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TBV-094; TBV-096; TBV-238; TBV-241; TBV-245; TBV-250; TBV-455; TBV-1246; TBV-4743

OFFICE OF CIVILIAN RADIOACTIVE WASTE MANAGEMENT
SYSTEM DESCRIPTION DOCUMENT REVISION HISTORY

1. SDD Title
Uncanistered Spent Nuclear Fuel Disposal Container System Description Document

2. Document Identifier (Including Rev. No. and Change No., if applicable)
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3. Revision	4. Description of Revision
00	Initial Issue This document is a complete revision of the superseded BBA000000-01717-1705-00004. 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."
01	Issued This is a complete revision. Added Section 2 (Design Description) to support Site Recommendation. Deleted Criteria 1.2.1.5, 1.2.4.1 and 1.2.4.2, and Sections 1.4.1 and 1.4.2, as they were deemed unnecessary by management decision. Revised Criteria 1.2.2.1.5, 1.2.2.1.7, 1.2.2.1.12, and 1.2.2.1.13 as a result of updated design information. Added Criteria 1.2.1.22, 1.2.1.23, 1.2.1.24, and 1.2.2.1.14 as a result of new design information. Revised format to eliminate two-volume configuration in compliance with the revised development plan.
01 ICN 01	Added Criterion 1.2.1.5 and revised Criterion 1.2.3.1 to support the Flexible Operations Concept of the MGR. Revised the Quality Assurance statement to include the relevant Technical Work Plan. Corrected reference in Criterion Performance Parameter Basis for Criterion 1.2.1.6. Updated references "Monitored Geologic Repository Project Description Document" to Rev 02 ICN 02, "Waste Package Design Sensitivity Report" to Rev 01, and "Technical Reports" to Rev 02. Added reference "Standard Review Plan for Dry Cask Storage Systems." Changes made to this document as part of this ICN are indicated by a change bar in the right margin.

CONTENTS

TABLES

FIGURES

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, outer cylinder lids (two on the top, one on the bottom), inner cylinder lids (one on the top, one on the bottom), 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. The waste package interfaces with the Emplacement Drift System's emplacement drift pallets upon which the waste packages are placed. 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 (QA) program applies to the development of this document. The activity evaluation for work package number 11612125N4, included in "Technical Work Plan for: Waste Package Design Description for SR" (p. 36), has determined the development of this document is subject to "Quality Assurance Requirements and Description" requirements. Electronic management of data is also controlled in accordance with "Technical Work Plan for: Waste Package Design Description for SR." This document was developed in accordance with AP-3.11Q, "Technical Reports."

1. SYSTEM 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 QA 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. (A later code edition acceptable to the U.S. Nuclear Regulatory Commission and the U.S. Department of Energy may be used in the final design.) For "Level D" (design basis event) conditions, breach has occurred, analytically, when 0.9 of the ultimate tensile strength is exceeded through the outer barrier thickness.

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 waste package'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 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) 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 (SDD). For the applicable version of the codes, standards, and regulatory documents, 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 1 and 2 (TBV-455). Tables 1 and 2 also identify parameters (size, weight, and inventory) that are provided for information only. 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 1. BWR Fuel

Assembly Class	Length* (in.)	Chan-neled Width* (in.)	Unchan-neled 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 x 10^3
Humboldt Bay	96	4.8	4.67	276	23	299	390	0.4 x 10^3
LaCrosse	103.5	5.91	5.62	386	69	455	333***	0.3 x 10^3 (333***)
Dresden 1	135.7	4.57	4.28	328	30	358	892	0.9 x 10^3
GE BWR 2, 3	173	5.61	5.36	619	80	699	18,813	3.5 x 10^4
GE BWR 4-6	177.8	5.61	5.36	588	80	668	39,295	1.3 x 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 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 x 10^3 (76**)
San Onofre 1	138.4	139.9	7.76	1,247	107	1,354	665 (665**)	1.0 x 10^3 (822**)
Haddam Neck	138.4	139.9	8.5	1,255	166	1,421	892 (888**)	1.5 x 10^3 (945**)
Indian Point 1	139.1	139.1	6.27	437	N/A	437	160 (160**)	0.2 x 10^3 (160**)
Fort Calhoun	147.7	158.5	8.12	1,220	67	1,287	570	1.1 x 10^3
Palisades	148.9	148.9	8.31	1,360	N/A	1,360	793	1.5 x 10^3
CE 14x14	158.8	169.6	8.11	1,270	77	1,347	4565	9.8 x 10^3
St. Lucie	159.7	170.6	8.13	1,300	66	1,366	544	1.9 x 10^3
WE 15x15	161.4	166.9	8.42	1,472	165	1,637	7,490	1.5 x 10^4
WE 14x14	161.4	166.3	7.76	1,302	130	1,432	4,093	7.8 x 10^3
WE 17x17	161.4	168.8	8.42	1,482	180	1,662	15,295	5.9 x 10^4
B&W 17x17	167.4	173.5	8.54	1,505	149	1,654	4	3.1 x 10^3
B&W 15x15	167.4	173.5	8.54	1,515	165	1,680	5,435	1.0 x 10^4
CE 16x16	178.6	190.8	8.14	1,430	72	1,502	2,340	8.1 x 10^3
CE System 80	180	194.8	8.16	1,430	N/A	1,430	1,132	8.1 x 10^3
South Texas	201.1	201.1	8.4	1,720	200	1,920	424	3.0 x 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 NG) 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, including any internal structures, shall be designed to include sufficient shielding to protect the waste package materials (in the as-emplaced condition) from radiation enhanced corrosion.

[F 1.1.1, 1.1.2]

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, 3.2.L][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 (6.35 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)]

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]

1.2.1.22 Manufacturing residual tensile stresses shall be maintained below 10 percent of yield strength of the outer barrier material for a depth of (TBD-235) from the outer surface.

[F 1.1.2, 1.1.6][MGR RD 3.1.G]

1.2.1.23 Static loads in the outer barrier shall not produce tensile stresses above 10 percent of yield strength of the outer barrier material at the interface with the emplacement pallet.

[F 1.1.2, 1.1.6][MGR RD 3.1.G]

1.2.1.24 Tensile stresses in the outer barrier shall be maintained below yield strength of the outer barrier material during a seismic event.

[F 1.1.2, 1.1.6]

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 6 MT (13,230 lb) (TBV-245) rock falling 3.3 m (10.8 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 stress resulting from a drop of the waste package with the emplacement pallet from a height of (TBD-235) onto an essentially unyielding surface 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 Frequency Category 2 Design Basis Earthquake during the preclosure period (TBV-1246). 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. (TBV-245)

[F 1.1.1, 1.1.6, 1.1.9][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 disposal container/waste package shall be designed to withstand the hypothetical fire criteria defined in 10 CFR 71 ("Packaging and Transportation of Radioactive Material"), 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 the effective multiplication factor (k_{eff}) is less than or equal to 0.95 under assumed accident conditions considering 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, the waste package shall be designed such that the total frequency of criticality occurrence is less than 1 in 10,000 years for all waste packages. (TBV-096)

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

1.2.2.1.14 During the postclosure period, any increase in radionuclide inventory due to a single criticality event from fissionable material emplaced in the waste package shall be less than 10 percent of the total radionuclide inventory available for release and transport for all waste packages to the accessible environment.

[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 3 (TBV-4743) (TBD-234) and the induced/handling external environments identified in Table 4 (TBD-276).

Table 3. Emplacement Drift External Environment

Environment	Range	Duration/Frequency of Occurrence
Microbe Influx	0–10^{14} microbes/yr/m of drift (TBV-4743)	10,000 yr
pH	7 - 11	10,000 yr
Colloid Concentration	$8 \times 10^{-6} – 6 \times 10^{-5}$ mg/ml (TBV-4743)	10,000 yr
Temperature	TBD-234	TBD-234
Humidity	0 - 95%	10,000 yr
Radiation	TBD-234	TBD-234
Water Seepage	4.6 mm/yr 12.2 mm/yr 17.8 mm/yr	To year 2,600 From year 2,600 to year 3,000 From year 3,000 to year 12,000
TBD-234	TBD-234	TBD-234

Table 4. 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 Waste package design shall reduce the dose rate at all external surfaces of a waste package to 1,450 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]

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

[F 1.1.4]

1.2.4.3 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 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]

1.2.4.4 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]

1.2.4.5 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]

1.2.4.6 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 Criticality 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 will be provided in a future revision.

2. DESIGN DESCRIPTION

Section 2 of this SDD summarizes information that is contained in other references. By assembling system specific information contained elsewhere (i.e., analyses, technical reports, etc.), Section 2 provides insight into the current state of the design of this system. However, due to the nature of design development, the information contained in this section will continue to change as the design matures.

2.1 SYSTEM DESIGN SUMMARY

The Uncanistered Spent Nuclear Fuel (UCF) disposal containers are designed to accept PWR and BWR fuel assemblies, and individually canistered SNF assemblies. There are currently five UCF disposal container designs (see note below):

1) 21-PWR Absorber Plate
2) 21-PWR Control Rod
3) 12-PWR Long
4) 44-BWR
5) 24-BWR

The UCF disposal containers are essentially right-circular cylinders, hereafter called shells. They are comprised of two shells, an inner shell of stainless steel that provides structural support, and an outer shell of high-nickel alloy that provides a corrosion-resistant barrier. The inner structural shell is inserted inside the outer corrosion-resistant shell. There are two lower lids that are welded to the shells at the time of fabrication. There are three upper lids that are welded in place after the disposal containers are loaded with the appropriate waste forms (reference "Design Analysis for UCF Waste Packages," Section 6).

The commercial spent nuclear fuel (CSNF) assemblies are loaded into baskets that form a regular array of square apertures. For the 21-PWR Absorber Plate and 44-BWR disposal container designs, these baskets are formed from interlocking plates of neutron-absorbing stainless steel material and aluminum, and are held in place by basket stiffeners and guides. In addition, each basket contains a structural steel tube into which the fuel assembly is loaded (reference "Design Analysis for UCF Waste Packages," Section 6).

NOTE: For Site Recommendation, only a subset of the detailed designs has been explicitly evaluated to ensure design criteria are met. Through a sensitivity analysis, it was decided which disposal container designs best represent the array of design configurations and waste forms. The analysis also considered which of the design criteria would be satisfied by those representative designs. The analysis resulted in the selection of four disposal container designs (reference "Waste Package Design

Sensitivity Report," Section 5) as adequate to jointly demonstrate compliance with design criteria for Site Recommendation:

- 21-PWR Absorber Plate
- 44-BWR
- 5-defense high-level waste (DHLW)/U.S. Department of Energy (DOE) SNF short
- naval SNF Long

For criteria that apply to multiple designs, a decision may be made to evaluate just one of the four designs to demonstrate compliance (reference "Waste Package Design Sensitivity Report," Section 6). When the criterion is specific to UCF designs, a decision may be made whether to evaluate the 21-PWR Absorber Plate disposal container or 44-BWR disposal container. The design and design criteria selected for Site Recommendation is provided in Section 2.5. More complete design information will be provided for all waste package designs as a part of the licensing process.

2.2 DESIGN ASSUMPTIONS

This section lists the assumptions used in the designs of the 21-PWR Absorber Plate disposal container and the 44-BWR disposal container that are provided for information only. Rationale statements are provided in the document referenced after each assumption.

2.2.1 Source Term

2.2.1.1 Use of Single PWR Fuel Assembly Design

The various PWR fuel assembly types can be approximated by a single assembly design and that the resulting source terms are negligibly affected by the use of a common geometry. The fuel assembly design used in the generation of the source terms is the Babcock and Wilcox (B&W) Mark B 15x15 fuel assembly (reference "Design Analysis for UCF Waste Packages," Section 5).

2.2.1.2 Use of a Single BWR Fuel Assembly Design

It is assumed that the various BWR fuel assembly types can be approximated by a single assembly design and that the resulting source terms are negligibly affected by the use of a common geometry. The fuel assembly design used in the generation of the source terms is the General Electric BWR/2,3 8x8 fuel assembly (reference "Design Analysis for UCF Waste Packages," Section 5).

2.2.2 Structural

2.2.2.1 21-PWR as Conservative for Pressurization Design-Basis Event

The internal pressure for the 21-PWR Absorber Plate waste package is conservatively assumed for the 44-BWR waste package (reference "Design Analysis for UCF Waste Packages," Section 5).

2.2.2.2 Material Property Temperatures for 21-PWR Slap Down Evaluation

Material properties at room temperature (20 degrees C) are assumed for in the absence of more appropriate data. Otherwise, material properties evaluated at 93 degrees C (200 degrees F) are used when available (reference "Design Analysis for UCF Waste Packages," Section 5).

2.2.2.3 Friction Coefficients

The coefficients of static and dynamic friction for steel on steel are used, instead of those for rock on nickel alloy, or nickel alloy on rock, which are not found in available literature (reference "Design Analysis for UCF Waste Packages," Section 5).

2.2.3 Thermal

2.2.3.1 Stratigraphic Thermal Transport Properties

The stratigraphy of the major geologic units near the center of the repository and the corresponding thermal transport properties of the rock comprising those strata were provided by the Natural Environment Program Operations (reference "Design Analysis for UCF Waste Packages," Section 5).

2.2.4 Shielding

2.2.4.1 Bounding CSNF Characteristics

It is assumed that the PWR and BWR SNF radiation source terms have the following characteristics: 5.5 wt% initial ^{235}U, 75.0 GWd/MTU burnup, and 5-year decay time for the active fuel region; and 0.711 wt% initial ^{235}U, 75.0 GWd/MTU burnup, and 5-year decay time for the hardware regions (reference "Design Analysis for UCF Waste Packages," Section 5).

2.2.4.2 Value for Axial Peaking Factor

The axial peaking factor for a PWR SNF assembly is 1.25. The axial peaking factor for a BWR SNF assembly is 1.4. Since the radiation source terms are generated with the assumption that the burnup is uniformly distributed within an

SNF assembly, an axial peaking factor is used for neutron and photon source strengths in the active fuel region (reference "Design Analysis for UCF Waste Packages," Section 5).

2.2.5. Criticality

2.2.5.1 Selection of PWR Fuel Assembly

The B&W 15x15 MK-B2 assembly design is the most limiting PWR fuel assembly design (reference "Design Analysis for UCF Waste Packages," Section 5).

2.2.5.2 Maximum Soluble Poison Concentration for PWR Fuel Assembly Depletion

The soluble poison concentration used for PWR fuel depletion is 946 ppm, which is the highest cycle average used in Crystal River Unit 3 (reference "Design Analysis for UCF Waste Packages," Section 5).

2.2.5.3 Upper Sub-critical Limit for Preclosure Period

An upper subcritical limit of 0.929 is used for the preclosure time period. This value provides a 5 percent margin in addition to 2.1 percent for bias and uncertainty in the method of calculation (reference "Design Analysis for UCF Waste Packages," Section 5).

2.2.5.4 Critical Limit for Postclosure Period

A critical limit ranging from 0.988 to 0.979 over the burnup range from 0.0 to 33.1 GWd/MTU was used for the postclosure time period (reference "Design Analysis for UCF Waste Packages," Section 5).

2.2.5.5 Settled Oxide Depth

It is assumed that the settled iron oxide configuration, with a 58 percent volume free space and intact fuel assembly arrays, is the most limiting fully degraded configuration (reference "Design Analysis for UCF Waste Packages," Section 5).

2.2.5.6 Composition of Degraded Basket Corrosion Product Mixture

The degraded basket corrosion product mixture is similar to that listed in "Supplemental Criticality Evaluation for Degraded Internal Configurations of a 21 PWR Waste Package" (reference "Design Analysis for UCF Waste Packages," Section 5).

2.3 DETAILED DESIGN DESCRIPTION

The UCF disposal containers are essentially right-circular cylinders, hereafter called shells. They are comprised of two shells, an inner shell of stainless steel that provides structural support and an outer shell of high-nickel alloy that provides a corrosion-resistant barrier to form a loosely fitting structure. The inner structural shell is inserted inside the outer corrosion-resistant shell. There are two lower lids that are welded to the shells at the time of fabrication. There are three upper lids that are welded in place after the waste packages are loaded with the appropriate waste forms.

The commercial spent nuclear fuel (CSNF) assemblies are loaded into baskets that form a regular array of square apertures. For the 21-PWR Absorber Plate and 44-BWR disposal container designs, these baskets are formed from interlocking plates of neutron-absorbing stainless steel material and aluminum, and are held in place by basket stiffeners and guides. In addition, each basket contains a structural steel tube into which the fuel assembly is loaded. Figure 1 is a view of the 21-PWR Absorber Plate disposal container configuration. Figure 2 is a view of the 44-BWR disposal container configuration. Figure 3 is a view of the upper lids closure welds.

The 21-PWR Absorber Plate disposal container design will accommodate twenty-one commercial PWR assemblies, containing an average of 0.431 MTU/assembly. The 21-PWR Absorber Plate design is the most common waste package making up approximately 5,690 of the waste packages (reference "Waste Packages and Source Terms for the Commercial 1999 Design Basis Waste Streams," Section 6). The 21 fuel tube openings are 226.4 mm x 226.4 mm each. The cavity is 4,585 mm long and the basket length is 4,575 mm. Additional waste package properties and waste form characteristics are shown in Tables 5 and 6, respectively (reference "Design Analysis for UCF Waste Packages," Attachment I).

Table 5. 21-PWR Absorber Plate Disposal Container/Waste Package Dimensions

Outer Diameter	1.564 m
Outer Length	5.165 m
Empty Mass	26,035 kg
Loaded Mass	42,277 kg

Reference: "Design Analysis for UCF Waste Packages," Attachment I

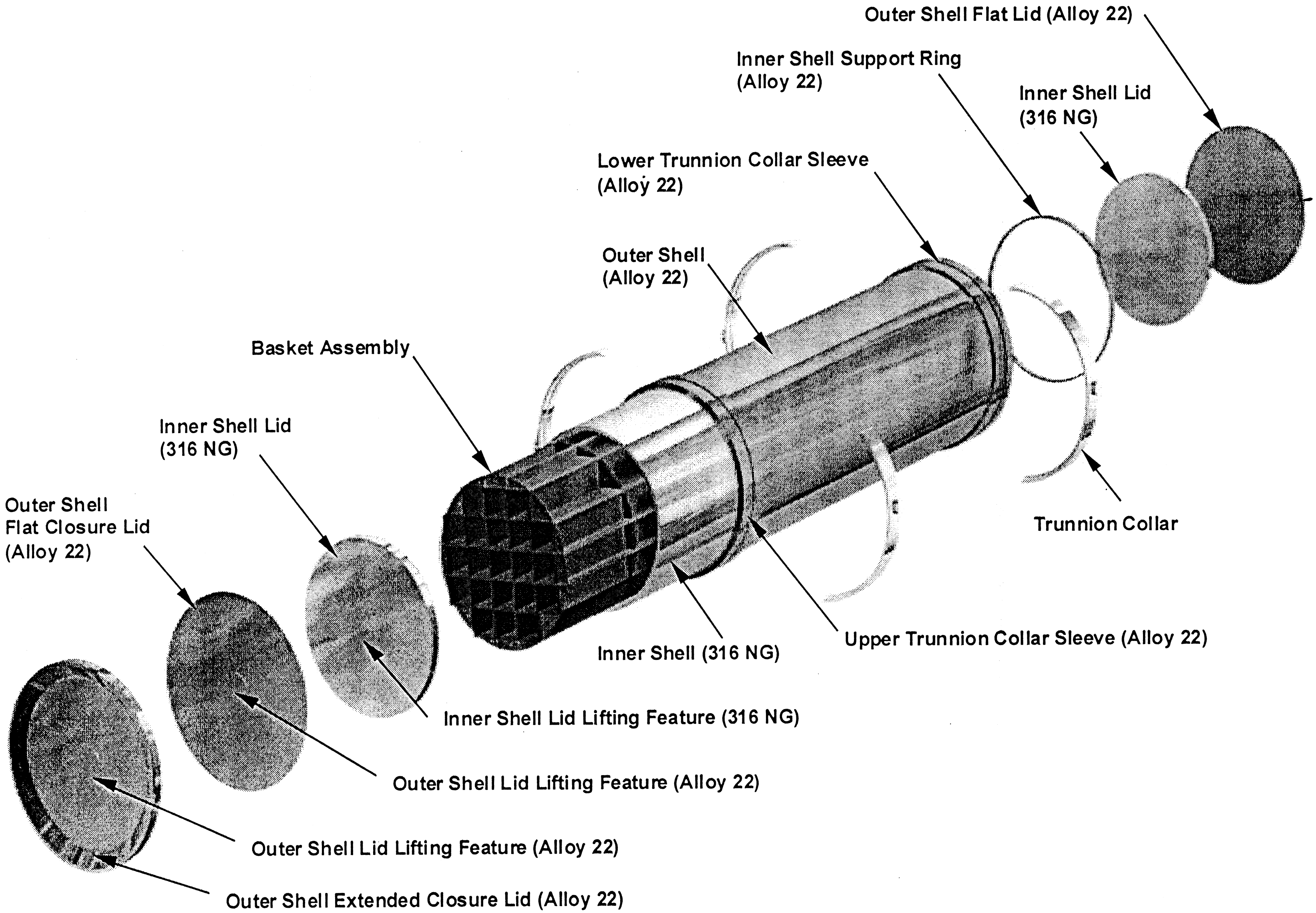

Figure 1. 21-PWR Absorber Plate Disposal Container Configuration

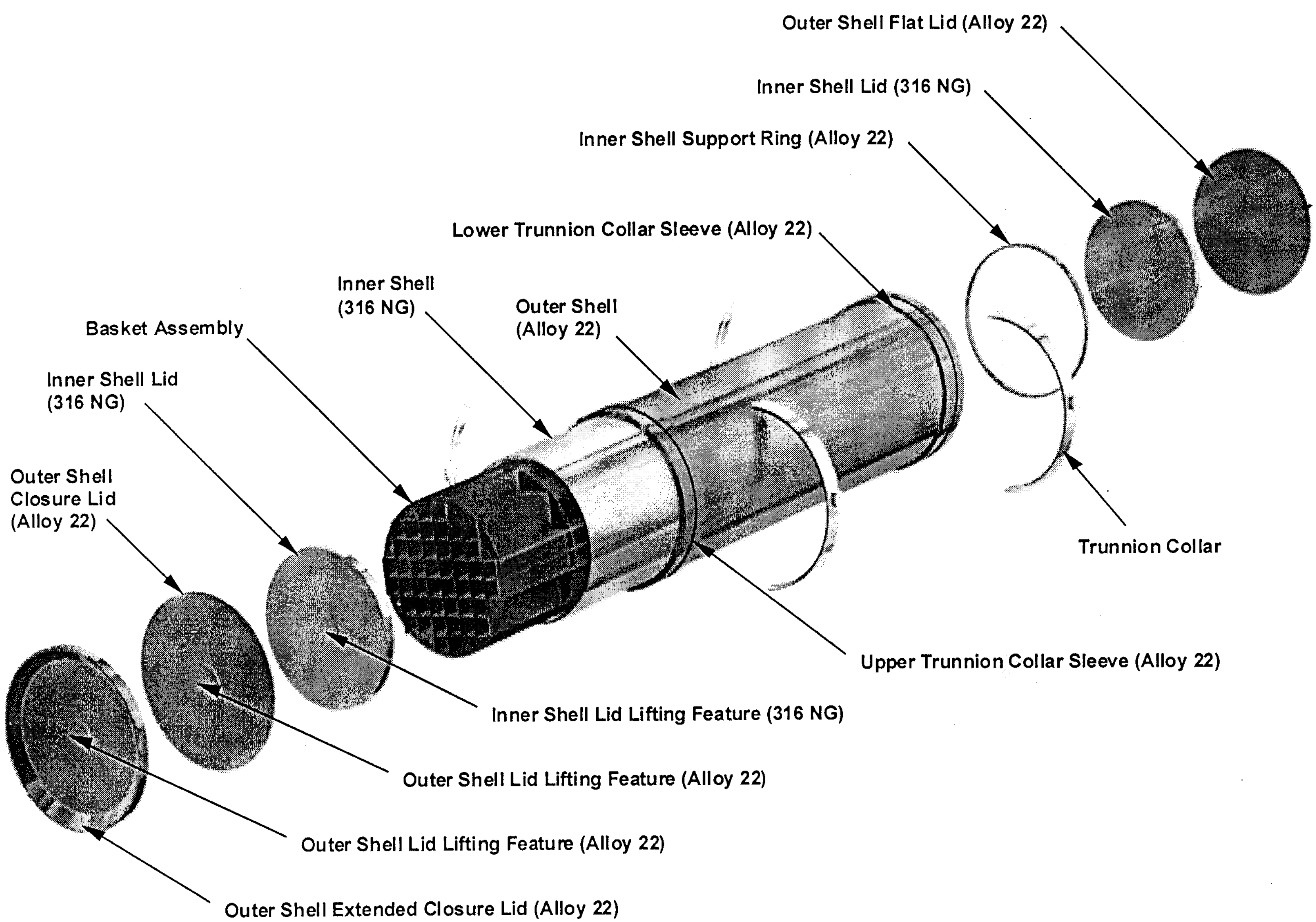

Figure 2. 44-BWR Disposal Container Configuration

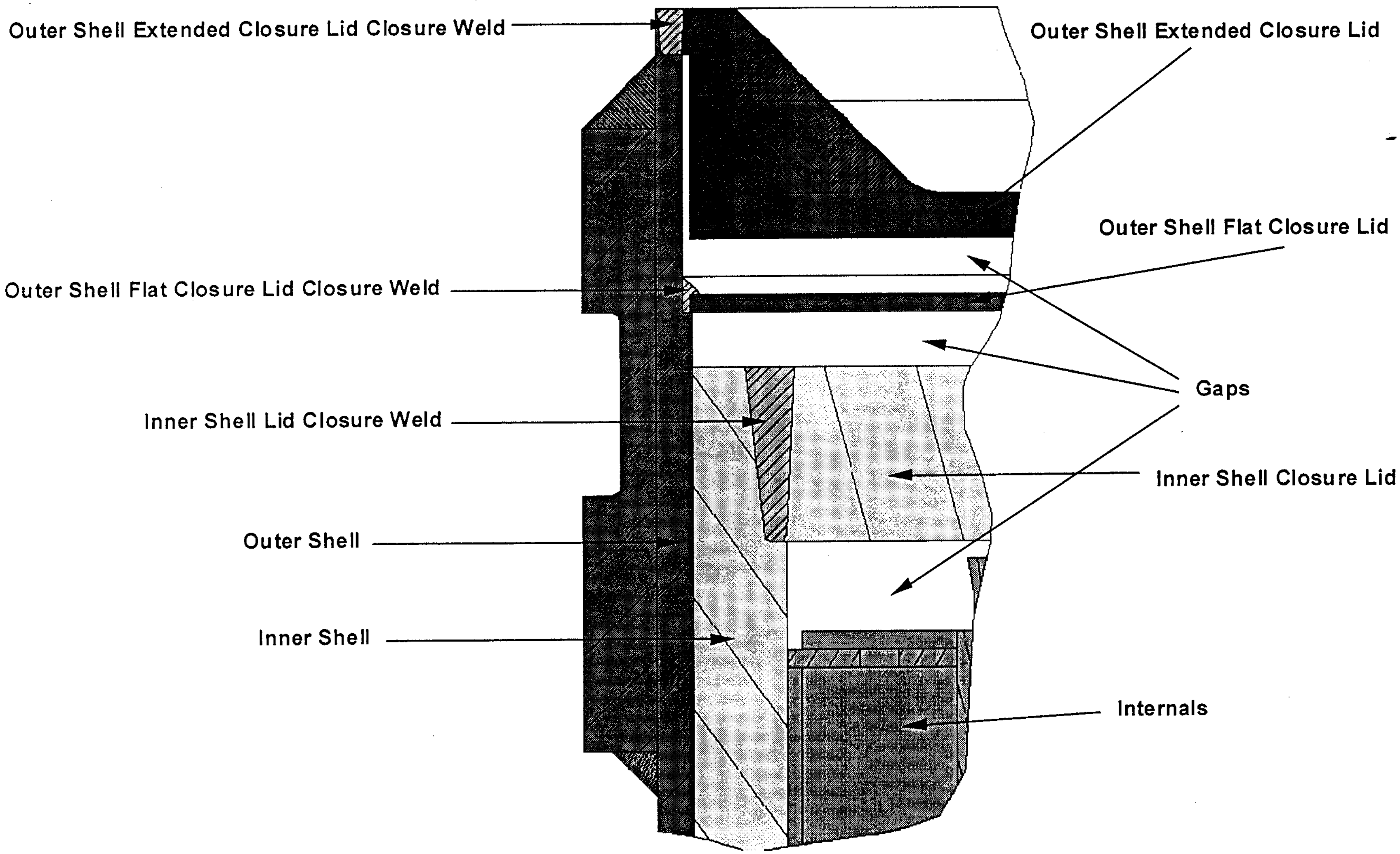

Figure 3. Closure Welds Configuration

Table 6. Waste Form Characteristics for the 21-PWR Absorber Plate Disposal Container/Waste Package

Assembly Maximum Outer Width (Note 1)	216.9 mm
Assembly Maximum External Length (Note 1)	4,572 mm
Assembly Maximum Mass (Note 1)	773.4 kg
Average WP Heat Generation Rate Based on Assembly Heat at Repository Arrival (Note 2)	11.33 kW
Number of Assemblies (Note 2)	119,480
Number of WPs (Note 2)	5,690
Average Burnup at Time of Loading (Note 2)	41.976 GWd/MTU
Average Initial U-235 Enrichment (Note 2)	3.75
Average MTU/Assembly (Note 2)	0.431
Average Assembly Age (Note 2)	25.2 years
Total MTU (Note 2)	51,529.19

Note 1: Reference "Waste Container Cavity Size Determination," pp. 23 and 24

Note 2: Reference "Waste Packages and Source Terms for the Commercial 1999 Design Basis Waste Streams," Table 11; Case A: 83,800 MTU WP Characteristics

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

The 44-BWR is designed to hold forty-four commercial BWR assemblies, containing an average of 0.178 MTU/Assembly. The 44-BWR design is the second most common waste package for the commercial assemblies making up approximately 3,732 of the waste packages needed (reference "Waste Packages and Source Terms for the Commercial 1999 Design Basis Waste Streams," Section 6). The 44 fuel tube openings are 155.3 mm x 155.3 mm each. The cavity is 4,585 mm long and has an inner diameter of 1,454 mm. The basket length is 4,575 mm. Additional 44-BWR waste package properties and characteristics are shown in Tables 7 and 8, respectively (reference "Design Analysis for UCF Waste Packages," Attachment II).

Table 7. 44-BWR Disposal Container/Waste Package Dimensions

Outer Diameter	1.594 m
Outer Length	5.165 m
Empty Mass	28,068 kg
Loaded Mass	42,517 kg

Reference: "Design Analysis for UCF Waste Packages," Attachment II)

Table 8. 44-BWR Disposal Container/Waste Package Characteristics

Assembly Maximum Outer Width (Note 1)	150.3 mm
Assembly Maximum External Length (Note 1)	4,534 mm
Assembly Maximum Mass (Note 1)	328.4 kg
Average WP Heat Generation Rate Based on Assembly Heat at Repository Arrival (Note 2)	7.00 kW
Number of Assemblies (Note 2)	164,165
Number of WPs (Note 2)	3,732
Average Burnup at Time of Loading (Note 2)	33.981 GWd/MTU
Average Initial U-235 Enrichment (Note 2)	3.02
Average MTU/Assembly (Note 2)	0.178
Average Assembly Age (Note 2)	25.08 years
Total MTU (Note 2)	29,140.39

Note 1: Reference "Waste Container Cavity Size Determination," p. 24
Note 2: Reference "Waste Packages and Source Terms for the Commercial 1999 Design Basis Waste Streams," Table 11; Case A: 83,800 MTU WP Characteristics
For definition of acronyms, symbols and units, see Appendix C.

2.4 COMPONENT DESCRIPTION

2.4.1 Internal Components

The UCF disposal containers have an internal basket assembly to support the fuel assemblies. The basket assembly is composed of absorber plates, fuel tubes, thermal shunts, basket stiffeners, and guides. Internal components materials of construction, dimensions, and quantities for the 21-PWR Absorber Plate and 44-BWR disposal container designs are provided in Table 9 and Table 10, respectively. The five basket assembly components, their functions, and descriptions are described below.

Table 9. 21-PWR Absorber Plate Disposal Container/Waste Package Components

Component Name	Material	Thickness	Quantity
Basket A-Sideguide	SA-516 K02700	10 mm	32
Basket A-Stiffener	SA-516 K02700	10 mm	64
Basket B-Sideguide	SA-516 K02700	10 mm	16
Basket B-Stiffener	SA-516 K02700	10 mm	32
Basket C-Stiffener	SA-516 K02700	10 mm	32
Basket Cornerguide	SA-516 K02700	10 mm	16
Fuel Basket A-Plate (Absorber Plate)	Neutronit A 978	7 mm	8
Fuel Basket B-Plate (Absorber Plate)	Neutronit A 978	7 mm	8
Fuel Basket C-Plate (Absorber Plate)	Neutronit A 978	7 mm	16
Fuel Basket D-Plate (Thermal Shunt)	SB-209 A96061 T4	5 mm	8
Fuel Basket E-Plate (Thermal Shunt)	SB-209 A96061 T4	5 mm	8
Fuel Basket Tube	SA-516 K02700	5 mm	21

Reference "Design Analysis for UCF Waste Packages," Attachment I

Table 10. 44-BWR Disposal Container/Waste Package Components

Component Name	Material	Thickness	Quantity
Basket B-Sideguide	SA-516 K02700	10 mm	32
Basket B-Stiffener	SA-516 K02700	10 mm	64
Basket Cornerguide	SA-516 K02700	10 mm	32
Basket Stiffener	SA-516 K02700	10 mm	64
Fuel Basket A-Plate (Absorber Plate)	Neutronit A 978	5 mm	4
Fuel Basket B-Plate (Absorber Plate)	Neutronit A 978	5 mm	4
Fuel Basket C-Plate (Absorber Plate)	Neutronit A 978	5 mm	16
Fuel Basket D-Plate (Absorber Plate)	Neutronit A 978	5 mm	16
Fuel Basket E-Plate (Absorber Plate)	Neutronit A 978	5 mm	16
Fuel Basket F-Plate (Thermal Shunt)	SB-209 A96061 T4	5 mm	8
Fuel Basket G-Plate (Thermal Shunt)	SB-209 A96061 T4	5 mm	8
Fuel Basket Tube	SA-516 K02700	5 mm	44

Reference "Design Analysis for UCF Waste Packages," Attachment II

2.4.1.1 Absorber Plates

The absorber plates, in addition to the thermal shunts (if applicable), pattern the arrangement of fuel assemblies inside the waste package. The basket assembly is customized to meet requirements for the size, type, and number of fuel assemblies the disposal container must hold, and to meet waste form requirements in doing so. The material composition and thickness of the absorber plates are tailored to provide fuel-basket structural strength to maintain fuel geometry, and to prevent criticality. With the exception of the 21-PWR Control Rod disposal container design, the absorber plates for the UCF disposal container designs are made of a stainless steel and boron alloy.

The most important function of the absorber plate is to reduce the potential for criticality. The neutron absorber material (boron) is an additive to the carrier material of the absorber plates that make the basket assembly. Corrosion behavior is important in keeping the neutron absorber material effectively in place to prevent criticality long after emplacement; thus, the chemical performance of the material in a variety of environments was an important criterion for material selection. The stainless steel-boron plates corrode slowly, typically by pitting. Because of this, the plates remain in place between the fuel assemblies even as they corrode.

Mechanical performance is an evaluation factor because the absorber plates must be able to sustain mechanical loads from handling, emplacement, and, if necessary, retrieval. To avoid the effects of processes that could accelerate stress–corrosion cracking, there will be no bends or structural welds on the stainless steel-boron plates.

The material of choice for the neutron absorber is Neutronit A978. The selection is based on the corrosion performance of this material compared to the other candidate materials and available boron concentration. The composition of Neutronit A978 (stainless steel-boron) is similar to ASTM A 240 Type 316 stainless steel, but with 1.6 percent boron (reference "Waste Package Neutron Absorber, Thermal Shunt, and Fill Gas Selection Report," Section 3.1.3).

2.4.1.2 Fuel Basket Tubes

The fuel basket tubes are long, square tubes that fit inside the baskets of the waste package. In addition to holding the fuel assemblies in place, the tubes structurally support the basket assembly. Thus, mechanical performance is the major criterion for material selection.

If the material selected for the tubes causes degradation of the waste form, release rates could be increased, and, if it causes degradation of the plates, criticality control could be compromised. Therefore, compatibility with other materials was selected as an important criterion.

The design does not rely on this component for postclosure performance, so a higher-cost corrosion-resistant material is not needed. ASTM A 516 Grade 70 (SA-516 K02700) was chosen based on mechanical performance (reference "Waste Package Internal Materials Selection Report").

2.4.1.3 Thermal Shunts

With the exception of the 24-BWR disposal container design, the UCF disposal container designs require thermal shunts. The shunts are placed alongside the absorber plates and are made of aluminum. Adding thermal shunts is a simple and effective method to significantly improve heat conduction between the center of the waste package and the outer edge of the basket, thereby providing a reliable means to keep the temperature of the fuel cladding within limits. Limiting fuel cladding temperatures helps protect the waste form by minimizing damage to the fuel cladding material.

Since thermal shunts provide an important additional path for conducting heat from commercial SNF to the walls of the waste package, thermal conductivity of the thermal shunt material is very important. The thermal shunts are needed only during the early periods of repository performance, when the decay heat of the SNF is high. Therefore, the material does not need to have a high degree of corrosion resistance.

The thermal shunts must have enough structural strength to withstand handling, emplacement and retrieval operations. However, this design component is not intended to maintain integrity of the basket assembly as are the absorber plates.

Aluminum alloy (SB-209 A96061 T4) is selected over copper because of concerns that, should a waste package be breached and water enter, copper may react with the chloride ions in the water, which could result in accelerated degradation of the zirconium-based cladding on the fuel, which would then eventually release radionuclides from the waste form (reference "Waste Package Neutron Absorber, Thermal Shunt, and Fill Gas Selection Report," Section 3.2.3).

2.4.1.4 Basket Guides and Stiffeners

The basket guides and stiffeners are used to maintain structural integrity of the waste package basket assembly (reference "Waste Package Internal Materials Selection Report," p. 2). The basket guides and stiffeners are 10 mm (0.39 in.) thick and made of ASTM 516 Grade 70 Carbon Steel (SA-516 K02700). The design does not rely on these components for postclosure performance, so higher-cost corrosion-resistant materials are not needed. ASTM A 516 Grade 70 was found to be the best choice based on the criteria (reference "Waste Package Internal Materials Selection Report," p.13).

2.4.2 External Components

2.4.2.1 Lids

The waste package lid configuration consists of three lids on the top of the waste package and two lids on the bottom of the waste package. Lid materials and dimensions are provided in Tables 11 and 12.

Table 11. 21-PWR Disposal Container/Waste Package Lids

Lid	Material	Thickness
Extended Outer Shell Lid (Top)	SB-575 N06022	25 mm
Outer Shell Flat Closure Lid (Top)	SB-575 N06022	10 mm
Inner Lid (Top)	SA-240 S31600	95 mm
Inner Lid (Bottom)	SA-240 S31600	95 mm
Outer Shell Flat Bottom Lid (Bottom)	SB-575 N06022	25 mm

Reference "Design Analysis for UCF Waste Packages," Attachment I

Table 12. 44-BWR Disposal Container/Waste Package Lids

Lid	Material	Thickness
Extended Outer Shell Lid (Top)	SB-575 N06022	25 mm
Outer Shell Flat Closure Lid (Top)	SB-575 N06022	10 mm
Inner Lid (Top)	SA-240 S31600	95 mm
Inner Lid (Bottom)	SA-240 S31600	95 mm
Outer Shell Flat Bottom Lid (Bottom)	SB-575 N06022	25 mm

Reference "Design Analysis for UCF Waste Packages," Attachment II

2.4.2.2 Lifting Features

The disposal containers/waste packages are designed to accept trunnion rings (see Figure 4). The use of trunnion rings permits attachments of fixtures that may be used for both vertical and horizontal handling of the waste package, as well as orientations between vertical and horizontal.

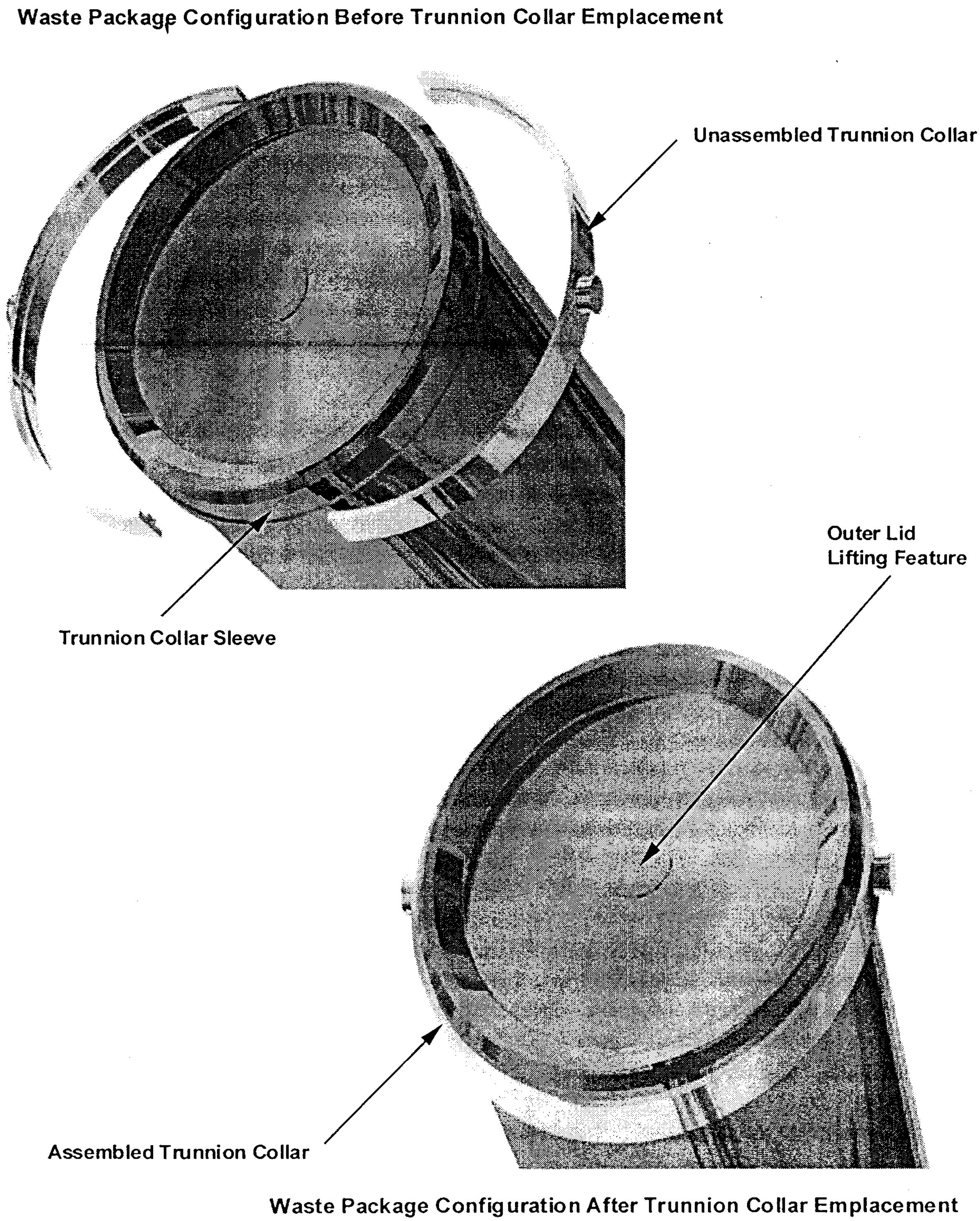

Figure 4. Trunnion Collar Configuration

Trunnion rings are removed after the waste package is placed on the emplacement pallet; therefore, the use of trunnion rings does not create a site for cracking due to crevice corrosion. Furthermore, trunnion rings are attached to a corresponding built-up area on the waste package (trunnion collar sleeve) and will not induce stresses that might exacerbate corrosion of the outer barrier. Materials of construction and dimensions for the trunnion collar sleeves are provided in Table 13.

Table 13. UCF Waste Package Trunnion Collar Sleeves

Component Name	Material	Thickness	Quantity
Upper Trunnion Collar Sleeve	SB-575 N06022	40 mm	1
Lower Trunnion Collar Sleeve	SB-575 N06022	40 mm	1

Reference "Design Analysis for UCF Waste Packages," Attachments I and II

2.4.3 Fill Gas

Fill gas is used to provide an inert atmosphere within the waste package and to facilitate heat conduction from the waste form to the waste package barriers. The fill gas can be a significant conductor of heat from the fuel to the basket, so thermal performance was deemed to be one of the most important criterion in choosing a gas. The fill gas should not degrade other components of the waste package, so compatibility with other materials was another important criterion. Helium is used routinely as the fill gas for the fuel rods; this standard practice indicates helium would have excellent compatibility with the SNF. Based on a review of data on thermal conductivity, and the fact that helium is chemically inert, it was chosen over other candidate gases, such as nitrogen, argon, and krypton (reference "Waste Package Neutron Absorber, Thermal Shunt, and Fill Gas Selection Report," Sections 3.3.1 through 3.3.3). The waste package will be filled with helium to approximately one atmosphere of pressure.

2.5 CRITERIA COMPLIANCE

This section contains the demonstration that the Site Recommendation UCF waste package design satisfies criteria from Section 1. It is limited to criteria that were selected to be demonstrated by either the 21-PWR Absorber Plate disposal container design or the 44-BWR disposal container design, and criteria determined to be necessary for Site Recommendation. The remaining criteria are either demonstrated by the other waste package designs, or will not be demonstrated until License Application (reference "Design Analysis for UCF Waste Packages," Section 6).

2.5.1 System Performance Criteria

2.5.1.1 Accommodation of Intact Fuel Assemblies (Criterion 1.2.1.1)

The two commercial fuel assembly types that the waste package must accommodate are PWR and BWR. The envelope of requirements for the limiting fuel assembly designs and the corresponding available volume within the waste package basket voids are shown in the Table 14. Compliance is demonstrated for this criterion (reference "Design Analysis for UCF Waste Packages," Section 6).

Table 14. Accommodation of Limiting Fuel Assemblies

Limiting Assembly Dimensions [a]			Basket Void Dimensions [c]	
Assembly Type	Length	Width	Length	Width
PWR	180 in. (457.2 cm)	8.54 in. (21.7 cm)	458.5 cm	22.64 cm
BWR	177.8 in. (451.6 cm)	5.91 in. (15.0 cm) [b]	458.5 cm	15.53 cm

[a] These are irradiated dimensions.
[b] These dimensions include the channels.
[c] Reference "Design Analysis for UCF Waste Packages," Attachments I and II

2.5.1.2 Accommodation of Sealed, Disposable, Single-element SNF Canisters (Criterion 1.2.1.2)

Canisters will be inserted into a waste package designed to accommodate fuel assemblies from the South Texas plant. This design will not be developed for Site Recommendation (reference "Design Analysis for UCF Waste Packages," Section 7).

2.5.1.3 Limiting Total Effective Dose (Criterion 1.2.1.3)

Satisfaction of this criterion will be demonstrated in a report to be prepared by the Performance Assessment Organization (reference "Design Analysis for UCF Waste Packages," Section 6).

2.5.1.4 Thickness and Composition of Cylinders (Criterion 1.2.1.4)

It is required that the inner cylinder of the waste package be constructed of stainless steel 316 NG. The inner cylinder is comprised of SA-240 S31600, which is the ASME equivalent of ASTM SS-316 (reference "Design Analysis for UCF Waste Packages," Attachments I and II). The SDD requires that the outer cylinder be constructed of alloy 22. The outer cylinder is comprised of SB-575 N06022, which is the ASME equivalent of ASTM Alloy 22 (reference "Design Analysis for UCF Waste Packages," Attachments I and II)

The required thickness of the waste package shells is dictated by a number of considerations including long-term performance as embodied in the Total Effective Dose, and reliability demonstrated by the survival against mechanical challenges to the waste packages. The compliance with shell radial dimension requirements is shown in Table 15 (reference "Design Analysis for UCF Waste Packages," Section 6).

Table 15. Compliance of Radial Waste Package Shell Dimensions

SDD Requirements		Shell Radial Dimensions	
Inner Cylinder	Outer Cylinder	Inner Cylinder	Outer Cylinder
5 cm (Nominal)	2 cm (Nominal)	5 cm	2 cm

2.5.1.5 Reserved

2.5.1.6 Peak-Cladding Temperature for Zircaloy-Clad Fuel (Criterion 1.2.1.6)

The current repository thermal design strategy requires an emplacement linear thermal power of 1.45 kW/m and a waste package skirt-to-skirt spacing of 0.1 m. The basic design does not incorporate backfill, but retains the option to install backfill prior to closure. It must be possible to close the repository 26 years after the emplacement of the final waste package. The repository is ventilated and the net ventilation efficiency is 70 percent.

Peak fuel cladding temperatures of the 21-PWR Absorber Plate design-basis waste package are calculated from finite element representations of the waste package internals. Surface temperature boundary conditions of the two-dimensional calculations are provided from drift balanced three-dimensional repository scale evaluations, and the respective fuel assemblies are homogenized with an effective thermal conductivity. Note that the 21-PWR Absorber Plate waste package thermal output is 11.8 kW.

These calculations result in peak fuel cladding temperatures for the 21-PWR Absorber Plate design-basis waste package. The peak temperature for the fuel cladding was calculated to be 282.2 degrees C, with the peak occurring 35 years after emplacement, thus demonstrating compliance (reference "Design Analysis for UCF Waste Packages," Section 6).

Compliance with the 570 degree C limit for short-term exposure to fire will not be demonstrated for Site Recommendation.

2.5.1.7 Prevention of Fuel Breach During Handling (Criterion 1.2.1.7)

This design criterion will not be addressed for Site Recommendation (reference "Design Analysis for UCF Waste Packages," Section 7).

2.5.1.8 Retrieval Contingency Period (Criterion 1.2.1.8)

Since heaviest waste package is the limiting waste package for the purposes of retrieval, the naval SNF waste package rather than a UCF waste package is evaluated (reference "Design Analysis for UCF Waste Packages," Section 6).

2.5.1.9 Limitation on Oxidizers within Waste Package (Criterion 1.2.1.9)

This design criterion will not be addressed for Site Recommendation (reference "Design Analysis for UCF Waste Packages," Section 7).

2.5.1.10 External Surface Roughness (Criterion 1.2.1.10)

This design criterion will not be addressed for Site Recommendation (reference "Design Analysis for UCF Waste Packages," Section 7).

2.5.1.11 Inspectability of Waste Packages (Criterion 1.2.1.11)

Compliance may be demonstrated by inspection of the sketches shown in Attachments I and II of "Design Analysis for UCF Waste Packages," that all surfaces are accessible for visual inspection and decontamination (reference "Design Analysis for UCF Waste Packages," Section 6).

2.5.1.12 Labeling (Criterion 1.2.1.12)

This design criterion will not be addressed for Site Recommendation (reference "Design Analysis for UCF Waste Packages," Section 7).

2.5.1.13 Benign Labeling (Criterion 1.2.1.13)

This design criterion will not be addressed for Site Recommendation (reference "Design Analysis for UCF Waste Packages," Section 7).

2.5.1.14 Legibility of Labeling (Criterion 1.2.1.14)

This design criterion will not be addressed for Site Recommendation (reference "Design Analysis for UCF Waste Packages," Section 7).

2.5.1.15 Accommodation of Filler (Criterion 1.2.1.15)

During the License Application Design Selection effort, incorporation of filler in a 21-PWR Absorber Plate waste package was investigated. This evaluation demonstrated that a filler material, such as iron shot or depleted uranium, can be incorporated into UCF waste packages. Compliance is demonstrated for this criterion (reference "Design Analysis for UCF Waste Packages," Section 6).

2.5.1.16 Capacity of Lifting Devices - Tensile Yield Strength (Criterion 1.2.1.16)

The 21-PWR disposal container is lifted using attachable trunnion rings that are designed to lock into collar sleeves (grooves) on the surface of the waste package. The applicable criterion is for the tensile yield strength of the material to be at least three times the maximum expected lifting stress. The structural response of the waste package to lifting is reported using maximum stress values obtained from the finite element solution to the problem.

Compliance is demonstrated because the tensile yield strength of alloy 22 (310 MPa) is more than three times larger than the maximum expected stress (15.5 MPa) of the alloy 22. Similarly, the tensile yield strength of stainless steel (207 MPa) is more than three times larger than the maximum expected stress (5.5 MPa) of the stainless steel (reference "Design Analysis for UCF Waste Packages," Section 6).

2.5.1.17 Capacity of Lifting Devices - Ultimate Tensile Strength (Criterion 1.2.1.17)

The horizontal and vertical lifting stress characteristics for UCF waste packages are identical to the description provided in Section 2.5.1.16. However, the applicable criterion is for the ultimate tensile strength of the material to be at least five times the maximum expected lifting stress.

Compliance is demonstrated because the ultimate tensile strength of alloy 22 (690 MPa) is more than five times larger than the maximum expected stress (15.5 MPa). Similarly, the ultimate tensile strength of stainless steel (517 MPa) is to be more than five times larger than the maximum expected stress (5.5 MPa) (reference "Design Analysis for UCF Waste Packages," Section 6).

2.5.1.18 Design for Normal Handling Operations (Criterion 1.2.1.18)

In order to withstand transfer, emplacement, and retrieval operations, the waste package is designed to be vertically and horizontally lifted using attachable trunnion rings. Sections 2.5.1.16 and 2.5.1.17 identify the waste package stress intensities that are a result of these operations, and demonstrate compliance to the maximum allowable stress intensities. This criterion is satisfied.

2.5.1.19 Use of Non-combustible and Heat-resistant Materials (Criterion 1.2.1.19)

Compliance may be demonstrated by inspection of the sketches shown in "Design Analysis for UCF Waste Packages," Attachments I and II, that the material call-outs for all components of the waste packages are metallic and, hence, non-combustible. Similarly, the metals selected, which are not necessarily refractory, are resistant to heat (reference "Design Analysis for UCF Waste Packages," Section 6).

2.5.1.20 Avoidance of Explosive or Pyrophoric Materials (Criterion 1.2.1.20)

Compliance may be demonstrated by inspection of the sketches shown in "Design Analysis for UCF Waste Packages," Attachments I and II, that the material call-outs for all components of the waste packages avoid both explosive and pyrophoric materials (reference "Design Analysis for UCF Waste Packages," Section 6).

2.5.1.21 Avoidance of Free Liquids within Waste Package (Criterion 1.2.1.21)

Compliance may be demonstrated by inspection of the sketches shown in "Design Analysis for UCF Waste Packages," Attachments I and II, that the material call-outs for all components of the waste package exclude the use of free liquids (reference "Design Analysis for UCF Waste Packages," Section 6).

2.5.1.22 Manufacturing Residual Tensile Stresses of the Outer Barrier (Criterion 1.2.1.22)

It was shown that a reduction of residual tensile stress to a value less than 10 percent of the yield strength of the outer barrier is possible. This is accomplished by a combination of induction annealing of the closure weld for the outer shell extended lid and laser peening of the closure lid for the outer shell closure lid. This stress was found to be of adequate depth, in Section 3.1.7.6.3 of "Waste Package Degradation Process Model Report," to yield an acceptable lifetime. Compliance is demonstrated for this criterion.

2.5.1.23 Static Loads in the Outer Barrier at the Interface with the Emplacement Pallet (Criterion 1.2.1.23)

The calculation "Tensile Stresses Developing in an Outer Shell of a Waste Package Mounted on an Emplacement Pallet" demonstrates that the maximum stress intensity in the waste package outer barrier Alloy 22 is 17 MPa. This stress intensity is less than 10 percent of the yield strength (310 MPa) of Alloy 22. The criterion is satisfied.

2.5.1.24 Tensile Stresses in the Outer Barrier during a Seismic Event (Criterion 1.2.1.24)

Postclosure seismic events will not be demonstrated for Site Recommendation.

2.5.2 Safety Criteria

2.5.2.1 Preclosure Rock Fall without Breach (Criterion 1.2.2.1.1)

The 21-PWR Absorber Plate disposal container design was selected for this evaluation because it is the most common design, making it the most likely to be hit by a rock. Evaluations represented a rockfall from the roof of the drift onto the unprotected waste package. The rock is assumed to drop from a height of 3.1 m and a design-basis rock size of 13 metric tons is represented. The results of this calculation are summarized in Table 16. Compliance is demonstrated by showing that the maximum calculated stress is less than 90 percent of the ultimate tensile strength of the shell materials (reference "Design Analysis for UCF Waste Packages," Section 6).

Table 16. Summary of Results for Rock Fall Calculation

Shell Composition	Shell Thickness	Calculated Maximum Stress	Ultimate Tensile Stress S_u ($0.9 \cdot S_u$)
Alloy 22	2.0 cm	563 MPa	690 MPa (621 MPa)
Stainless Steel	5.0 cm	291 MPa	517 MPa (465 MPa)

2.5.2.2 Preclosure Impact on End of Waste Package without Breach (Criterion 1.2.2.1.2)

This design criterion will not be addressed for Site Recommendation (reference "Design Analysis for UCF Waste Packages," Section 7).

2.5.2.3 Preclosure Vertical Drop with Breaching (Criterion 1.2.2.1.3)

Since heaviest waste package with greatest internal load is the limiting waste package for the purposes of normal handling operations, the naval SNF waste package rather than a UCF waste package is evaluated (reference "Design Analysis for UCF Waste Packages," Section 6).

2.5.2.4 Preclosure Horizontal Drop without Breaching (Criterion 1.2.2.1.4)

Since heaviest waste package with the greatest resulting load on the shells is the limiting waste package for the purposes of a horizontal drop onto an unyielding surface, the naval SNF waste package rather than a UCF waste package is evaluated (reference "Design Analysis for UCF Waste Packages," Section 6).

2.5.2.5 Preclosure Horizontal Drop without Puncture (Criterion 1.2.2.1.5)

The structural response of the BWR-44 waste package was evaluated for this event. The results of the evaluations are shown in Table 17. The resulting stresses for both shells comprising the waste package are less than 90 percent of the ultimate tensile strength of those materials; therefore, the waste packages will not breach as a result of puncture and compliance to this criterion is demonstrated (reference "Design Analysis for UCF Waste Packages," Section 6).

Table 17. Summary of Results of Puncture Drop Calculation for 44-BWR Waste Package

Waste Package Material	Maximum Stress Intensity	Ultimate Tensile Stress S_U ($0.9 \cdot S_U$)
Alloy 22	354 MPa	690 MPa (621 MPa)
Stainless Steel	274 MPa	517 MPa (465 MPa)
Carbon Steel A516 Grade 70	273 MPa	482 MPa (434 MPa)

2.5.2.6 Preclosure Slap Down without Breaching (Criterion 1.2.2.1.6)

The 21-PWR Absorber Plate disposal container design was used to demonstrate that this criterion has been met. The results are shown in Table 18. The resulting stresses for all components comprising the waste package are less than 90 percent of the ultimate tensile strength of those materials; therefore, the waste package will not breach as a result of the slap-down and compliance to this criterion is demonstrated (reference "Design Analysis for UCF Waste Packages," Section 6).

Table 18. Summary of Results of Slap Down Calculation for 21-PWR Absorber Plate Waste Package

Waste Package Component	Maximum Stress Intensity	Maximum Membrane Stress Intensity	Maximum Membrane Plus Bending Stress Intensity	S_U ($0.9 \cdot S_U$)
Outer shell and lids	552.8 MPa	486.2 MPa	552.8 MPa	690 MPa (621 MPa)
Inner shell and lids	327.0 MPa	286.1 MPa	325.6 MPa	517 MPa (465 MPa)

2.5.2.7 Sustain Design Basis Earthquake (Criterion 1.2.2.1.7)

Since the waste package with the largest diameter, and hence the highest center of gravity, is the limiting waste package for seismic events, the 5-DHLW/DOE SNF waste package rather than a UCF waste package is evaluated (reference "Design Analysis for UCF Waste Packages," Section 6).

2.5.2.8 Sustain Preclosure Missile with Breaching (Criterion 1.2.2.1.8)

The criterion for the waste package to withstand a 0.5-kg missile impact, with a 5.7 m/sec velocity (1 cm in diameter), has been demonstrated by the 21-PWR Absorber Plate and 44-BWR waste package designs. The calculated minimum velocity required to perforate the shell, as a result of such an impact, for both designs, is 322 m/sec. Compliance with this criterion is demonstrated by an empirical comparison. A missile with a velocity of 5.7 m/sec has only 1.8 percent of the necessary impact velocity to compromise the waste package integrity (reference "Design Analysis for UCF Waste Packages," Section 6).

2.5.2.9 Sustain Preclosure Design Basis Transporter Accident (Criterion 1.2.2.1.9)

This design criterion will not be addressed for Site Recommendation (reference "Design Analysis for UCF Waste Packages," Section 7).

2.5.2.10 Preclosure Internal Pressure Limit (Criterion 1.2.2.1.10)

The pressurization of a 21-PWR Absorber Plate waste package, due to the rupture of all of the fuel rods contained in the waste package, was evaluated. The calculation used basic relationships from solid mechanics to determine the maximum stresses in the waste package shells. In this evaluation, the inner lid is assumed to fail before the outer lid; however, no structural credit is assumed for the outer lid. Evaluations were performed over uniform waste package temperature ranging from 20 degrees C to 600 degrees C.

The peak stresses (membrane and bending) at the junction of the shell and lid from these evaluations are shown in Table 19. The ultimate tensile strength of the stainless steel (SA-240 S31600) at the highest temperature (1000 degrees F, which is 538 degrees C) available in the ASME code is 64.4 ksi (443.9 MPa). The resulting stresses for the waste package inner shell are less than 90 percent of the ultimate tensile strength of the stainless steel; therefore, the waste package will not breach as a result of pressurization, and compliance with this criterion is demonstrated (reference "Design Analysis for UCF Waste Packages," Section 6).

Table 19. Summary of Internal Pressurization Calculations

Temperature	21-PWR Total Stress	44-BWR Total Stress
20°C	63 MPa	66 MPa
200°C	102 MPa	107 MPa
400°C	145 MPa	151 MPa
600°C	187 MPa	195 MPa

2.5.2.11 Sustain Fire (Criterion 1.2.2.1.11)

The demonstration of compliance to the 570 degree C temperature limit for short-term exposure to a fire will not be performed for Site Recommendation (reference "Design Analysis for UCF Waste Packages," Section 7).

2.5.2.12 Avoidance of Preclosure Criticality and Margin (Criterion 1.2.2.1.12)

The demonstration of margin to criticality during the preclosure period was calculated using the 21-PWR Absorber Plate waste package design. The evaluation was performed using the methodology described in the "Disposal Criticality Analysis Methodology Topical Report." The margin criteria to criticality for the preclosure period is 0.929 (reference "Design Analysis for UCF Waste Packages," Section 6).

The resulting preclosure loading curve is shown in Figure 5. Any fuel that has a burnup/enrichment pair equal to or above the loading curve line can be loaded unaltered into the 21-PWR Absorber Plate waste package, and criticality will not occur during preclosure. Compliance with this criterion is demonstrated (reference "Design Analysis for UCF Waste Packages," Section 6).

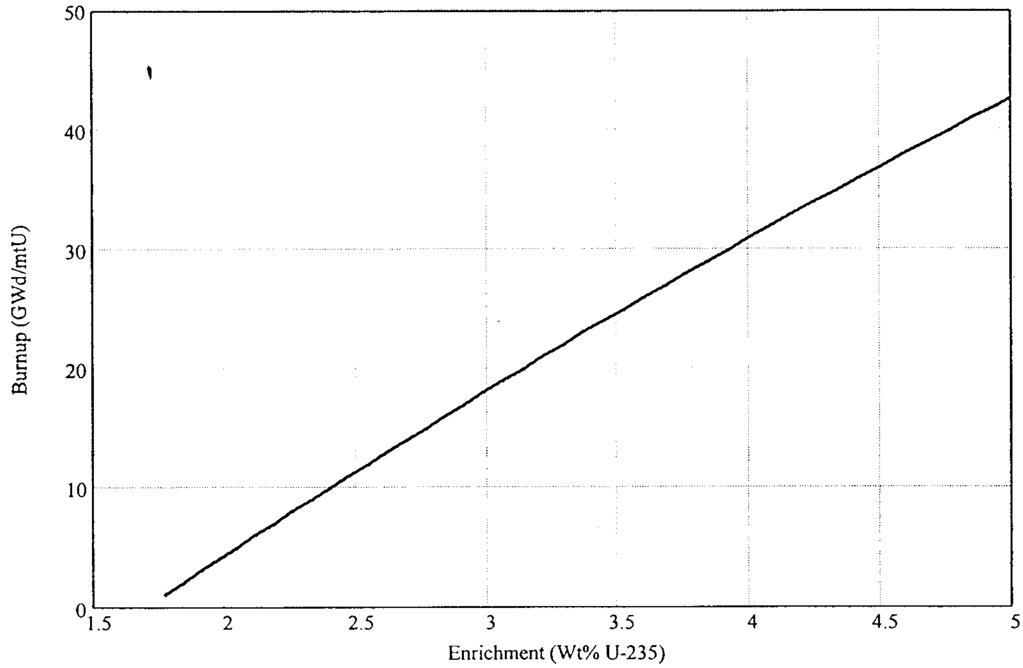

Figure 5. Preclosure Loading Curve for 0.929 Upper Subcritical Limit

2.5.2.13 Probability of Postclosure Criticality Occurrences (Criterion 1.2.2.1.13)

The evaluation methodology used to demonstrate compliance with this criterion is summarized in "Waste Package Design Methodology Report," Section 6.5 and fully described in the "Disposal Criticality Analysis Methodology Topical Report." Compliance with this criterion requires estimating the sum of all probabilities of the frequencies of criticality from all waste forms and waste package designs, and will not be completed prior to License Application.

For Site Recommendation, compliance is demonstrated for commercial SNF primarily as a result of the fact that the postulated time to the earliest breach in the waste package barrier exceeds 10,000 years. This is due to the use of an extremely corrosion-resistant material for the waste package outer barrier and a titanium drip shield covering the tops of the waste packages that is designed to divert water away from the waste packages (reference "Probability of Criticality Before 10,000 Years: Commercial SNF," Section 6).

2.5.2.14 Postclosure Radionuclide Inventory (Criterion 1.2.2.1.14)

The evaluated probability of criticality before 10,000 years for commercial SNF is below the threshold evaluation criteria for TSPA of less than one chance in 10,000 of occurring over 10,000 years. Compliance for Site Recommendation has been demonstrated (reference "Probablility of Criticality Before 10,000 Years: Commercial SNF," Section 6).

2.5.3 System Environment Criteria

2.5.3.1 Emplacement Drift Environment (Criterion 1.2.3.1)

This design criterion will not be addressed for Site Recommendation (reference "Design Analysis for UCF Waste Packages," Section 7).

2.5.4 System Interfacing Criteria

2.5.4.1 Limitation on Surface Radiation Dose Rate (Criterion 1.2.4.1)

The limit on radiation dose rate at the surface of the waste package is 1,450 rem/hr. The second-level confidence interval for the estimate of the maximum dose rates on the external surfaces of the 21-PWR Absorber Plate waste package is 1330±60 rem/hr. This interval for a 44-BWR waste package is 1409±32 rem/hr. Compliance with this criterion is demonstrated (reference "Design Analysis for UCF Waste Packages," Section 6.2.3.1).

2.5.4.2 Maximum Heat-generation Rate (Criterion 1.2.4.2)

The maximum heat-generation rate is imposed by management direction. The waste package thermal design must ensure that such a thermal loading, in conjunction with the balance of the engineered barrier system and natural system, will not result in exceeding the peak fuel-cladding temperature criterion. This is demonstrated in Section 2.5.1.6 (reference "Design Analysis for UCF Waste Packages," Section 6).

2.5.4.3 Limitation on Inventory of CSNF (Criterion 1.2.4.3)

The total inventory of CSNF has no affect on the waste package performance provided that other design criteria for the repository are satisfied (reference "Design Analysis for UCF Waste Packages," Section 6).

2.5.4.4 Vertical Loading of Waste Forms (Criterion 1.2.4.4)

Compliance may be demonstrated by inspection of the sketches shown in "Design Analysis for UCF Waste Packages," Attachments I and II, that the waste package may be loaded in a vertical orientation (reference "Design Analysis for UCF Waste Packages," Section 6).

2.5.4.5 Both Vertical and Horizontal Handling Possible (Criterion 1.2.4.5)

As may be seen by reviewing the sketches shown in Attachments I and II of "Design Analysis for UCF Waste Packages," the waste packages are designed to accept trunnion rings. The use of such rings permits attachments of fixtures that may be used for both vertical and horizontal handling of the waste package, as well as orientations between vertical and horizontal.

These trunnion rings are removed after the waste package is placed on the emplacement pallet; therefore, the use of such rings does not create a site for crevice corrosion cracking. Further, the trunnion rings are attached to a corresponding built-up area on the waste package and will not induce stresses that might exacerbate corrosion of the outer shell. Compliance is demonstrated for this criterion (reference "Design Analysis for UCF Waste Packages," Section 6).

2.5.4.6 Welding Times (Criterion 1.2.4.6)

This design criterion will not be addressed for Site Recommendation (reference "Design Analysis for UCF Waste Packages," Section 7).

2.5.5 Codes and Standards Criteria

2.5.5.1 1995 Boiler and Pressure Vessel Code (Section III, Div. 1, Sub-section NG-1995) (Criterion 1.2.6.1)

There are no codes or standards that apply directly to the design of disposal containers; however, "1995 ASME Boiler and Pressure Vessel Code" has been chosen as a guide for setting stress limits for the waste package components. Applications of subsections of Section III of the code are shown in Table 20 (reference "Design Analysis for UCF Waste Packages," Section 6).

Table 20. Applicability of 1995 ASME Boiler and Pressure Vessel Code

Analysis Type	Component	Section III, Subsection Applied	Service Limits [a]
Static	Shells	Subsection NB	Level A
	Basket	Subsection NG	Level A
Seismic	Shells	Subsection NB, Appendix F	Level D
	Basket	Subsection NG, Appendix F	Level D
Rock Fall	Shells	Subsection NB, Appendix F	Level D
Puncture	Shells	Subsection NB, Appendix F	Level D
	Basket	Subsection NG, Appendix F	Level D
Slap Down	Shells	Subsection NB, Appendix F	Level D
	Basket	Subsection NG, Appendix F	Level D

[a] Level A Service Limits are for normal operation, and Level D Service Limits are for off-normal conditions

As may be seen from this table, Subsection NG of the code is used for operations consistent with normal activities with the Level A Service Limits. From the code, the limitation on membrane and bending stresses at Level A are:

$$P_m + P_b \le 1.5 \cdot S_m = S_y$$

Here, P_m is the membrane stress, P_b is the bending stress and S_m is the design stress intensity for the material. For design purposes, the design stress is assumed to be two-thirds of the yield stress; therefore, the allowable total stress–including both membrane and bending–is equal to the yield stress (S_y) (reference "Design Analysis for UCF Waste Packages," Section 6).

For Level D Service Limits, Sub-section NB of the "1995 ASME Boiler and Pressure Vessel Code" are used, as shown in Table 20. From the code, the limitation on membrane and bending stresses at Level D are:

$$P_m + P_b < 0.9 \cdot S_u$$

Here, S_u is the ultimate tensile strength of the material (reference "Design Analysis for UCF Waste Packages," Section 6).

2.5.5.2 1995 Boiler and Pressure Vessel Code (Section III, Div. 1, Sub-section NB-1995) (Criterion 1.2.6.2)

Sub-section NB of the ASME code is used as shown in Table 20. Level A and Level D service limits are the same as in Section 2.5.5.1. This code was implemented.

2.5.5.3 "Nuclear Criticality Control of Special Actinide Elements" (ANSI/ANS-8.15-1981) (Criterion 1.2.6.3)

"Nuclear Criticality Control of Special Actinide Elements" is specifically cited in the "Disposal Criticality Analysis Methodology Topical Report" as a governing requirements document. Since this document defines the methodology used for all criticality evaluations, this standard has been applied to all criticality results cited in this document (reference "Design Analysis for UCF Waste Packages," Section 6).

2.5.5.4 "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors" (ANSI/ANS-8.1-1998) (Criterion 1.2.6.4)

"Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors" is specifically cited in the "Disposal Criticality Analysis Methodology Topical Report" as a governing requirements document. Since this document defines the methodology used for all criticality evaluations, this standard has been applied to all criticality results cited in this document (reference "Design Analysis for UCF Waste Packages," Section 6).

2.5.5.5 "Criteria for Nuclear Criticality Safety Controls in Operations with Shielding and Confinement" (ANSI/ANS-8.10-1983) (Criterion 1.2.6.5)

"Criteria for Nuclear Criticality Safety Controls in Operations with Shielding and Confinement" is specifically cited in the "Disposal Criticality Analysis Methodology Topical Report" as a governing requirements document. Since this document defines the methodology used for all criticality evaluations, this standard has been applied to all criticality results cited in this document (reference "Design Analysis for UCF Waste Packages," Section 6).

2.5.5.6 "Criticality Safety Criteria for the Handling, Storage, and Transportation of LWR Fuel Outside Reactors" (ANSI/ANS-8.17-1984) (Criterion 1.2.6.6)

"Criticality Safety Criteria for the Handling, Storage, and Transportation of LWR Fuel Outside Reactors" is specifically cited in the "Disposal Criticality Analysis Methodology Topical Report" as a governing requirements document. Since this document defines the methodology used for all criticality evaluations, this standard has been applied to all criticality results cited in this document (reference "Design Analysis for UCF Waste Packages," Section 6).

3. SYSTEM OPERATIONS

This section will be completed in a later revision.

4. SYSTEM MAINTENANCE

This section will be completed in a later revision.

APPENDIX A CRITERION BASIS STATEMENTS

This section presents the criterion basis statements for criteria in Section 1.2. Descriptions of the traces to "Monitored Geologic Repository Requirements Document" (MGR RD) 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 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 (size, weight, and inventory) is included for information only. 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 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 that 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 5.2.12).

1.2.1.5 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" (Section 5.2.23).

II. Criterion Performance Parameter Basis

N/A

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, in support of MGR RD 3.2.L, 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 "Standard Review Plan for Dry Cask Storage Systems" (NUREG-1536), Section 4.V.5.b.

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 waste packages 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 3.2.H, which requires the MGR to include provisions which 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 waste package 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.

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. v and 2-16).

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 the Total System Performance Assessment (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 "American National Standard for Radioactive Materials - Special Lifting Devices for Shipping Containers Weighing 10000 Pounds (4500 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 "American National Standard for Radioactive Materials - Special Lifting Devices for Shipping Containers Weighing 10000 Pounds (4500 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), which requires the disposal container system perform its intended safety function during normal operations.

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

1.2.1.22 Criterion Basis Statement

I. Criterion Need Basis

This criterion is needed to ensure residual stresses do not initiate stress corrosion cracking in the waste package.

II. Criterion Performance Parameter Basis

The stress limit of 10 percent of the outer material yield strength is provided in "Waste Package Degradation Process Model Report," Section 3.1.9.4.

1.2.1.23 Criterion Basis Statement

I. Criterion Need Basis

This criterion is needed to ensure static loads do not initiate stress corrosion cracking in the waste package.

II. Criterion Performance Parameter Basis

The stress limit of 10 percent of the outer material yield strength is provided in "Waste Package Degradation Process Model Report," Section 3.1.9.4.

1.2.1.24 Criterion Basis Statement

I. Criterion Need Basis

This criterion is needed to ensure seismic loads do not result in permanent strains that could initiate stress corrosion cracking in the waste package. Maintaining stresses below yield stress ensures that permanent strains are not induced.

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 Disposal Container System Description Documents" (Attachment 1, p. 1) (Input Transmittal RSO-RSO-99333.Tb).

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 "Preclosure Design Basis Events Related to Waste Packages" (Section 7.2.1).

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 Disposal Container System Description Documents" (Attachment 1, p. 1) (Input Transmittal RSO-RSO-99333.Tb).

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 Disposal Container System Description Documents" (Attachment 1, p. 1) (Input Transmittal RSO-RSO-99333.Tb).

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 Disposal Container System Description Documents" (Attachment 1, p. 1) (Input Transmittal RSO-RSO-99333.Tb).

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 Disposal Container System Description Documents" (Attachment 1, p. 1) (Input Transmittal RSO-RSO-99333.Tb).

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 Disposal Container System Description Documents" (Attachment 1, p. 1) (Input Transmittal RSO-RSO-99333.Tb).

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 Disposal Container System Description Documents" (Attachment 1, p. 1) (Input Transmittal RSO-RSO-99333.Tb).

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 modified from "Nuclear Safety Criteria for Disposal Container System Description Documents" (Attachment 1, p. 1) (Input Transmittal RSO-RSO-99333.Tb).

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 drop height is to be determined.

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 Disposal

Container System Description Documents" (Attachment 1, p. 1) (Input Transmittal RSO-RSO-99333.Tb).

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 Disposal Container System Description Documents" (Attachment 1, p. 1) (Input Transmittal RSO-RSO-99333.Tb).

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 Disposal Container System Description Documents" (Attachment 1, p. 1) (Input Transmittal RSO-RSO-99333.Tb).

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 Disposal Container System Description Documents" (Attachment 1, p. 1) (Input Transmittal RSO-RSO-99333.Tb).

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 Disposal Container System Description Documents" (Attachment 1, p. 2) (Input Transmittal RSO-RSO-99333.Tb).

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 Disposal Container System Description Documents" (Attachment 1, p. 2) (Input Transmittal RSO-RSO-99333.Tb).

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 Disposal Container System Description Documents" (Attachment 1, p. 2) (Input Transmittal RSO-RSO-99333.Tb).

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 Disposal Container System Description Documents" (Attachment 1, p. 2) (Input Transmittal RSO-RSO-99333.Tb).

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 Disposal Container System Description Documents" (Attachment 1, p. 2) (Input Transmittal RSO-RSO-99333.Tb).

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 Disposal Container System Description Documents" (Attachment 1, p. 2) (Input Transmittal RSO-RSO-99333.Tb).

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 defined 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 Disposal Container System Description Documents" (Attachment 1, p. 2) (Input Transmittal RSO-RSO-99333.Tb).

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 Disposal Container System Description Documents" (Attachment 1, p. 2) (Input Transmittal RSO-RSO-99333.Tb).

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 Disposal Container System Description Documents" (Attachment 1, p. 2) (Input Transmittal RSO-RSO-99333.Tb). 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 6.12gl and 6.13gl contained in the "MGR Compliance Program Guidance Package for the Uncanistered Spent Nuclear Fuel (SNF) Disposal Container."

II. Criterion Performance Parameter Basis

The performance parameter for this requirement is taken from "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants" (p. 5).

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. Conformance to the parameter basis must be met assuming occurrence of design basis events, including those with the potential to flood the disposal container prior to its sealing and the misload of assemblies into a disposal container. Allowance for the bias in the method of calculation and the uncertainty in the experiments used to validate the method of calculation for the calculated effective multiplication factor shall be justified in addition to conformance to the parameter basis.

1.2.2.1.13 Criterion Basis Statement

I. Criterion Need Basis

This criterion is a design screening component of the postclosure criticality methodology 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 total frequency is to be measured as the sum of all of the frequencies of criticality from all waste forms and waste package designs. This total frequency is representative of the entire potential repository. Prior to License Application, conformance to the parameter basis may be established from the combination of all of the frequencies of the individual configurations which have been established to date, representative of the entire contents of the repository of the waste forms evaluated. This is established as the evaluation of all waste package and waste form configurations which may be potentially critical will not be completed until License Application.

This criterion, in combination with Criterion 1.2.2.1.14, is designed to ensure that criticality occurrences in the repository will not impact the TSPA assessment of expected annual dose to the critical group during the first 10,000 years after permanent closure. The establishment of the specific parameter is such that intermediate criticality risk is less than the uncertainty in the TSPA calculation of expected annual dose to the critical group. The intermediate criticality risk is defined as the sum of the products of frequency of criticality occurrence (for a single SNF waste package, as a function of time), multiplied by the radionuclide inventory (measured in curies) due to that criticality, divided by the radionuclide inventory of a single waste package, with the integral taken over the entire timeframe considered for repository performance evaluation. Both the radionuclide inventory and the increment due to criticality shall be evaluated at 1,000 years following the criticality shutdown.

1.2.2.1.14 Criterion Basis Statement

I. Criterion Need Basis

This criterion is needed to comply with MGR RD 3.1.C. The criterion is a design screening component of the postclosure disposal criticality methodology 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 radionuclide inventory increase from a single criticality shall be determined at 1,000 years following the end of the criticality by comparing the radionuclide increment due to the criticality (at that time) divided by the total radionuclide inventory available for release and transport to the accessible environment (at that time). Available release is defined as radionuclide inventory from waste packages that have breached barriers and the waste form that have degraded sufficiently to expose the radionuclides to dissolving water.

This criterion, in combination with Criterion 1.2.2.1.13, is designed to ensure that criticality occurrences in the repository will not impact the TSPA assessment of expected annual dose to the critical group during the first 10,000 years after permanent closure. The establishment of the specific parameter, 10 percent inventory increase, is such that intermediate criticality risk is less than the uncertainty in the TSPA calculation of expected annual dose to the critical group, where intermediate criticality risk is defined in Criterion Basis Statement 1.2.2.1.13.

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 microbe influx, colloid concentration, temperature, and radiation 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 pH, humidity, and water seepage parameters are taken from the "Monitored Geologic Repository Project Description Document" (Sections 5.2.25, 5.2.26, and 5.2.28). The induced handling environments are to be determined.

1.2.4.1 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 is taken from "Design Analysis for UCF Waste Packages," Section 6.2.3.1. The resultant value is conservatively rounded up to the nearest 10 rem/hr.

1.2.4.2 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 5.2.13).

1.2.4.3 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 will be updated 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.

1.2.4.4 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

1.2.4.5 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

1.2.4.6 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 Criticality 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 system architecture and QA classification are identified in Table 21. The QA classifications are established in Table 1 of "Classification of the MGR Uncanistered Spent Nuclear Fuel Disposal Container System," which defines the overall system as QL-1. The next level of system architecture and assumed QA classification are identified in Table 21.

Table 21. 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 Absorber Plate Disposal Container	X			
21-PWR Control Rod Disposal Container	X			
12-PWR Long Disposal Container	X			
44-BWR Disposal Container	X			
24-BWR Disposal Container	X			

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

APPENDIX C ACRONYMS, SYMBOLS, AND UNITS

C.1 ACRONYMS

This section provides a listing of acronyms used in this document.

ASME	American Society of Mechanical Engineers
BWR	boiling water reactor
B&W	Babcock and Wilcox
CE	Combustion Engineering
CQ	conventional quality
CRWMS	Civilian Radioactive Waste Management System
CSNF	commercial spent nuclear fuel
DHLW	Defense High-Level Waste
DOE	U. S. Department of Energy
F	function
GE	General Electric
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
UCF	uncanistered spent nuclear fuel
WE	Westinghouse Electric
WP	waste package

C.2 SYMBOLS AND UNITS

This section provides a listing of symbols and units used in this document.

%	percent
±	plus or minus
°C	degrees Celsius
°F	degrees Fahrenheit
μin	microinch
μm	micrometer
cm	centimeter
CO_2	carbon dioxide
CO	carbon monoxide
ft	feet
GWd/MTU	gigawatt-days per MTU
H_2	hydrogen
H_2O	water
hr	hour
Hz	Hertz
in.	inch
k_{eff}	effective multiplication factor
kg	kilogram
km	kilometer
kW	kilowatt
lb	pound
m	meter
mg	milligram
mi	mile
ml	milliliter
mm	millimeter
MPa	megaPascal
mrem	one thousandth of a rem
MT	metric ton
MTU	metric tons uranium
O_2	oxygen
pH	hydrogen ion concentration potential
ppm	parts per million
psia	pounds per square inch-atmospheric
rem	Roentgen equivalent man
sec	second
vol%	percent of volume
wt%	percent of weight
yr	year

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.

D.1 Issue 1—Label Legibility

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

D.2 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.

D.3 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.

D.4 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.

D.5 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.

D.6 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.

D.7 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.

D.8 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.

D.9 Issue 9— Waste Package Performance

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

D.10 Issue 10— Waste Package Postclosure Criticality Requirements

Postclosure criticality requirements will be determined for a future revision.

D.11 Issue 11— 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.

D.12 Issue 12— 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 this SDD. References list the Accession number or Technical Information Catalog number at the end of the reference, where applicable.

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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.