

1.2.2.1.10 During the preclosure period, the waste package shall be designed to withstand a maximum internal pressure of 1.01 MPa (146 psia) (TBV-245) without breaching. (TBV-245)

[F 1.1.1, 1.1.6][MGR RD 3.1.C, 3.1.G][10 CFR 63.111(a)(2), 63.111(b)(2), 63.112(e)(8)]

1.2.2.1.11 The waste package/disposal container shall be designed to withstand the hypothetical fire criteria defined in 10 CFR 71 ("Packaging and Transportation of Radioactive Materials"), Section 73(c)(4). (TBV-245)

[F 1.1.1, 1.1.2, 1.1.6][MGR RD 3.1.C, 3.1.G][10 CFR 63.111(a)(2), 63.111(b)(2), 63.112(e)(8)]

1.2.2.1.12 During the preclosure period, the disposal container/waste package shall be designed such that nuclear criticality shall not be possible unless at least two unlikely, independent, and concurrent or sequential changes have occurred in the conditions essential to nuclear criticality safety. The system must be designed for criticality safety assuming occurrence of design basis events, including those with the potential to flood (TBD-235) the disposal container prior to its sealing and the misload of assemblies into a disposal container (TBD-235). The calculated effective multiplication factor (k_{eff}) must be sufficiently below unity to show at least a 5 percent margin after allowance for the bias in the method of calculation and the uncertainty in the experiments used to validate the method of calculation. (TBV-245)

[F 1.1.3][MGR RD 3.1.C, 3.1.G][10 CFR 63.111(a)(2), 63.111(b)(2), 63.112(e)(6), 63.112(e)(8)]

1.2.2.1.13 During the postclosure period, criticality events due to fissionable material emplaced in the waste package shall not increase the total radionuclide inventory of the waste package by more than 1 percent. The percentage of radionuclide inventory increase for the waste package shall be measured by the sum of the products of probability of criticality occurrence (for a single uncanistered SNF waste package, as a function of time), multiplied by the radionuclide inventory increment (measured in curies) due to that criticality, divided by the radionuclide inventory of a single uncanistered SNF waste package, with the sum taken over time and any other parameters which characterize the occurrence of criticality. Both the radionuclide inventory and the increment due to criticality shall be evaluated at 1,000 years following the criticality shutdown. (TBV-096)

[F 1.1.3][MGR RD 3.1.C][10 CFR 63.113(b)]

1.2.2.2 Non-nuclear Safety Criteria

Non-nuclear safety criteria for this system will be identified in a later revision, as necessary.

1.2.3 System Environment Criteria

1.2.3.1 The waste package shall meet all performance requirements during and after exposure to the emplacement drift external environments identified in Table I-7 (TBD-234) and the induced/handling environments identified in Table I-8 (TBD-276).

Table I-7. Emplacement Drift External Environment

Environment	Range	Duration/Frequency of Occurrence
Microbe Influx	0–10^{14} microbes/yr/m of drift	10,000 yr
pH	8.2-10.2	10,000 yr
Colloid Concentration	$8x10^{-6}$ – $6x10^{-5}$ mg/ml	10,000 yr
Temperature	TBD-234	TBD-234
Humidity	TBD-234	TBD-234
Radiation	TBD-234	TBD-234
TBD-234	TBD-234	TBD-234

Table I-8. Induced/Handling External Environment

Environment	Range	Duration/Frequency of Occurrence
Vibration	TBD-276	TBD-276
Shock	TBD-276	TBD-276
Acceleration	TBD-276	TBD-276
TBD-276	TBD-276	TBD-276

Note: For definition of acronyms, symbols and units, see Appendix C.

[F 1.1.9][MGR RD 3.1.C, 3.4.2.C][10 CFR 63.113(b)]

1.2.4 System Interfacing Criteria

1.2.4.1 The system shall be designed in accordance with the interface agreements defined in "Interface Control Document for the Waste Packages/Disposal Containers and the Surface Repository Facilities and Systems for Mechanical, Envelope and Functional Interfaces."

[F 1.1.13][MGR RD 3.2.C]

1.2.4.2 The system shall be designed in accordance with the interface agreements defined in "Interface Control Document for Waste Packages and the Mined Geologic Disposal System Repository Subsurface Facilities and Systems for Mechanical and Envelope Interfaces."

[F 1.1.13][MGR RD 3.2.C]

1.2.4.3 Waste package design shall reduce the dose rate at all external surfaces of a waste package to (TBD-3764) rem/hr or less. This criterion identifies a disposal container interface with the Waste Emplacement/Retrieval System, Disposal Container Handling System, and Performance Confirmation Emplacement Drift Monitoring System.

[F 1.1.6, 1.1.13][MGR RD 3.1.B, 3.1.G]

1.2.4.4 The waste package shall be designed to have a maximum thermal output of 11.8 kW.

[F 1.1.4]

1.2.4.5 The quantity of waste forms disposed of in this suite of disposal containers, in combination with the non-fuel components and canistered SNF disposal containers, shall not exceed 63,000 metric tons uranium (MTU) for the first repository. This criterion identifies the primary uncanistered SNF disposal container interface with the canistered SNF and non-fuel components disposal containers.

[F 1.1.1][MGR RD 3.1.A, 3.2.A]

1.2.4.6 The disposal container shall be designed to be loaded and sealed in a vertical orientation. This criterion identifies the primary disposal container interface with the Assembly Transfer System and the Disposal Container Handling System.

[F 1.1.13]

1.2.4.7 The disposal container/waste package shall be designed to be handled in both horizontal and vertical orientations. This criterion identifies the primary disposal container interface with the Disposal Container Handling System and the Waste Emplacement/Retrieval System.

[F 1.1.13]

1.2.4.8 The disposal container/waste package shall be designed to support required welding times. This criterion identifies a primary disposal container interface with the Disposal Container Handling System.

[F 1.1.13]

1.2.5 Operational Criteria

Operational criteria for this system will be identified in a later revision, if necessary.

1.2.6 Codes and Standards Criteria

1.2.6.1 The disposal container shall be designed in accordance with applicable sections of "1995 ASME Boiler and Pressure Vessel Code" (Section III, Division 1, Subsection NG–1995).

[MGR RD 3.3.A]

1.2.6.2 The disposal container shall be designed in accordance with applicable sections of "1995 ASME Boiler and Pressure Vessel Code" (Section III, Division 1, Subsection NB-1995).

[MGR RD 3.3.A]

1.2.6.3 The disposal container shall be designed in accordance with applicable sections of "Nuclear Criticality Control of Special Actinide Elements" (ANSI/ANS-8.15-1981).

[MGR RD 3.3.A]

1.2.6.4 The disposal container shall be designed in accordance with applicable sections of "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors" (ANSI/ANS-8.1-1998).

[MGR RD 3.3.A]

1.2.6.5 The disposal container shall be designed in accordance with applicable sections of "Criteria for Nuclear Safety Controls in Operations with Shielding and Confinement" (ANSI/ANS-8.10-1983).

[MGR RD 3.3.A]

1.2.6.6 The disposal container shall be designed in accordance with applicable sections of "Criticality Safety Criteria for the Handling, Storage, and Transportation of LWR Fuel Outside Reactors" (ANSI/ANS-8.17-1984).

[MGR RD 3.3.A]

1.3 SUBSYSTEM DESIGN CRITERIA

Subsystem design criteria for this system will be identified in a later revision, if necessary.

1.4 CONFORMANCE VERIFICATION

This section outlines the methods to be used to verify the conformance of the system with its design criteria.

1.4.1 The methods of conformance verification to be used are:

Analysis. Analysis is the process of accumulating results and conclusions intended to verify that a requirement has been satisfied. Analytical verification of compliance may include compilation and interpretation of results of tests, demonstrations, and examinations of lower-level components of the system. Analysis may also include logical arguments, modeling, calculations, tradeoff studies, reports (design and/or tradeoff), and other relevant information to verify compliance with a requirement, when physical testing of a system is impracticable.

Examination. Examination is the process of conducting careful observation and inspection, without use of special laboratory appliances and procedures, to verify compliance with specified requirements. Examination is a relatively direct method, involving, at most, simple physical manipulation or measurement. It is generally non-destructive and does not necessarily involve operation of the system being evaluated.

Demonstration. Demonstration is the qualitative process of displaying or operating a system or item in or near its operational environment to verify compliance with requirements. It differs from testing in that it is generally a qualitative and direct determination of the performance of a function and is performed without special instrumentation or other special equipment.

Test. Test is the quantitative process whereby data are collected, under controlled conditions, to document the performance of a product with respect to a standard. Manipulation and analysis of data derived from testing is an integral part of the method. Special instrumentation and scientific procedures are commonly employed. A test may be conducted in a laboratory or in the field (in situ).

1.4.2 Table I-9 correlates the criteria with the method to be used to verify compliance with the criteria. In the following table, items marked "N/A" (not applicable) have no verification required. These items are titles or contain explanatory materials. The other columns "Analysis," "Demo," "Exam," and "Test" refer to the verification methods identified in Section 1.4.1.

Table I-9. Conformance Verification

Criterion		Verification Method Code				
Number	Title	N/A	Analysis	Exam	Demo	Test
1.2	SYSTEM DESIGN CRITERIA	X				
1.2.1	System Performance Criteria	X				
1.2.1.1			X		X	
1.2.1.2			X		X	
1.2.1.3			X			
1.2.1.4			X	X		
1.2.1.5			X			
1.2.1.6			X			
1.2.1.7			X			
1.2.1.8			X			
1.2.1.9			X			
1.2.1.10				X		
1.2.1.11			X	X		
1.2.1.12				X		
1.2.1.13			X	X		
1.2.1.14			X	X		
1.2.1.15			X			
1.2.1.16			X			
1.2.1.17			X			
1.2.1.18			X			
1.2.1.19			X			
1.2.1.20			X			
1.2.1.21			X			
1.2.2	Safety Criteria	X				
1.2.2.1	Nuclear Safety Criteria	X				
1.2.2.1.1			X		X	
1.2.2.1.2			X		X	
1.2.2.1.3			X		X	
1.2.2.1.4			X		X	
1.2.2.1.5			X		X	
1.2.2.1.6			X		X	
1.2.2.1.7			X			
1.2.2.1.8			X		X	
1.2.2.1.9			X		X	
1.2.2.1.10			X		X	
1.2.2.1.11			X			
1.2.2.1.12			X			
1.2.2.1.13			X			
1.2.2.2	Non-nuclear Safety Criteria	X				

Table I-9. Conformance Verification (Continued)

Criterion		Verification Method Code				
Number	**Title**	**N/A**	**Analysis**	**Exam**	**Demo**	**Test**
1.2.3	System Environment Criteria	X				
1.2.3.1			X			
1.2.4	System Interfacing Criteria	X				
1.2.4.1			X			
1.2.4.2			X			
1.2.4.3			X			
1.2.4.4			X			
1.2.4.5			X			
1.2.4.6			X	X		
1.2.4.7			X	X		
1.2.4.8			X		X	
1.2.5	Operational Criteria	X				
1.2.6	Codes and Standards Criteria	X				
1.2.6.1			X			
1.2.6.2			X			
1.2.6.3			X			
1.2.6.4			X			
1.2.6.5			X			
1.2.6.6			X			
1.3	SUBSYSTEM DESIGN CRITERIA	X				
1.4	CONFORMANCE VERIFICATION	X				

APPENDIX A CRITERION BASIS STATEMENTS

This section presents the criterion basis statements for criteria in Section 1.2 of Volume I. Descriptions of the traces to the "Monitored Geologic Repository Requirements Document" (MGR RD) (as modified by input transmittal "Preliminary Draft Requirements from the Monitored Geologic Repository Requirements Document") and "Revised Interim Guidance Pending Issuance of New U.S. Nuclear Regulatory Commission (NRC) Regulations (Revision 01, July 22, 1999), for Yucca Mountain, Nevada" are shown as applicable. In anticipation of the interim guidance being promulgated as a Code of Federal Regulations, it will be referred to as "10 CFR 63" in this system description document.

1.2.1.1 Criterion Basis Statement

I. Criterion Need Basis

This requirement, in conjunction with Criterion 1.2.1.2, establishes the types of intact uncanistered spent nuclear fuel (SNF) that the suite of uncanistered SNF disposal containers must contain. This requirement only identifies the assembly classes to be used in design. Additional information is provided (size, weight, and inventory) that may be used in design. Non-intact assemblies will also be disposed of in this suite of disposal containers, and are addressed in Criterion 1.2.1.2.

This criterion provides a lower level decomposition of MGR RD 3.2.B, by specifying the intact assembly classes that will make up a portion of the commercial SNF to be disposed at the Monitored Geologic Repository (MGR).

II. Criterion Performance Parameter Basis

The data for unchanneled width, number of stored assemblies, and projected number of assemblies are obtained from "Characteristics of Commercial SNF Assemblies to be Disposed of at the MGDS" (pp. 11-14). All other data are obtained from "Qualification of Spent Nuclear Fuel Assembly Characteristics for Use as a Design Basis" (pp. 14, 16, and 18). While the "Qualification of Spent Nuclear Fuel Assembly Characteristics for Use as a Design Basis" also provides a projection of SNF inventory, the projections from "Characteristics of Commercial SNF Assemblies to be Disposed of at the MGDS" are favored since they are presented as an order of magnitude number in that analysis. Orders of magnitude are less susceptible to constantly changing projections (as additional information is gathered from reactor discharge data), yet provide basic information on the magnitude of the fuel that will be available for disposal at the MGR.

1.2.1.2 Criterion Basis Statement

I. Criterion Need Basis

This requirement establishes the need for the accommodation of failed, and therefore individually canistered, SNF as discussed in Section 4 of the "Monitored Geologic Repository Draft Disposability Interface Specification."

This criterion provides a lower level decomposition of MGR RD 3.2.B, by specifying the individually canistered assemblies that will make up a portion of the commercial SNF to be disposed of at the MGR.

II. Criterion Performance Parameter Basis

This requirement is adapted from the "Monitored Geologic Repository Draft Disposability Interface Specification" disposability standards 2.2.20.1 and 2.2.21.1 that limit the dimensions of disposable single element canisters.

1.2.1.3 Criterion Basis Statement

I. Criterion Need Basis

This criterion is needed to define the overall level of performance of the repository to which this system will contribute. This criterion supports MGR RD 3.1.C and 10 CFR 63.113(b).

II. Criterion Performance Parameter Basis

The performance parameters are taken from 10 CFR 63.113(b).

1.2.1.4 Criterion Basis Statement

I. Criterion Need Basis

This criterion is required for the system to comply with management direction put into effect via the "Monitored Geologic Repository Project Description Document," which places constraints on the materials of construction and the nominal material thickness of each of two concentric cylinders that make up the disposal container. This criterion is also a consideration of 10 CFR 63.113(a), which requires the MGR include multiple barriers, including an engineered barrier system.

II. Criterion Performance Parameter Basis

The material and nominal thickness parameters are obtained from the "Monitored Geologic Repository Project Description Document" (Section 2.2.1.1.10).

1.2.1.5 Criterion Basis Statement

I. Criterion Need Basis

This criterion is a consideration of MGR RD 3.1.C and 10 CFR 63.113(b), and specifically addresses the performance assessment requirements of 10 CFR 63.63.114(d) and 10 CFR 63.114(e). This criterion is needed to ensure the rate of "failure" of the waste packages remains within acceptable limits. The term includes juvenile failure, operational failure, and failure as a result of other unexpected or design basis events.

The term "reliability" is defined as the probability that the waste package will not breach due to non-mechanistic failure at a rate of more than one waste package per 10,000 at any time during the first 10,000 years after emplacement.

II. Criterion Performance Parameter Basis

The 10,000-year parameter is derived from "Performance Assessment Input for Performance Allocation Study" (Section 5.1) (Input Transmittal RSO-PA-99262.T). The reliability parameter has not yet been determined.

1.2.1.6 Criterion Basis Statement

I. Criterion Need Basis

This criterion is a consideration of MGR RD 3.1.C and is needed to ensure the expected annual dose to the critical group during the first 10,000 years after permanent closure does not exceed 25 mrem, as required by 10 CFR 63.113(b). This criterion is intended to protect any undamaged cladding as a radionuclide containment barrier to assist in prevention of release of radionuclides (waste isolation).

This criterion is also needed in consideration of 10 CFR 63.111(a)(2), 10 CFR 63.111(b)(2), and 10 CFR 63.112(e)(8) to ensure the disposal container system performs its intended safety function assuming the occurrence of a design basis event.

This criterion is supported by Guidance Statements 6.7g2, 6.12g1, 6.12g3, and 6.13g1 contained in the "MGR Compliance Program Guidance Package for the Uncanistered Spent Nuclear Fuel (SNF) Disposal Container."

II. Criterion Performance Parameter Basis

The temperature value for normal conditions is taken from the "Thermal Loading Study for FY 1996" (Executive Summary, p. ix) which recommends the 350 degrees C criterion on cladding. The temperature value for exposure to fire is taken from "Nuclear Safety Criteria for the Disposal Container System Description Documents (SDDs)" (Attachment I, p. 7) (Input Transmittal RSO-RSO-99333.Ta).

1.2.1.7 Criterion Basis Statement

I. Criterion Need Basis

This requirement is a consideration of 10 CFR 63.111(a)(2) to protect against radiation exposure and release of radioactive materials during normal handling operations. Also, this criterion is provided to protect canisters and cladding as barriers to radionuclide release in consideration of 10 CFR 63.113(b).

This criterion is supported by Guidance Statements 6.12g1 and 6.13g1 contained in the "MGR Compliance Program Guidance Package for the Uncanistered Spent Nuclear Fuel (SNF) Disposal Container."

II. Criterion Performance Parameter Basis

N/A

1.2.1.8 Criterion Basis Statement

I. Criterion Need Basis

This requirement contributes to the ability to retrieve waste packages as required by 10 CFR 63.111(e)(1). This requirement dictates a time period in which the disposal containers must be capable of being moved after emplacement.

This criterion is supported by Guidance Statements 6.12g1 and 6.13g1 contained in the "MGR Compliance Program Guidance Package for the Uncanistered Spent Nuclear Fuel (SNF) Disposal Container."

II. Criterion Performance Parameter Basis

The 300-year time period is taken from MGR RD 3.2H, which requires the MGR support a deferral of closure for up to 300 years.

1.2.1.9 Criterion Basis Statement

I. Criterion Need Basis

This requirement is intended to preserve the waste form condition and configuration. This criterion is a consideration of the waste retrievability requirement of 10 CFR 63.111(e)(1). In addition, the possibility of a breach of the disposal container due to gas generation is reduced, in consideration of 10 CFR 63.113(b).

A non-oxidizing environment is desirable to prevent the conditions necessary for early corrosion of the waste form prior to waste package breach and provides defense-in-depth. It is therefore reasonable to be applied to the uncanistered SNF disposal container design.

II. Criterion Performance Parameter Basis

Criterion Analysis: Waste Package Non-oxidizing Internal Environment

Purpose

The purpose of this analysis is to establish the need for a non-oxidizing internal disposal container environment through bounding analysis and as a means to ensure defense-in-depth.

Assumptions

1. A high areal mass loading of 19.8 to 24.7 kgU/m^2 (80 to 100 MTU/acre) (see Note) will be selected for the final repository design based on MGR RD 3.2.I.

2. The decay heat curve is assumed to accurately represent the SNF currently in storage in the United States and it is not expected to change much ("Thermal Evaluation of Preliminary 21 PWR AUCF Design," p. 20).

3. The size of a significant proportion of the waste packages is large and contains 21 pressurized water reactor (PWR) assemblies ("Waste Package Size Study Report," pp. vii and 3-9).

Criterion Analysis

1. Some of the cladding covering the SNF pins will have perforations that could allow air in the disposal container to contact the UO_2. Although the exact percentage of failed pins is not important, the current conservative estimate is about 0.1 percent. Some estimates of the failed pins range, depending on when they were produced, from 0.006 to 0.07 percent ("The Technical Basis for the Classification of Failed Fuel in the Back-End

of the Fuel Cycle," pp. 4-1 through 4-4). Thus, a non-zero number of pins will have perforations upon disposal.

2. The SNF disposed of in the MGR has a temperature history upon disposal that produces high temperatures of the assemblies in the first 100 years following emplacement with subsequent decay in temperature. For a large (21 PWR) disposal container in a repository with high areal mass loading, assembly temperatures will peak above 250 degrees C to 300 degrees C ("Thermal Evaluation of Preliminary 21 PWR AUCF Design," p. 42, Figure 7.4-2; and "Thermal Loading Study for FY 1996," p. 3-61, Figure 3.4-10) in the first 100 years after emplacement. While the peak temperatures could change depending on fuel characteristics (burnup, age, and enrichment) and repository loading, most waste packages will have peak temperatures that are likely to be in the range mentioned above, or higher. Thus, the individual SNF pins will experience a period of high temperatures. Temperatures are not, however, expected to exceed 350 degrees C since that is a design criterion imposed on the waste package. For temperatures above 350 degrees C, not only will unzipping (discussed in Section 3 below) occur much faster in the presence of oxygen, but failure as a result of creep rupture is much more significant and would increase the number of failed pins.

3. If oxygen is present as part of the atmosphere that the individual pin is exposed to, and oxygen can reach the UO_2 through a perforation, there is the possibility to oxidize the UO_2 into U_3O_8. If this happens, clad unzipping can occur, which would ultimately expose more of the SNF pellets to water when the waste package is breached ("Thermal Loading Study for FY 1996," pp. 3-58 through 3-66). Also, U_3O_8 has a higher solubility than UO_2. The rate at which oxidation of UO_2 and subsequent clad unzipping occurs is a function of temperature. The time it takes for clad unzipping to occur is approximately 4.5 times slower at 250 degrees C than it is at 300 degrees C and it is approximately 22 times slower at 200 degrees C than it is at 300 degrees C ("Thermal Loading Study for FY 1996," pp. 3-58 through 3-66). If oxygen is present, has access to the UO_2 (through a perforation), and the temperatures are high, clad unzipping can occur. Temperatures will be high (above 200 degrees C) for a period of time and a nonzero fraction of pins will have perforations. Thus, the only thing that can be changed is to keep oxygen away from the SNF while it is at high temperatures. Keeping oxygen away from average fuel for at least 100 years minimizes the unzipping for that SNF. In the case of the design basis SNF, oxygen must be kept from the SNF for about 1,000 years to minimize unzipping.

4. Reactive gases (O_2, H_2, H_2O, CO_2, and CO) present will react with cladding and with the SNF. If sufficient concentrations exist, thinning of the cladding or splitting of the cladding can occur ("Evaluation of Cover Gas Impurities and Their Effects on the Dry Storage of LWR Spent Fuel," [PNL-6365, p. 1]). Even if the disposal canisters are evacuated and backfilled with an inert gas there will be small concentrations of these inert gases as a result of impurities in the backfill and outgassing from the SNF. An analysis done by the Pacific Northwest Laboratory, using examinations of a large number of SNF pins, found that if oxidizing gas concentrations are maintained less than or equal to 0.25

percent of volume, this is much too low a concentration to degrade the cladding (PNL-6365, p. 20). This is a precedent that was set for storage casks that were to be satisfactory for 40 years (when the highest SNF temperatures exist).

Conclusions

The disposal container should maintain a non-oxidizing environment within the waste package during the period before waste package breach.

The disposal container shall maintain the concentration of reactive gases (O_2, H_2, H_2O, CO_2, and CO) to 0.25 percent of volume.

The majority of the inputs including the inputs of the analysis used to arrive at the calculations of SNF temperature and extent of clad unzipping are unqualified. However, sufficient information is known to conclude that the SNF pins and cladding will experience high temperatures (significantly above 200 degrees C) after emplacement, that clad perforations will exist, and that if UO_2 is exposed to oxygen at high temperatures it will oxidize and result in clad unzipping. Thus, sufficient bounding analysis was used to establish that oxygen should be kept from the SNF during the period when temperatures are high.

Additionally, this conclusion is conservative in that it provides for defense-in-depth. The conclusion of the level of reactive gases is labeled as TBV because the analysis (PNL-6365) was not conducted under an approved quality program.

Note: While it is recognized that the parameters in this parameter basis statement are based on the Viability Assessment reference design, the resulting parameters are likely not incompatible with the Enhanced Design Alternative II design. Resolution of TBV-094 will indicate if the limits imposed upon reactive gas concentrations are adequate for the Enhanced Design Alternative II design. It is not expected that any subsequent changes in information would alter the conclusion stated above.

1.2.1.10 Criterion Basis Statement

I. Criterion Need Basis

This requirement supports decontamination of the disposal container/waste package. By limiting surface roughness of the disposal container, the decontamination process will not be impeded. This requirement is derived from the "Mined Geologic Disposal System Functional Analysis Document," function 1.4.3.2.2.4.5, and "Design Criteria for an Independent Spent Fuel Storage Installation (Dry Type)" (ANSI/ANS-57.9-1992), Section 6.2.2.1.2(5).

Limiting surface roughness may also reduce the time required to perform work in the vicinity of radioactive materials, which is required by 10 CFR 63.112(e)(2).

This criterion is supported by Guidance Statement 7.1g1 contained in the "MGR Compliance Program Guidance Package for the Uncanistered Spent Nuclear Fuel (SNF) Disposal Container."

II. Criterion Performance Parameter Basis

A published reference for the recommended surface finish of tools, equipment, casks, containers etc., which may become radioactively contaminated does not exist; therefore, the following rule of thumb based on both Commercial Nuclear and Nuclear Navy experience will be used as the basis for selecting the surface finishes. (Note: The highest number has the roughest finish.)

250 μin: Use in applications where the item is not expected/designed to become radioactively contaminated. However, if exposed to radioactive contamination, this finish can still be decontaminated.

125 μin: Use in applications where the item is more likely to become radioactively contaminated than the 250 case, but still not routinely exposed to contamination.

63 μin: Use in applications where the item is expected to be routinely exposed to radioactive contamination.

The 250 μin value is selected for the disposal container because the container is not expected to become contaminated due to waste handling operations. In addition, the postclosure performance of one finish over another is not a distinguishing factor for long term performance in the models used for TSPA.

1.2.1.11 Criterion Basis Statement

I. Criterion Need Basis

This requirement guides disposal container/waste package design away from a design that would be difficult to decontaminate by precluding undesirable external geometries (e.g., blind holes). This requirement is intended to be assessed against the as-designed disposal container, without regard to actual disposal container use, which would preclude surface visibility (e.g., disposal container emplacement on pedestals would preclude visibility of the pedestal to disposal container contact points).

Also, accessibility of the waste package surfaces to visual inspections may reduce the time required to perform work in the vicinity of radioactive materials, which is required by 10 CFR 63.112(e)(2).

II. Criterion Performance Parameter Basis

N/A

1.2.1.12 Criterion Basis Statement

I. Criterion Need Basis

This criterion supports the tracking of all waste packages as required by MGR RD 3.1.C, MGR RD 3.3.K, and 10 CFR 63.78. This criterion also supports the MGR RD 3.1.B requirement to implement the applicable provisions of "Standards for Protection Against Radiation" (10 CFR 20). Also, identification of waste package contents may reduce the time required to perform work in the vicinity of radioactive materials, which is required by 10 CFR 63.112(e)(2).

Waste packages located in surface and subsurface facilities of the MGR are "accessible only to individuals authorized to ... work in the vicinity of the containers..." and are located in storage vaults or hot cells. Therefore, labeling of waste packages is subject to the exemptions provided by 10 CFR 20.1905(e).

II. Criterion Performance Parameter Basis

N/A

1.2.1.13 Criterion Basis Statement

I. Criterion Need Basis

Label material and method of attachment to the waste package must be considered so that the waste package will not be impaired in its ability to limit the dose rate specified in 10 CFR 63.113(b).

II. Criterion Performance Parameter Basis

N/A

1.2.1.14 Criterion Basis Statement

I. Criterion Need Basis

This requirement establishes the length of time that the labels must be legible. This requirement supports MGR RD 3.1.C and is a decomposition of 10 CFR 63.112(e)(2) in that legibility by remote means may reduce the time required to perform work in the vicinity of radioactive materials. Labels are needed to support the tracking of all waste packages as required by MGR RD 3.3.K and 10 CFR 63.78. This criterion also supports the MGR RD 3.1.D requirement to implement the applicable provisions of "Physical Protection of Plants and Materials" (10 CFR 73), Section 45(d)(1)(iii).

II. Criterion Performance Parameter Basis

N/A

1.2.1.15 Criterion Basis Statement

I. Criterion Need Basis

The requirement preserves the option of filling disposal containers by requiring disposal container design to be compatible with a repository design with or without filler material in the disposal containers. The preservation of the filler option is required in MGR RD 3.3.N.

II. Criterion Performance Parameter Basis

The incomplete list of candidate filler materials is obtained from MGR RD 3.3.N.

1.2.1.16 Criterion Basis Statement

I. Criterion Need Basis

This criterion requires that the disposal container lifting features be designed to withstand handling loads and is needed to reduce the probability of the occurrence of a design basis event in consideration of 10 CFR 63.111(a)(2), 10 CFR 63.111(b)(2), and 10 CFR 63.112(e)(8). Reducing the probability of a design basis event may also reduce the time required to perform work in the vicinity of radioactive materials, which is a consideration of 10 CFR 63.112(e)(2).

II. Criterion Performance Parameter Basis

The factors-of-safety are obtained from Section 4.2.1.1 of the "Radioactive Materials - Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4,500 kg) or More" (ANSI N14.6-1993). The scope of ANSI N14.6-1993 encompasses special lifting devices and those features of the attachment members of the containers that affect the function and safety of the lift.

The Yucca Mountain Project may develop (for use in its disposal container designs) different values based on loading conditions that are representative of repository operations, if they are justified. A technical report would provide the supporting technical justification for the project-specific values along with a rationale for not using ANSI N14.6-1993. The stress design factors specified in ANSI N14.6-1993 will be used in the disposal container designs unless project-specific values are required.

1.2.1.17 Criterion Basis Statement

I. Criterion Need Basis

This criterion requires that the disposal container lifting features be designed to withstand handling loads and is needed to reduce the probability of the occurrence of a design basis event in consideration of 10 CFR 63.111(a)(2), 10 CFR 63.111(b)(2), and 10 CFR 63.112(e)(8). Reducing the probability of a design basis event may also reduce the time required to perform work in the vicinity of radioactive materials, which is a consideration of 10 CFR 63.112(e)(2).

II. Criterion Performance Parameter Basis

The factors-of-safety are obtained from Section 4.2.1.1 of the "Radioactive Materials - Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4,500 kg) or More" (ANSI N14.6-1993). The scope of ANSI N14.6-1993 encompasses special lifting devices and those features of the attachment members of the containers that affect the function and safety of the lift.

The Yucca Mountain Project may develop (for use in its disposal container designs) different values based on loading conditions that are representative of repository operations, if they are justified. A technical report would provide the supporting technical justification for the project-specific values along with a rationale for not using ANSI N14.6-1993. The stress design factors specified in ANSI N14.6-1993 will be used in the disposal container designs unless project-specific values are required.

1.2.1.18 Criterion Basis Statement

I. Criterion Need Basis

This criterion is needed for the disposal container/waste package to comply with 10 CFR 63.111(a)(2), 10 CFR 63.111(b)(2), and 10 CFR 63.112(e)(8), which require the disposal container system perform its intended safety function assuming the occurrence of design basis events.

This criterion is supported by Guidance Statements 6.12g3 and 6.13g1 contained in the "MGR Compliance Program Guidance Package for the Uncanistered Spent Nuclear Fuel (SNF) Disposal Container." The Guidance Statements require the disposal container be designed considering normal loading conditions.

II. Criterion Performance Parameter Basis

N/A

1.2.1.19 Criterion Basis Statement

I. Criterion Need Basis

This criterion is needed in consideration of thermal loads and fire hazards in support of MGR RD 3.1.C, 10 CFR 63.111(a)(2), 10 CFR 63.111(b)(2), and 10 CFR 63.112(e)(8), which require the disposal container system perform its intended safety function assuming the occurrence of design basis events.

II. Criterion Performance Parameter Basis

N/A

1.2.1.20 Criterion Basis Statement

I. Criterion Need Basis

This criterion is needed in consideration of fire hazards (pyrophoric materials), explosion hazards (explosive materials), and thermal loads (conditions resulting in the ignition of a pyrophoric material and the results of an explosion or fire). This criterion supports MGR RD 3.1.C, 10 CFR 63.111(a)(2), 10 CFR 63.111(b)(2), and 10 CFR 63.112(e)(8), which require the disposal container system perform its intended safety function assuming the occurrence of design basis events.

II. Criterion Performance Parameter Basis

N/A

1.2.1.21 Criterion Basis Statement

I. Criterion Need Basis

This criterion is needed to support the functions of the waste package to prevent adverse reactions involving the waste form and to provide conditions needed to maintain the physical and chemical stability of the waste form.

II. Criterion Performance Parameter Basis

N/A

1.2.2.1.1 Criterion Basis Statement

I. Criterion Need Basis

This criterion is needed in consideration of 10 CFR 63.111(a)(2), 10 CFR 63.111(b)(2), and 10 CFR 63.112(e)(8) and to ensure the disposal container system performs its intended safety function assuming the occurrence of a design basis event. The general wording for this requirement is taken from "Nuclear Safety Criteria for the Disposal Container System Description Documents (SDDs)" (Attachment I, p. 1) (Input Transmittal RSO-RSO-99333.Ta).

This criterion is supported by Guidance Statements 6.12g1, 6.12g3, and 6.13g1 contained in the "MGR Compliance Program Guidance Package for the Uncanistered Spent Nuclear Fuel (SNF) Disposal Container."

II. Criterion Performance Parameter Basis

The event parameters are taken from "Nuclear Safety Criteria for the Disposal Container System Description Documents (SDDs)" (Attachment I, p. 1) (Input Transmittal RSO-RSO-99333.Ta).

1.2.2.1.2 Criterion Basis Statement

I. Criterion Need Basis

This criterion is needed in consideration of 10 CFR 63.111(a)(2), 10 CFR 63.111(b)(2), and 10 CFR 63.112(e)(8) and to ensure the disposal container system performs its intended safety function assuming the occurrence of a design basis event. The general wording for this requirement is taken from "Nuclear Safety Criteria for the Disposal Container System Description Documents (SDDs)" (Attachment I, p. 1) (Input Transmittal RSO-RSO-99333.Ta).

This criterion is supported by Guidance Statements 6.12g1, 6.12g3, and 6.13g1 contained in the "MGR Compliance Program Guidance Package for the Uncanistered Spent Nuclear Fuel (SNF) Disposal Container."

II. Criterion Performance Parameter Basis

The event parameters are taken from "Nuclear Safety Criteria for the Disposal Container System Description Documents (SDDs)" (Attachment I, p. 1) (Input Transmittal RSO-RSO-99333.Ta).

1.2.2.1.3 Criterion Basis Statement

I. Criterion Need Basis

This criterion is needed in consideration of 10 CFR 63.111(a)(2), 10 CFR 63.111(b)(2), and 10 CFR 63.112(e)(8) and to ensure the disposal container system performs its intended safety function assuming the occurrence of a design basis event. The general wording for this requirement is taken from "Nuclear Safety Criteria for the Disposal Container System Description Documents (SDDs)" (Attachment I, p. 1) (Input Transmittal RSO-RSO-99333.Ta).

This criterion is supported by Guidance Statements 6.7g1, 6.12g1, 6.12g3, and 6.13g1 contained in the "MGR Compliance Program Guidance Package for the Uncanistered Spent Nuclear Fuel (SNF) Disposal Container."

II. Criterion Performance Parameter Basis

The event parameters are taken from "Nuclear Safety Criteria for the Disposal Container System Description Documents (SDDs)" (Attachment I, p. 1) (Input Transmittal RSO-RSO-99333.Ta).

1.2.2.1.4 Criterion Basis Statement

I. Criterion Need Basis

This criterion is needed in consideration of 10 CFR 63.111(a)(2), 10 CFR 63.111(b)(2), and 10 CFR 63.112(e)(8) and to ensure the disposal container system performs its intended safety function assuming the occurrence of a design basis event. The general wording for this requirement is taken from "Nuclear Safety Criteria for the Disposal Container System Description Documents (SDDs)" (Attachment I, p. 1) (Input Transmittal RSO-RSO-99333.Ta).

This criterion is supported by Guidance Statements 6.7g1, 6.12g1, 6.12g3, and 6.13g1 contained in the "MGR Compliance Program Guidance Package for the Uncanistered Spent Nuclear Fuel (SNF) Disposal Container."

II. Criterion Performance Parameter Basis

The event parameters are taken from "Nuclear Safety Criteria for the Disposal Container System Description Documents (SDDs)" (Attachment I, p. 1) (Input Transmittal RSO-RSO-99333.Ta).

1.2.2.1.5 Criterion Basis Statement

I. Criterion Need Basis

This criterion is needed in consideration of 10 CFR 63.111(a)(2), 10 CFR 63.111(b)(2), and 10 CFR 63.112(e)(8) and to ensure the disposal container system performs its intended safety function assuming the occurrence of a design basis event. The general wording for this requirement is taken from "Nuclear Safety Criteria for the Disposal Container System Description Documents (SDDs)" (Attachment I, p. 1) (Input Transmittal RSO-RSO-99333.Ta).

This criterion is supported by Guidance Statements 6.7g1, 6.12g1, 6.12g3, and 6.13g1 contained in the "MGR Compliance Program Guidance Package for the Uncanistered Spent Nuclear Fuel (SNF) Disposal Container."

II. Criterion Performance Parameter Basis

The event parameters are taken from "Nuclear Safety Criteria for the Disposal Container System Description Documents (SDDs)" (Attachment I, p. 1) (Input Transmittal RSO-RSO-99333.Ta).

1.2.2.1.6 Criterion Basis Statement

I. Criterion Need Basis

This criterion is needed in consideration of 10 CFR 63.111(a)(2), 10 CFR 63.111(b)(2), and 10 CFR 63.112(e)(8) and to ensure the disposal container system performs its intended safety function assuming the occurrence of a design basis event. The general wording for this requirement is taken from "Nuclear Safety Criteria for the Disposal Container System Description Documents (SDDs)" (Attachment I, p. 1) (Input Transmittal RSO-RSO-99333.Ta).

This criterion is supported by Guidance Statements 6.12g1, 6.12g3, and 6.13g1 contained in the "MGR Compliance Program Guidance Package for the Uncanistered Spent Nuclear Fuel (SNF) Disposal Container."

II. Criterion Performance Parameter Basis

The event parameters are taken from "Nuclear Safety Criteria for the Disposal Container System Description Documents (SDDs)" (Attachment I, p. 1) (Input Transmittal RSO-RSO-99333.Ta).

1.2.2.1.7 Criterion Basis Statement

I. Criterion Need Basis

This criterion is needed in consideration of 10 CFR 63.111(a)(2), 10 CFR 63.111(b)(2), and 10 CFR 63.112(e)(8) and to ensure the disposal container system performs its intended safety function assuming the occurrence of a design basis event. The general wording for this requirement is taken from "Nuclear Safety Criteria for the Disposal Container System Description Documents (SDDs)" (Attachment I, p. 2) (Input Transmittal RSO-RSO-99333.Ta).

This criterion is supported by Guidance Statements 6.12g1, 6.12g3, and 6.13g1 contained in the "MGR Compliance Program Guidance Package for the Uncanistered Spent Nuclear Fuel (SNF) Disposal Container."

II. Criterion Performance Parameter Basis

The subsurface event parameters are taken from "Nuclear Safety Criteria for the Disposal Container System Description Documents (SDDs)" (Attachment I, pp. 2 and 3) (Input Transmittal RSO-RSO-99333.Ta).

1.2.2.1.8 Criterion Basis Statement

I. Criterion Need Basis

This criterion is needed in consideration of 10 CFR 63.111(a)(2), 10 CFR 63.111(b)(2), and 10 CFR 63.112(e)(8) and to ensure the disposal container system performs its intended safety function assuming the occurrence of a design basis event. The general wording for this requirement is taken from "Nuclear Safety Criteria for the Disposal Container System Description Documents (SDDs)" (Attachment I, p. 3) (Input Transmittal RSO-RSO-99333.Ta).

This criterion is supported by Guidance Statements 6.12g1, 6.12g3, and 6.13g1 contained in the "MGR Compliance Program Guidance Package for the Uncanistered Spent Nuclear Fuel (SNF) Disposal Container."

II. Criterion Performance Parameter Basis

The event parameters are taken from "Nuclear Safety Criteria for the Disposal Container System Description Documents (SDDs)" (Attachment I, p. 3) (Input Transmittal RSO-RSO-99333.Ta).

1.2.2.1.9 Criterion Basis Statement

I. Criterion Need Basis

This criterion is needed in consideration of 10 CFR 63.111(a)(2), 10 CFR 63.111(b)(2), and 10 CFR 63.112(e)(8) and to ensure the disposal container system performs its intended safety function assuming the occurrence of a design basis event. The general wording for this requirement is taken from "Nuclear Safety Criteria for the Disposal Container System Description Documents (SDDs)" (Attachment I, p. 3) (Input Transmittal RSO-RSO-99333.Ta).

This criterion is supported by Guidance Statements 6.12g1, 6.12g3, and 6.13g1 contained in the "MGR Compliance Program Guidance Package for the Uncanistered Spent Nuclear Fuel (SNF) Disposal Container."

II. Criterion Performance Parameter Basis

The event parameters are taken from "Nuclear Safety Criteria for the Disposal Container System Description Documents (SDDs)" (Attachment I, p. 3) (Input Transmittal RSO-RSO-99333.Ta).

1.2.2.1.10 Criterion Basis Statement

I. Criterion Need Basis

This criterion is needed in consideration of 10 CFR 63.111(a)(2), 10 CFR 63.111(b)(2), and 10 CFR 63.112(e)(8) and to ensure the disposal container system performs its intended safety function assuming the occurrence of a design basis event (the rupture of 100 percent of the fuel rods). The general wording for this requirement is taken from "Nuclear Safety Criteria for the Disposal Container System Description Documents (SDDs)" (Attachment I, p. 3) (Input Transmittal RSO-RSO-99333.Ta).

This criterion is supported by Guidance Statements 6.12g1, 6.12g3, and 6.13g1 contained in the "MGR Compliance Program Guidance Package for the Uncanistered Spent Nuclear Fuel (SNF) Disposal Container."

II. Criterion Performance Parameter Basis

The event parameters are taken from "Nuclear Safety Criteria for the Disposal Container System Description Documents (SDDs)" (Attachment I, p. 3) (Input Transmittal RSO-RSO-99333.Ta).

1.2.2.1.11 Criterion Basis Statement

I. Criterion Need Basis

This criterion is needed in consideration of 10 CFR 63.111(a)(2), 10 CFR 63.111(b)(2), and 10 CFR 63.112(e)(8) and to ensure the disposal container system performs its intended safety function assuming the occurrence of a design basis event. This criterion is in consideration of a fire event that is defiined in "Packaging and Transportation of Radioactive Materials" (10 CFR 71), Section 73(c)(4).

Until a comprehensive analysis of fire hazards at the MGR is performed and establishes the credibility and/or magnitude of a design basis fire for the waste package, the hypothetical fire criteria for transportation casks, from 10 CFR 71.73(c)(4), is assumed. Therefore, this criterion remains to be verified.

The fire event is defined as exposure of the waste package fully engulfed in an average flame temperature of at least 800 degrees C (1,475 degrees F) for a period of 30 minutes, with an average emissivity coefficient of at least 0.9. For purposes of calculation, the surface absorptivity shall be either that value which the waste package may be expected to possess if exposed to the fire specified or 0.8, whichever is greater, and the convective coefficient shall be that value which may be demonstrated to exist if the waste package were exposed to the fire specified.

The general wording for this requirement is taken from "Nuclear Safety Criteria for the Disposal Container System Description Documents (SDDs)" (Attachment I, p. 3) (Input Transmittal RSO-RSO-99333.Ta).

This criterion is supported by Guidance Statements 6.7g2, 6.12g1, 6.12g3, and 6.13g1 contained in the "MGR Compliance Program Guidance Package for the Uncanistered Spent Nuclear Fuel (SNF) Disposal Container."

II. Criterion Performance Parameter Basis

The criterion is taken from "Nuclear Safety Criteria for the Disposal Container System Description Documents (SDDs)" (Attachment I, p. 3) (Input Transmittal RSO-RSO-99333.Ta).

1.2.2.1.12 Criterion Basis Statement

I. Criterion Need Basis

This requirement applies the criticality requirement from 10 CFR 63.112(e)(6) to the disposal container/waste package design during the preclosure period.

The general wording for this requirement is taken from "Nuclear Safety Criteria for the Disposal Container System Description Documents (SDDs)" (Attachment I, pp. 3 and 4) (Input Transmittal RSO-RSO-99333.Ta). The reference provides input regarding Design Basis Events, therefore this requirement partially supports 10 CFR 63.111(a)(2), 10 CFR 63.111(b)(2), and 10 CFR 63.112(e)(8).

This criterion is supported by Guidance Statements 6.12g1 and 6.13g1 contained in the "MGR Compliance Program Guidance Package for the Uncanistered Spent Nuclear Fuel (SNF) Disposal Container."

II. Criterion Performance Parameter Basis

The performance parameters for this requirement are taken from "Nuclear Safety Criteria for the Disposal Container System Description Documents (SDDs)" (Attachment I, pp. 3 and 4) (Input Transmittal RSO-RSO-99333.Ta). Sections IV.1 and IV.2 (p. 6-1) of "Standard Review Plan for Dry Cask Storage Systems" (NUREG-1536) support the effective multiplication factor parameter.

1.2.2.1.13 Criterion Basis Statement

I. Criterion Need Basis

This criterion is needed to ensure the expected annual dose to the critical group during the first 10,000 years after permanent closure does not exceed 25 mrem, as required by 10 CFR 63.113(b).

II. Criterion Performance Parameter Basis

The actual content and quantification of the postclosure criticality criterion has not been confirmed. As a result, this criterion will need to be verified.

The 1 percent inventory increase measured at 1,000 years was chosen as the standard below which the consequences of postclosure criticalities would be held because the effect of the relatively small incremental increase in the inventory (of 1 percent due to criticality events) will be negligible compared to the effect of other total system performance related parameters, such as water infiltration rate, which can vary by over 100 percent.

1.2.3.1 Criterion Basis Statement

I. Criterion Need Basis

This requirement defines the external (outside the waste package) environment for which the disposal container should be designed. 10 CFR 63.113(b) is traced because this requirement considers the waste package influenced emplacement drift environment and its impact on the capability of the disposal container system to limit the expected annual dose to the average member of the critical group to 25 mrem at any time during the first 10,000 years after permanent closure of the repository.

Also, in consideration of MGR RD 3.4.2.C, this criterion defines the induced handling environment (credible loads) the disposal container/waste package must withstand.

II. Criterion Performance Parameter Basis

The environment parameters are taken from "Performance Allocation Study Preliminary Results" (Table 4), which is the attachment to the input transmittal entitled "Manager System Requirements/System Description Documents." The induced handling environments are to be determined.

1.2.4.1 Criterion Basis Statement

I. Criterion Need Basis

This criterion is needed to ensure mechanical interface consistency between the design of this system and the repository surface facilities and systems. This is done by specifying the design be done in accordance with the interface agreements defined in "Interface Control Document for the Waste Packages/Disposal Containers and the Surface Repository Facilities and Systems for Mechanical, Envelope and Functional Interfaces." This criterion supports the waste handling operations of MGR RD 3.2.C.

II. Criterion Performance Parameter Basis

N/A

1.2.4.2 Criterion Basis Statement

I. Criterion Need Basis

This criterion is needed to ensure mechanical interface consistency between the design of this system and the subsurface facilities and systems. This is done by specifying the design be done in accordance with the interface agreements defined in "Interface Control Document for Waste Packages and the Mined Geologic Disposal System Repository Subsurface Facilities and Systems for Mechanical and Envelope Interfaces." This criterion supports the waste handling operations of MGR RD 3.2.C.

II. Criterion Performance Parameter Basis

N/A

1.2.4.3 Criterion Basis Statement

I. Criterion Need Basis

This requirement is needed as an interface between the waste package and the waste emplacement system to allow adequate waste package transporter shielding design for an acceptable dose rate at the external surfaces of the transporter (in support of "Standards for Protection Against Radiation" [10 CFR 20], Subparts A, B, and C). This requirement is not intended to yield disposal container design features that are added solely for the purpose of shielding (unshielded waste packages are recommended in the "Waste Package Size Study Report," p. 6-5), but is intended to establish the expected maximum dose rate the waste emplacement system will be designed to reduce.

This criterion is supported by Guidance Statements 6.12g1 and 6.13g1 contained in the "MGR Compliance Program Guidance Package for the Uncanistered Spent Nuclear Fuel (SNF) Disposal Container."

II. Criterion Performance Parameter Basis

The maximum dose rate at the external surface of the waste packages has not yet been determined.

1.2.4.4 Criterion Basis Statement

I. Criterion Need Basis

This criterion is required for the system to comply with management direction put into effect via the "Monitored Geologic Repository Project Description Document," which places a constraint on the maximum heat output of individual waste packages. This criterion is also required to allow the design of the transporter used in the Waste Emplacement/Retrieval System. A maximum heat load criterion provides a bounding heat load that must be sustained by the transporter during emplacement operations.

II. Criterion Performance Parameter Basis

The maximum thermal output limit is obtained from the "Monitored Geologic Repository Project Description Document" (Section 2.2.1.1.11).

1.2.4.5 Criterion Basis Statement

I. Criterion Need Basis

This requirement is needed to comply with MGR RD 3.1.A and 3.2.A. This criterion defines the split of commercial SNF disposed of in the uncanistered SNF disposal container, the canistered SNF disposal container, and the non-fuel components disposal container.

II. Criterion Performance Parameter Basis

The maximum quantity of commercial SNF for the first repository is taken from MGR RD 3.2.A. This requirement is intended to be updated in the future as a more detailed interface between this disposal container and the canistered SNF disposal container is developed. When updated, it will indicate the split of MTU between the uncanistered SNF disposal container and the canistered SNF disposal container.

1.2.4.6 Criterion Basis Statement

I. Criterion Need Basis

This criterion is required to align the disposal container design with the surface repository disposal container handling operations.

II. Criterion Performance Parameter Basis

N/A

1.2.4.7 Criterion Basis Statement

I. Criterion Need Basis

This criterion is required to align the disposal container/waste package design with the surface repository disposal container handling operations.

II. Criterion Performance Parameter Basis

N/A

1.2.4.8 Criterion Basis Statement

I. Criterion Need Basis

This criterion is needed to ensure the disposal container design accommodates welding and sealing equipment used by the Disposal Container Handling System and facilitates optimum welding times.

II. Criterion Performance Parameter Basis

N/A

1.2.6.1 Criterion Basis Statement

I. Criterion Need Basis

In support of MGR RD 3.3.A, the "1995 ASME Boiler and Pressure Vessel Code" (Section III, Division 1, Subsection NG-1995) provides nuclear industry specific codes, standards and conformity assessment programs. American Society of Mechanical Engineers (ASME) codes and standards are internationally recognized for the design, manufacturing and installation of mechanical devices. Requirements set forth in "Domestic Licensing of Production and Utilization Facilities" (10 CFR 50), Section 55 are specific in the use of ASME Boiler and Pressure Codes as "quality standards commensurate with the importance of the safety function to be performed" (10 CFR 50.55(a)(1)).

II. Criterion Performance Parameter Basis

N/A

1.2.6.2 Criterion Basis Statement

I. Criterion Need Basis

In support of MGR RD 3.3.A, the "1995 ASME Boiler and Pressure Vessel Code" (Section III, Division 1, Subsection NB-1995) provides nuclear industry specific codes, standards and conformity assessment programs. ASME codes and standards are internationally recognized for the design, manufacturing and installation of mechanical devices. Requirements set forth in "Domestic Licensing of Production and Utilization Facilities" (10 CFR 50), Section 55 are specific in the use of ASME Boiler and Pressure Codes as "quality standards commensurate with the importance of the safety function to be performed" (10 CFR 50.55(a)(1)).

II. Criterion Performance Parameter Basis

N/A

1.2.6.3 Criterion Basis Statement

I. Criterion Need Basis

The criterion supports MGR RD 3.3.A. "Nuclear Criticality Control of Special Actinide Elements" (ANSI/ANS-8.15-1981) is cited as an industry standard used in the development of the "Disposal Criticality Analysis Methodology Topical Report."

II. Criterion Performance Parameter Basis

N/A

1.2.6.4 Criterion Basis Statement

I. Criterion Need Basis

The criterion supports MGR RD 3.3.A. "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors" (ANSI/ANS-8.1-1998) is cited as an industry standard used in the development of the "Disposal Criticality Analysis Methodology Topical Report."

II. Criterion Performance Parameter Basis

N/A

1.2.6.5 Criterion Basis Statement

I. Criterion Need Basis

The criterion supports MGR RD 3.3.A. "Criteria for Nuclear Safety Controls in Operations with Shielding and Confinement" (ANSI/ANS-8.10-1983) is cited as an industry standard used in the development of the "Disposal Criticality Analysis Methodology Topical Report."

II. Criterion Performance Parameter Basis

N/A

1.2.6.6 Criterion Basis Statement

I. Criterion Need Basis

The criterion supports MGR RD 3.3.A. "Criticality Safety Criteria for the Handling, Storage, and Transportation of LWR Fuel Outside Reactors" (ANSI/ANS-8.17-1984) is cited as an industry standard used in the development of the "Disposal Criticality Analysis Methodology Topical Report."

II. Criterion Performance Parameter Basis

N/A

APPENDIX B ARCHITECTURE AND CLASSIFICATION

The QA classification as established in "Classification of the MGR Uncanistered Spent Nuclear Fuel Disposal Container System" defines the overall system as QL-1. The next level of system architecture and assumed QA classification are identified in Table I-10.

Table I-10. System Architecture and Quality Assurance Classification

System Architecture	QL-1	QL-2	QL-3	CQ
Uncanistered Spent Nuclear Fuel Disposal Container System	X			
21 PWR / No Absorber Plates	X			
21 PWR / With Absorber Plates	X			
21 PWR / With Absorber Rods / No Absorber Plates	X			
44 BWR / No Absorber Plates	X			
44 BWR / With Absorber Plates	X			
24 BWR / With Thick Absorber Plates	X			

Note: For definition of acronyms, symbols and units, see Appendix C

APPENDIX C ACRONYMS, SYMBOLS, AND UNITS

This section provides a listing of acronyms, symbols, and units used in Volume I.

C.1 Acronyms

ASME	American Society of Mechanical Engineers
BWR	Boiling water reactor
DOE	U. S. Department of Energy
F	Function
CQ	Conventional Quality
CRWMS	Civilian Radioactive Waste Management System
DHLW	Defense High-Level Waste
DOE	U. S. Department of Energy
M&O	Management and Operating Contractor
MGDS	Mined Geologic Disposal System
MGR	Monitored Geologic Repository
MGR RD	Monitored Geologic Repository Requirements Document
NFC	Non-fuel Component
NRC	U. S. Nuclear Regulatory Commission
PWR	Pressurized Water Reactor
QA	Quality Assurance
QL	Quality Level
SDD	System Description Document
SNF	Spent Nuclear Fuel
TBD	To Be Determined
TBV	To Be Verified
TSPA-VA	Total System Performance Assessment-Viability Assessment

C.2 Symbols

CO_2	carbon dioxide
CO	carbon monoxide
g	acceleration due to gravity
H_2	hydrogen
H_2O	water
O_2	oxygen
C	Celsius
F	Fahrenheit

C.3 Units

cm	centimeter
ft	feet
hr	hour
Hz	Hertz
in.	inch
kg	kilogram
km	kilometer
kW	kilowatt
lb	pound
m	meter
mg	milligram
mi	mile
ml	milliliter
mm	millimeter
mrem	one thousandth of a rem
MPa	megapascal
MT	metric ton
MTU	metric tons uranium
pH	hydrogen ion concentration potential
psia	pounds per square inch-atmospheric
rem	Roentgen equivalent man
sec	second
yr	year
μin	microinch
μm	micrometer

APPENDIX D FUTURE REVISION RECOMMENDATIONS AND ISSUES

This appendix identifies issues and actions that require further evaluation. The disposition of these issues and actions could alter the functions and design criteria that are allocated to this system in future revisions to this document. However, the issues and actions identified in this appendix do not require TBDs or TBVs beyond those already identified.

Issue 1-- Label Legibility

Future criteria must consider observation by both human and electronic means.

Issue 2-- Disposal Container Lid Matching

The need for unique identification of the fabricated disposal container to a set of matched inner and outer lids needs to be investigated. If disposal containers will be matched to their lids, an identification system between the disposal container and its lids will be needed. The design criteria for such an identification will then need to be identified.

Issue 3-- Disposal Container Capacity

The identification of the individual disposal container capacities (21 PWR, 24 BWR, 44 BWR) will be identified as an interface with the Assembly Transfer System.

Issue 4-- Handling Interface

The handling interface with the Disposal Container Handling System needs to be identified for empty disposal container handling, loaded disposal container handling, and lid handling.

Issue 5-- Welding and Inspection Interface

The welding and inspection interface with the Disposal Container Handling System needs to be identified for inner and outer lid welding operations.

Issue 6-- Inerting Interface and Operational Requirements

The filling of the container with an inert gas needs to be identified as an interface with the Disposal Container Handling System. Applicable disposal container requirements need to be determined.

Issue 7-- Metric vs. Standard Units

Consistent display of both metric and standard units should be incorporated into criteria. Conversion leads to inconsistencies in the number of significant digits, accuracy, and summed values.

Issue 8-- Identification as Disposal Container or Waste Package

A consistent method of identification of the waste form container as a "disposal container" or a "waste package" needs to be determined.

Issue 9-- Disposal Container Materials of Construction

Criterion 1.2.1.4, which is a design constraint imposed by the "Monitored Geologic Repository Project Description Document," will be moved to the Design Description section (Section 2) of this document in the next revision.

Issue 10-- Waste Package Performance

Performance allocations for the waste packages to meet overall regulatory requirements need to be determined.

Issue 11-- Waste Package Postclosure Criticality Requirements

Postclosure criticality requirements will be determined for a future revision.

Issue 12—Assembly Class Dimensions

References used for the dimensions of the assembly classes listed in Criterion 1.2.1.1 must be reconciled with new data anticipated to be issued in the future. The data is not critical for Site Recommendation. The next revision of this document will include reconciled data.

Issue 13—Single Element Canister Dimensions

The next revision to this document will update Criterion 1.2.1.2 (single element canister dimensions) to be consistent with the programmatic interface agreement.

APPENDIX E REFERENCES

This section provides a listing of references used in Volume I.

"1995 ASME Boiler and Pressure Vessel Code." The American Society of Mechanical Engineers. 1995 Edition. July 1, 1995. New York, New York: The American Society of Mechanical Engineers. TIC: 245287.

"Characteristics of Commercial SNF Assemblies to be Disposed of at the MGDS." CRWMS M&O. BBAA00000-01717-0200-00061, Rev. 01. March 26, 1998. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19980417.0685.

"Characteristics of Potential Repository Wastes." Oak Ridge National Laboratory. DOE/RW-0184-R1. July 1992. Oak Ridge, Tennessee: U.S. Department of Energy, Office of Scientific and Technical Information. TIC: 204940.

"Classification of the MGR Uncanistered Spent Nuclear Fuel Disposal Container System." CRWMS M&O. ANL-UDC-SE-000001, Rev. 00. August 31, 1999. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19990928.0216.

"Criteria for Nuclear Criticality Safety Controls in Operations with Shielding and Confinement." American Nuclear Society. ANS 8.10. 1983. Chicago, Illinois: American Nuclear Society. TIC: 205015.

"Criticality Safety Criteria for the Handling, Storage, and Transportation of LWR Fuel Outside Reactors." American Nuclear Society. ANS 8.17. 1984. La Grange Park, Illinois: American Nuclear Society. TIC: 2497.

"Deficiency Report VAMO-98-D-132." Office of Civilian Radioactive Waste Management. September 25, 1998. Washington, D.C.: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ACC: MOL.19991029.0063.

"Design Criteria for an Independent Spent Fuel Storage Installation (Dry Type)." American Nuclear Society. ANSI/ANS-57.9-1992. 1992. La Grange Park, Illinois: American Nuclear Society. TIC: 3043.

"Disposal Criticality Analysis Methodology Topical Report." CRWMS M&O. YMP/TR-004Q, Rev. 0. November 1998. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19990308.0035.

"Domestic Licensing of Production and Utilization Facilities." Nuclear Regulatory Commission. 10 CFR 50. 1998. Washington, D.C.: U.S. Government Printing Office. TIC: 238441.

"Evaluation of Cover Gas Impurities and Their Effects on the Dry Storage of LWR Spent Fuel." Knoll, R. W., Gilbert, E. R., Battelle Memorial Institute. PNL-6365. November 1987. Richland, Washington: Pacific Northwest Laboratory. TIC: 213789.

"Interface Control Document for the Waste Packages/Disposal Containers and the Surface Repository Facilities and Systems for Mechanical, Envelope and Functional Interfaces." CRWMS M&O. B00000000-01717-8100-00021. November 30, 1998. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19981229.0160.

"Interface Control Document for Waste Packages and the Mined Geologic Disposal System Repository Subsurface Facilities and Systems for Mechanical and Envelope Interfaces." CRWMS M&O. B00000000-01717-8100-00009, Rev. 00. May 31, 1998. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19980826.0139.

"Manager System Requirements/System Description Documents." CRWMS M&O. RSO-RSO-99311.T. October 15, 1999. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19991019.0380.

"MGR Compliance Program Guidance Package for the Uncanistered Spent Nuclear Fuel (SNF) Disposal Container." CRWMS M&O. BBA000000-01717-5600-00001, Rev. 01, ICN 01. October 15, 1999. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19991022.0190.

"Mined Geologic Disposal System Functional Analysis Document." CRWMS M&O. B00000000-01717-1708-00006, Rev. 01. September 26, 1996. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19961217.0184.

"Monitored Geologic Repository Draft Disposability Interface Specification." CRWMS M&O. B00000000-01717-4600-00108, Rev. 01. August 1998. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19981102.0001.

"Monitored Geologic Repository Project Description Document." CRWMS M&O. B00000000-01717-1705-00003, Rev. 00, DCN 01. October 29, 1999. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19991117.0160.

"Monitored Geologic Repository Requirements Document." U.S. Department of Energy. YMP/CM-0025, Rev. 3, DCN 01. April 1999. Las Vegas, Nevada: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ACC: MOL.19990429.0228.

"Nuclear Criticality Control of Special Actinide Elements." American Nuclear Society. ANSI/ANS-8.15-1981. 1981. La Grange Park, Illinois: American Nuclear Society. TIC: 231624.

"Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors." American Nuclear Society. ANS 8.1. 1998. La Grange Park, Illinois: American Nuclear Society. Readily Available.

"Nuclear Safety Criteria for the Disposal Container System Description Documents (SDDs)." CRWMS M&O. RSO-RSO-99333.Ta. November 17, 1999. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19991117.0105.

"Packaging and Transportation of Radioactive Material." Nuclear Regulatory Commission. 10 CFR 71. January 1, 1998. Washington, D.C.: U.S. Government Printing Office.

"Performance Assessment Input for Performance Allocation Study." CRWMS M&O. RSO-PA-99262.T. November 3, 1999. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19991110.0273.

"Physical Protection of Plants and Materials." Nuclear Regulatory Commission. 10 CFR 73. January 1, 1998. Washington, D.C.: U.S. Government Printing Office. TIC: 238423.

"Preliminary Draft Requirements from the Monitored Geologic Repository Requirements Document." CRWMS M&O. RSO-RSO-99324.Ta. November 10, 1999. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19991115.0135.

"Qualification of Spent Nuclear Fuel Assembly Characteristics for Use as a Design Basis." CRWMS M&O. B00000000-01717-0200-00002, Rev. 04. March 11, 1996. Vienna, Virginia: CRWMS M&O. ACC: MOV.19960731.0003.

"Quality Assurance Requirements and Description." U.S. Department of Energy. DOE/RW-0333P, Rev. 8. June 5, 1998. Washington, D.C.: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ACC: MOL.19980601.0022.

"Radioactive Materials - Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4,500 Kg) or More." American National Standards Institute. ANSI N14.6-1993. 1993. New York, New York: American National Standards Institute. TIC: 236261.

"Revised Interim Guidance Pending Issuance of New U.S. Nuclear Regulatory Commission (NRC) Regulations (Revision 01, July 22, 1999), for Yucca Mountain, Nevada." U.S. Department of Energy. OL&RC:SB-1714. September 3, 1999. North Las Vegas, Nevada: U.S. Department of Energy, Office of Civilian Radioactive Waste Management, Yucca Mountain Site Characterization Office. ACC: MOL.19990910.0079.

"SDD Development/Maintenance (Q SDDs) (WP# 16012126M5)." CRWMS M&O. October 11, 1999. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19991025.0001.

"Standard Review Plan for Dry Cask Storage Systems." U.S. Nuclear Regulatory Commission. NUREG-1536, Final Report. January 1997. Washington, D.C.: U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation. TIC: 232373.

"Standards for Protection Against Radiation." Nuclear Regulatory Commission. 10 CFR 20. 1998. Washington, D.C.: U.S. Government Printing Office. TIC: 238505.

"Technical Reports." CRWMS M&O. AP-3.11Q, Rev. 0, ICN 1. November 24, 1999. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19991130.0149.

"The Technical Basis for the Classification of Failed Fuel in the Back-End of the Fuel Cycle." Jones, Robert H. EPRI TR-108237, Final Report. July 1997. Los Gatos, California: Electric Power Research Institute. TIC: 236839.

"Thermal Evaluation of Preliminary 21 PWR AUCF Design." CRWMS M&O. BBAA00000-01717-0200-00011, Rev. 00. May 10, 1996. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19961014.0117.

"Thermal Loading Study for FY 1996." CRWMS M&O. B00000000-01717-5705-00044, Rev. 01. November 8, 1996. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19961217.0121.

"Waste Package Size Study Report." CRWMS M&O. B00000000-01717-5705-00079, Rev. 01. July 20, 1998. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19981020.0108.

APPENDIX F MULTIPLE REACTOR CLASSES

This Appendix provides a listing (from "Characteristics of Potential Repository Wastes," Table 2.2.2 [TBV-455]) of the individual reactors that use assemblies from a particular multiple reactor class.

GENERAL ELECTRIC BWR/2,3

Dresden 2 (BWR/3)
Dresden 3 (BWR/3)
Millstone 1 (BWR/3) *
Monticello (BWR/3)
Nine Mile Point (BWR/3)
Oyster Creek (BWR/2)
Pilgrim (BWR/3)
Quad Cities 1 (BWR/3)
Quad Cities 2 (BWR/3)

GENERAL ELECTRIC BWR/4-6

Browns Ferry 1 (BWR/4)
Browns Ferry 2 (BWR/4)
Browns Ferry 3 (BWR/4)
Brunswick 1 (BWR/4)
Brunswick 2 (BWR/4)
Clinton (BWR/6)
Cooper Station (BWR/4)
Duane Arnold (BWR/4)
Enrico Fermi 2 (BWR/4)
Fitzpatrick (BWR/4)
Grand Gulf 1 (BWR/6)
Grand Gulf 2 (BWR/6) **
Hatch 1 (BWR/4)
Hatch 2 (BWR/4)
Hope Creek (BWR/4)
LaSalle 1 (BWR/5)
LaSalle 2 (BWR/5)
Limerick 1 (BWR/4)
Limerick 2 (BWR/4)
Nine Mile Point 2 (BWR/5)
Peach Bottom 2 (BWR/4)
Peach Bottom 3 (BWR/4)
Perry 1 (BWR/6)
Perry 2 (BWR/6) **
River Bend 1 (BWR/6)
Shoreham (BWR/4) *
Susquehanna 1 (BWR/4)
Susquehanna 2 (BWR/4)
Vermont Yankee (BWR/4)
Washington Nuclear 2 (BWR/5)

BABCOCK & WILCOX 15 X 15

Arkansas Nuclear One, Unit 1
Crystal River 3
Davis-Besse
Oconee 1
Oconee 2
Oconee 3
Rancho Seco *
Three Mile Island 1
Three Mile Island 2 *

* Reactors are permanently shutdown.
** Reactor completion (or continued commercial use) is uncertain at this time.

BABCOCK & WILCOX 17 X 17

Bellefonte 1 **
Bellefonte 2 **
Washington Nuclear 1 **

COMBUSTION ENGINEERING 14 X 14

Calvert Cliffs 1
Calvert Cliffs 2
Maine Yankee *
Millstone 2
St. Lucie 1

COMBUSTION ENGINEERING 16 X 16

Arkansas Nuclear One, Unit 2
San Onofre 2
San Onofre 3
Waterford 3

COMBUSTION ENGINEERING SYSTEM 80

Palo Verde 1
Palo Verde 2
Palo Verde 3
Washington Nuclear 3 **

WESTINGHOUSE 14 X 14

Ginna
Kewaunee
Point Beach 1
Point Beach 2
Prairie Island 1
Prairie Island 2

WESTINGHOUSE 15 X 15

Cook 1
Indian Point 2
Indian Point 3
Robinson 2
Surry 1
Surry 2
Turkey Point 3
Turkey Point 4
Zion 1 *
Zion 2 *

* Reactors are permanently shutdown.
** Reactor completion (or continued commercial use) is uncertain at this time.

WESTINGHOUSE 17 X 17

Beaver Valley 1	Beaver Valley 2
Braidwood 1	Braidwood 2
Byron 1	Byron 2
Callaway	Catawba 1
Catawba 2	Comanche Peak 1
Comanche Peak 2	Cook 2
Diablo Canyon 1	Diablo Canyon 2
Farley 1	Farley 2
Harris	McGuire 1
McGuire 2	Millstone 3
North Anna 1	North Anna 2
Salem 1	Salem 2
Seabrook 1	Sequoyah 1
Sequoyah 2	Summer
Trojan *	Vogtle 1
Vogtle 2	Watts Bar 1
Watts Bar 2 **	Wolf Creek

SOUTH TEXAS

South Texas 1	South Texas 2

* Reactors are permanently shutdown.
** Reactor completion (or continued commercial use) is uncertain at this time.