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QA: L

Civilian Radioactive Waste Management System
Management & Operating Contractor

MINED GEOLOGIC DISPOSAL SYSTEM ADVANCED CONCEPTUAL DESIGN REPORT

VOLUME III OF IV
ENGINEERED BARRIER SEGMENT/WASTE PACKAGE

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1. INTRODUCTION

This is the first phase of a three-phase design effort, which consists of Advanced Conceptual Design (ACD), License Application Design, and Final Procurement and Construction Design. This volume of the Mined Geologic Disposal System (MGDS) ACD report provides a review of the Engineered Barrier Segment/Waste Package (WP) development process to date. The evaluations and results presented in this report support the design goals of the *Waste Package Implementation Plan* (YMP 1995), and address the design driving requirements provided in the *Engineered Barrier Design Requirements Document* (YMP 1994b).

The design requirements for the Engineered Barrier Segment are provided in the *Engineered Barrier Design Requirements Document.* This document provides all of the requirements which the Engineered Barrier Segment design will meet at this stage of design and the basis upon which the design will be evaluated upon completion of the final designs. Conceptual design, by definition, is not a compete or final design and does not meet all of the requirements. A conceptual design is intended to address those requirements which are the major design drivers and show that a final design can be completed with a high degree of confidence. The Engineered Barrier Segment Conceptual Design addresses the major design requirements in the *Engineered Barrier Design Requirements Document.* The related design assumptions are provided in Section 3 and are closely related to the program requirements presented in Section 2. A complete cross referencing and accounting of all *Engineered Barrier Design Requirements Document* requirements cannot be presented until a final design has been completed.

There are two sets of nomenclature used throughout this report. One set is of a generic nature and is most used when discussing high-level program requirements and direction and when talking in general terms. (An example of which is the generic term "repository" as used to refer to the entire disposal site and operation, and the engineered barrier system as defined by 10 CFR 60.) The second set is more specific to the program and reflects the systems engineering approach adopted by the Management and Operating Contractor (M&O) to configure the systems, structures, and components that make up the CRWMS (such as the terms "Repository Segment" and "Engineered Barrier Segment"). These refer to specific and well defined Configuration Items as subsets of the MGDS Element (e.g., the Engineered Barrier Segment architecture is shown in Figure 3-5). Generic terms are usually shown in lower case letters. Configuration Items are always shown in initial upper case and end in the given hierarchial words assigned to the Configuration Item architecture structure (e.g., Waste Package Subsystem).

During the ACD phase a number of Engineered Barrier Segment/WP design options were developed and evaluated for adequacy with respect to applicable performance and design requirements. Previous ACD work is documented in the *Initial Summary Report for Repository/Waste Package Advanced Conceptual Design* (CRWMS M&O 1994i). This report is intended to document the recommended Engineered Barrier Segment/WP designs which are recommended to be the basis of follow on design efforts. The Engineered Barrier Segment/WP ACD provides a reference from which the work will proceed towards final design.

The function of the Engineered Barrier Segment/WP is to safely contain and isolate spent nuclear fuel (SNF) and high-level wastes (HLW). Furthermore, the Engineered Barrier Segment will allow the retrieval of SNF from the repository as specified in the *Engineered Barrier Design Requirements Document* (YMP 1994b). The time period for Engineered Barrier Segment performance is from the initial loading of waste and closure of the WP through at least 10,000 years following permanent closure of the repository facility. The federal regulations in 10 CFR 60.113 specify post closure performance requirements during the containment period of 300 to 1,000 years and do not place an upper limit on the period of performance as clarified in U.S. Nuclear Regulatory Commission staff position 60-001 (NRC 1990a). Requirements are discussed in Section 2 of this volume.

As defined in 10 CFR 60.2, the Engineered Barrier System is the waste package and the underground facility. The MGDS program chose to define an Engineered Barrier Segment as presented in Section 6 of the *Engineered Barrier Design Requirements Document* in order to control the interfaces required for design in an efficient fashion. The Engineered Barrier Segment is a subset of the Engineered Barrier System and includes the WPs and other engineered features of the underground facility provided to support the Contain and Isolate Waste function (such as emplacement drift openings; backfill; and invert, as defined in Volume II). This volume describes the Waste Package Subsystem conceptual design. The balance of the Engineered Barrier Segment design is described in Volume II.

As presented in Volume I of this report, the repository will provide space for waste emplacement in horizontal drifts, the exact size and spacing depending on the technologies now under study. As described in Sections 2 and 3 of this volume, the geologic repository is part of the controlled area and consists of both surface and subsurface areas. The architecture and the allocation of functions to the Engineered Barrier Segment components are provided in Section 3. The Engineered Barrier Segment components will provide the primary facilities, hardware, processes, and procedures that comprise the physical means to accomplish all Engineered Barrier Segment functions.

The WP component of the Engineered Barrier Segment will be designed to accommodate SNF and HLW forms. Waste forms including canistered waste forms, will be placed in the WP container and sealed. A detailed presentation of the functions and requirements for the Engineered Barrier Segment is provided in Sections 2 and 3. Briefly, the major design considerations are as follows: Nuclear criticality will be controlled by such methods as maintaining adequate spacing between the waste forms in the WP or providing neutron absorbers. The strategy for demonstrating criticality control also places heavy emphasis on taking credit for the decrease in criticality potential of spent nuclear fuel as compared to that of new, unirradiated fuel.

1.1 MINED GEOLOGIC DISPOSAL SYSTEM ADVANCED CONCEPTUAL DESIGN VOLUME III OBJECTIVE

The objective of this volume of the MGDS ACD report is to describe the Engineered Barrier Segment design, of which the WP is the major component, and to present the body of supporting analysis which the design is based upon. Conceptual design information presented in this report is derived from and is based on other more detailed documents, which are available. The Engineered Barrier Segment/WP volume of the MGDS ACD report is intended to put forth a compilation of Engineered Barrier Segment/WP developed concepts with sufficient information concerning

alternate designs and repository conditions to show that the selected design presented in Volume I is the best design to move forward towards construction. The information is presented in a form and format which can be used to review technical progress and adequacy as well as programmatic status. This volume is formatted to include sections of technical and programmatic information which correspond to goals identified in the *Waste Package Implementation Plan*. Information for each of the identified areas may not be available at this time; however, the reader will be able to assess both the progress of work and the level of effort required in other areas to fully complete an Engineered Barrier Segment/WP design. As such, each section will indicate areas where future efforts are required and in which efforts are currently on-going.

This report is structured using as a guideline the prescribed content in the U.S. Department of Energy, DOE Order 4700.1, *Project Management System*, and the U.S. Nuclear Regulatory Commission Regulatory Guide 4.17, *Standard Format and Content of Site Characterization Reports for High-Level Waste Geologic Repositories*." It is designed to be a compendium of information available to date on the ACD of Engineered Barrier Segment/WP development. As such, the content of this volume is presented in condensed version, and in many cases refers to more detailed documents that are available from the M&O Records Processing Center. The intent of this volume is to serve as a current reference source of information of the ACD, and can be used to review the program technical adequacy, compliance with requirements, interface control, and project planning and scheduling. The report contains references to design information from Repository Surface and Subsurface Design and Systems Engineering presented in Volume II of this report concerning intended WP handling.

1.2 ENGINEERED BARRIER DESIGNS

The Engineered Barrier Segment functions are to isolate and contain the waste in a fashion which allows determination of compliance to requirements by evaluation of system performance. The performance and operation requirements are presented in Sections 2 and 3 of this volume. The Engineered Barrier Segment is designed to accomplish the intended function by thoroughly designing each component to function synergistically. The major components of the Engineered Barrier Segment include, but are not limited to, the WP, the drift invert material and design, the WP emplacement pedestal or support, drift backfill material, if it is to be used, drip shields for the WPs, drift opening design and other components of the emplacement drift which can be designed to isolate and contain the waste.

The information presented in Volume III concerning the Engineered Barrier Segment focuses on the containment and isolation of radioactive waste functions and includes the selection of materials, design of the structures, and assessing the long term performance of the Engineered Barrier Segment designs. The physical construction of the emplacement drifts, drift openings, and drift inverts is provided in Volume II which presents the construction methods for the underground facility. Discussion of the material selection and performance characteristics which may be assigned to drift invert, backfill, and drip shields is presented in Sections 4, 6.2, and 8. Discussion of the types of events which the Engineered Barrier Segment will have to endure, the anticipated degradation modes, and the method of evaluating the effectiveness of the Engineeered Barrier Segment is provided in Section 7.

The major component of the Engineered Barrier Segment is the WP which has been the major focus of the Engineered Barrier Segment design effort to date. Also, since the largest portion of the long-term performance of the Engineered Barrier Segment has been allocated to the WP design, a brief introduction to the WP design is provided in the following sections. Over the repository lifetime, the WP containment barriers will perform various functions that will change with time. During the 100-year or longer operational period, the barriers will function as the vessel for handling, emplacing, and retrieving (if necessary) the contents of the WP. This will be followed by the 1,000-year containment period, and then the period of controlled release extending beyond 10,000 years. During the first 1,000 years following repository closure, the containment barriers will be relied upon to provide substantially complete containment of the radionuclides. Beyond 1,000 years the containment barriers of the Engineered Barrier Segment are designed to impede release of radionuclides by aqueous transport from those WPs that have breached during the subsequent controlled release period. However, breached containment barriers are expected to continue to inhibit transport of liquid water into, and radionuclides out of, the WPs. The use of a multibarrier design will result in a lower breach rate distributed over a longer period of time. This will help ensure that the release rate limits are met.

As a consequence of historical usage, the term WP is used frequently throughout this report; however, this usage refers to the Waste Package Subsystem Configuration Item, and not the 10 CFR 60.2 generic definition. An empty WP is a disposal container (Disposal Container Subsystem Element). Waste container is a generic term for a Waste Package Subsystem without the Shielding Subsystem Element and the Packing and Absorbent Materials Subsystem Element (see Figure 3-5 and the glossary).

1.2.1 Waste Package Conceptual Design Evolution

From the beginning of the conceptual design phase, the WP development process has been evolutionary. As systems strategies and long term performance strategies evolved, there has been a corresponding response in WP design options. At the start of conceptual design, there were seven WP design options:

1. Metallic Multi-Barrier
2. Metallic Multi-Barrier Shielded
3. Small Metallic Multi-Barrier Borehole
4. Nonmetallic/Metallic Multi-Barrier
5. Multi-Purpose Canister (MPC) with Disposal Container
6. Universal Cask (Multi-Purpose Unit)
7. Site Characterization Plan-Conceptual Design Report (Single Container) Borehole.

In addition, there were initial designs for Defense High-Level Waste (DHLW) and Uncanistered Spent Nuclear Fuel (UCF). The work on these concepts is documented in the *Initial Summary Report for Repository/Waste Package Advanced Conceptual Design* (CRWMS M&O 1994i).

As presented in Volume I of this report, the repository facility is designed based upon the premise that a Canistered Fuel (CF) canister will be implemented as the prime SNF waste handling component of the system to ship SNF to the repository. In order to assess the performance of the CF

system in a repository environment, the MPC conceptual design (CRWMS M&O 1994k) was selected to be the basis for evaluation of the CF container concepts. Thus, the use of the term MPC is used for the term CF in this document whenever analysis results are presented. However, the term CF represents a larger class of containers than the MPC. The results presented in this volume for the CF container are specific to the MPC conceptual design and will be different if a alternate CF design is selected for use in the MGDS. The analysis presented in this volume for the CF container is representative of the design and operational considerations which the MGDS must examine for any type of CF canister which may be selected.

Early in 1994 the SNF waste handling and disposal design basis was formally changed from the original smaller *Site Characterization Plan* (DOE 1988) uncanistered SNF WP design concept, by adoption of the larger drift emplaced CF-based concept. This was a fundamental change in MGDS design philosophy, from small, thin-walled package borehole emplacement to large, thick-walled package in-drift emplacement. The nominal CF container will be designed to accommodate a very high percentage of all known and expected SNF assembly characteristics (geometric, thermal, and neutronic). The SNF not accomodated would be considered nonstandard fuel and handled separately. With the implementation of the CF canister, a disposal container design and handling concepts for MGDS disposal needed to evolve to accommodate the CF canister concept. The MPC Conceptual Design Report (CRWMS M&O 1994k) noted that:

1. The large majority of the reactor sites can accommodate one or the other of the two primary sizes of CF canisters.

2. There will be a number of reactor sites that will not be capable of utilizing the CF canisters and will continue to ship SNF in casks as uncanistered fuel.

3. No radionuclide containment credit will be taken for the CF canister's outer shell.

4. The MPC will be designed to meet Code of Federal Regulation Titles 10 CFR 71 and 10 CFR 72, Transportation and Storage respectively, and be as compatible with Title 10 CFR 60 as possible. However, other forms of CF canisters could be received by a repository.

This change in design basis had the effect of focusing ACD activities, somewhat narrowing the range of investigations. The final ACD WP nominal designs are the unshielded metallic multibarrier configuration. Within this report, the term WP or waste package is often used to refer to what may more properly be termed the waste container or waste disposal container (i.e. the formal definition of the waste package is the loaded sealed waste container plus any shielding, packing, and other absorbent materials immediately surrounding an individual waste container). In addition to accomodating the CF, UCF WP designs are also required. Presently, the UCF WPs are designed to SNF capacities similar to those of the CF conceptual designs. The disposal container multiple barriers are essentially the same whether for a CF disposal container or a UCF WP. The UCF WP design must include a basket design, but the UCF basket only needs to meet the thermal limitations, structural needs, and criticality requirements for disposal.

Finally, a WP design to dispose of HLW from domestic fuel reprocessing plants, both commercial- and defense-related, is also required. Since the majority of this form of HLW is from the DOE defense-related programs, the MGDS organization has historically referred to this category of waste as DHLW. DHLW is defined as the highly radioactive material resulting from the reprocessing of any solid material derived from such liquid waste that contains fission products in sufficient concentrations, and other highly radioactive material that has been determined by the U.S. Nuclear Regulatory Commission, consistent with the law, as requiring permanent isolation. As used in this section, DHLW is defined to include commercial and defense HLW from reprocessing operations and not the SNF itself. Note that the 10 CFR 60 definition of HLW includes SNF.

The combination of the MPC implementation decision and the Yucca Mountain Site Characterization Office direction to narrow the focus to relatively few design concepts as early as practical has resulted in reducing the WP design selection to the following three basic WP concepts:

1. UCF WP
2. DHLW WP
3. CF WP

Several conceptual SNF WP designs were developed during ACD to accommodate both large and small WP sizes and to investigate alternative means of fabrication. By the nature of both the ACD phase and the beginning of subsequent design phases, essentially all of the design inputs needed for the design work are as yet unqualified. Thus, discussions of these designs in the following subsections are discussions of unverified conceptual designs rather than of designs, which may be supported by qualified design inputs. Those design inputs yet requiring qualification include CF physical characteristics including the number of SNF assemblies, confirmation of the quantity of CF to be disposed of as CF if any, disposal container barrier materials and thicknesses, criticality control material selection, design basis fuel selections for thermal and neutronics analyses, selection of repository thermal loading design basis, and emplacement parameters including drift size and WP spacing.

1.2.2 Waste Package Design Descriptions

The functions of the multibarrier shell are to provide a containment barrier, act as a structural member subject to external loadings, heat conduction, and provide sufficient shielding to limit the radiation induced corrosion (radiolysis) of the containment barriers. The functions of the UCF basket are structural support, heat conduction, and criticality control.

All of the designs incorporate the multibarrier disposal container configuration. The same handling configuration is used for all WPs, whether a CF disposal container, UCF WP, or DHLW disposal container. An extension of the WP outer barrier, similarly on each end as a "skirt," results in the WP ends being recessed. There are three holes equally spaced around each skirt in this design. As required, each set of holes may be engaged in various ways: e.g., a fixture with radially actuated pins that move similar to the bolts in a safe door; or instead clevises may be affixed to the holes. Handling a closed WP in the horizontal position may require a lifting fixture with shoes to engage each skirt extension as a "shelf." This configuration provides for handling the WPs both empty and

after filling and closure. Alternative WP handling configurations will be investigated during subsequent design phases.

Design layouts for 13 specific designs, representing three WP design concepts, are included in Appendix B, as shown in Table 1-1.

Table 1-1. Design Layouts

Figure Series Description	Figure Numbers
Large Waste Package Tube Design (21-PWR)	B.1-1 through B.1-12
Large Waste Package Tube Design (44-BWR)	B.1-13 through B.1-23
Small Waste Package Tube Design (12-PWR)	B.1-24 through B.1-34
Small Waste Package Tube Design (24-BWR)	B.1-35 through B.1-45
Large Low TL/Humid Waste Package Tube Design (21-PWR)	B.1-46 through B.1-57
Large Low TL/Humid Waste Package Tube Design (44-BWR)	B.1-58 through B.1-68
Small Low TL/Humid Waste Package Tube Design (12-PWR)	B.1-69 through B.1-79
Small Low TL/Humid Waste Package Tube Design (24-BWR)	B.1-80 through B.1-90
Defense High-Level Waste Disposal Container	B.2-1 through B.2-7
Large Canistered Fuel Disposal Container (21-PWR/40-BWR)	B.3-1 through B.3-4
Small Canistered Fuel Disposal Container (12-PWR/24-BWR)	B.3-5 through B.3-8
Large Low TL/Humid MPC Disposal Container (21-PWR/40-BWR)	B.3-9 through B.3-12
Small Low TL/Humid MPC Disposal Container (12-PWR/24-BWR)	B.3-13 through B.3-16

1.2.2.1 Canistered Fuel Waste Package Concept

The CF waste package is being developed to accommodate either multipurpose or dual purpose (storage and disposal) canistered fuel. The conceptual design is a multibarrier concept consisting of two containment barriers, no packing, no filler, and with the basket designed as integral to the canister. In order to assess the performance of the CF container in a repository environment, the MPC conceptual design (CRWMS M&O 1994k) was selected to be the basis for evaluation of the CF container concepts. Thus, the term MPC is used for the term CF in this document; however, the term CF represents a larger class of canisters than the MPC.

The CF disposal container conceptual designs based upon the conceptual MPC design resulted in just the two CF sizes, large and small; the large CF size containing either 21 PWR or 40 BWR SNF assemblies, and the small CF containing either 12 PWR or 24 BWR SNF assemblies. The sealed CF canister will be placed into the CF disposal container, following which each of the two closure lids will be welded into place. Conceptual designs were developed and analyzed for both high and low repository thermal loads; however, the repository conceptual design is for the high thermal load

as presented in Appendix B. The CF disposal container sketches are presented in Figures 1-1 and 1-2.

The CF designs are sized to contain the largest possible number of SNF assemblies, subject to several constraints as discussed in the *MPC Conceptual Design Report*. The small and large CF conceptual design configurations (CRWMS M&O 1994k) correspond primarily to reactor facility crane limitations of 75 ton and 125 ton, respectively. Upon arrival at the repository, the sealed CF canister will be placed into a multibarrier CF disposal container. The CF canister remains sealed unless it becomes necessary to add some criticality control feature such as filler material or disposable control rods, in which case the CF canister opened at the repository, where the criticality control feature would be added. The CF design must include a basket design capable of meeting the criticality and thermal limitation requirements for storage, transportation, and disposal.

1.2.2.2 Defense High-Level Waste Waste Package Concept

The Savannah River Site HLW glass pour canister geometry was chosen as the basis for sizing the DHLW disposal container. The number of pour canisters per DHLW disposal container was chosen as four, to result in disposal container diameter approximately the same as for the large CF and UCF disposal container. The sealed pour canisters will be placed into the disposal container, following which each of the two closure lids will be welded into place.

The four-canister DHLW disposal container is shown in Figure 1-3. The selected cupronickel outer barrier material is satisfactory for either the high thermal load/dry case or the low thermal load/humid case.

1.2.2.3 Uncanistered Fuel Waste Package Concept

The UCF WP sketches are graphically presented in Figures 1-4, 1-5, 1-6, and 1-7. Conceptual designs were developed and analyzed for both high and low repository thermal loads; however, the repository conceptual design is for the high thermal load as presented in Volume I of this report. Uncanistered fuel assemblies will be individually placed into the disposal container, followed by addition of any filler material or disposable control rods if necessary, following which each of the two closure lids will be welded into place.

The UCF WP designs satisfy the design requirements by employing stainless steel-natural boron alloys in the basket construction, to provide for both criticality control and for the structural functions attributed to the basket (support and separation of the fuel assemblies). For the case of the UCF WP, the stainless steel-boron alloy material may be relied upon to perform the dual functions of criticality control and structural support due to the substantially lower imposed structural loadings associated with disposal, as compared to the higher structural loadings which are imposed upon the CF basket design due to transportation and storage structural loading criteria.

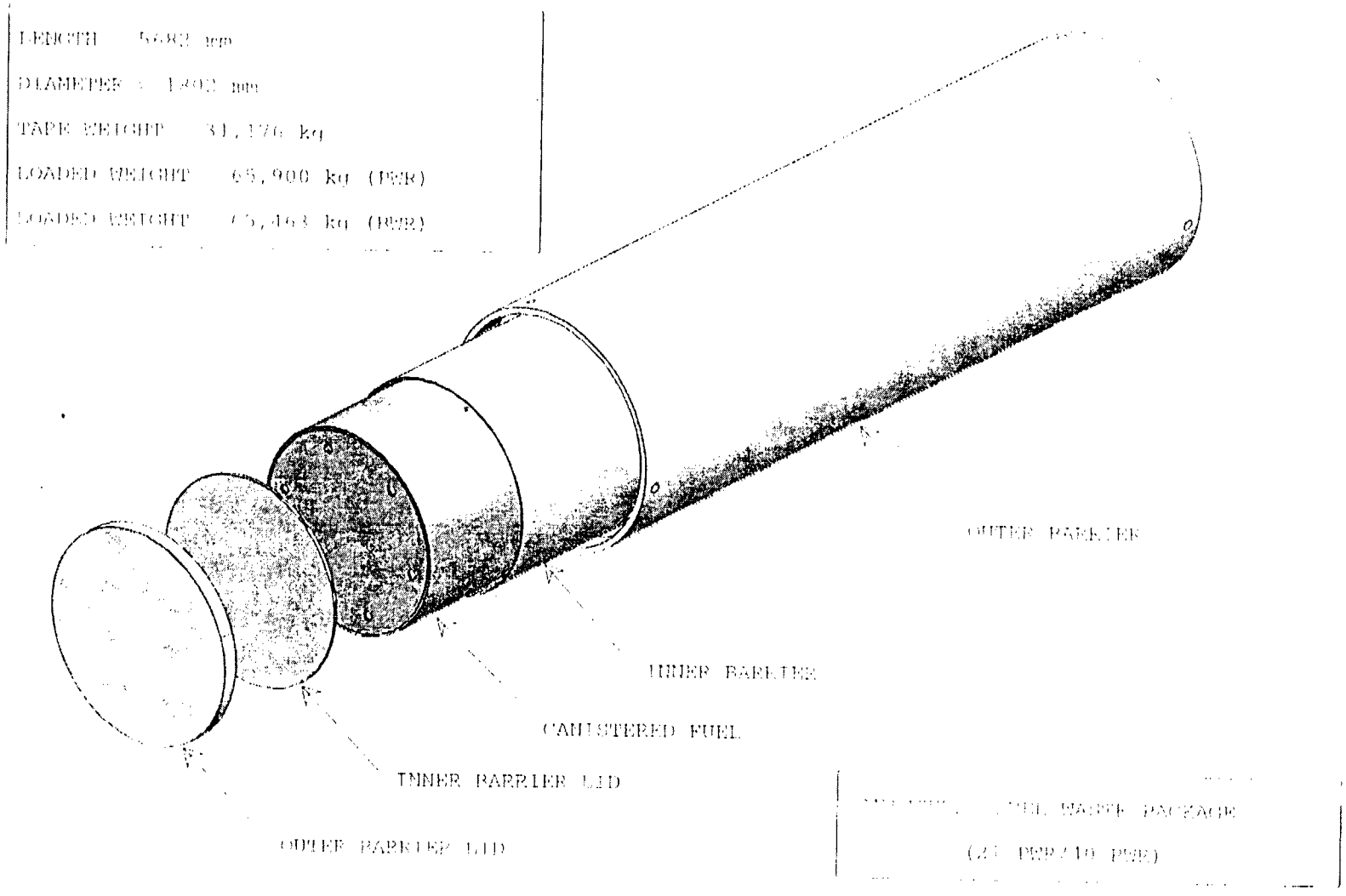

Figure 1-1. Large Canistered Fuel Waste Package Major Components

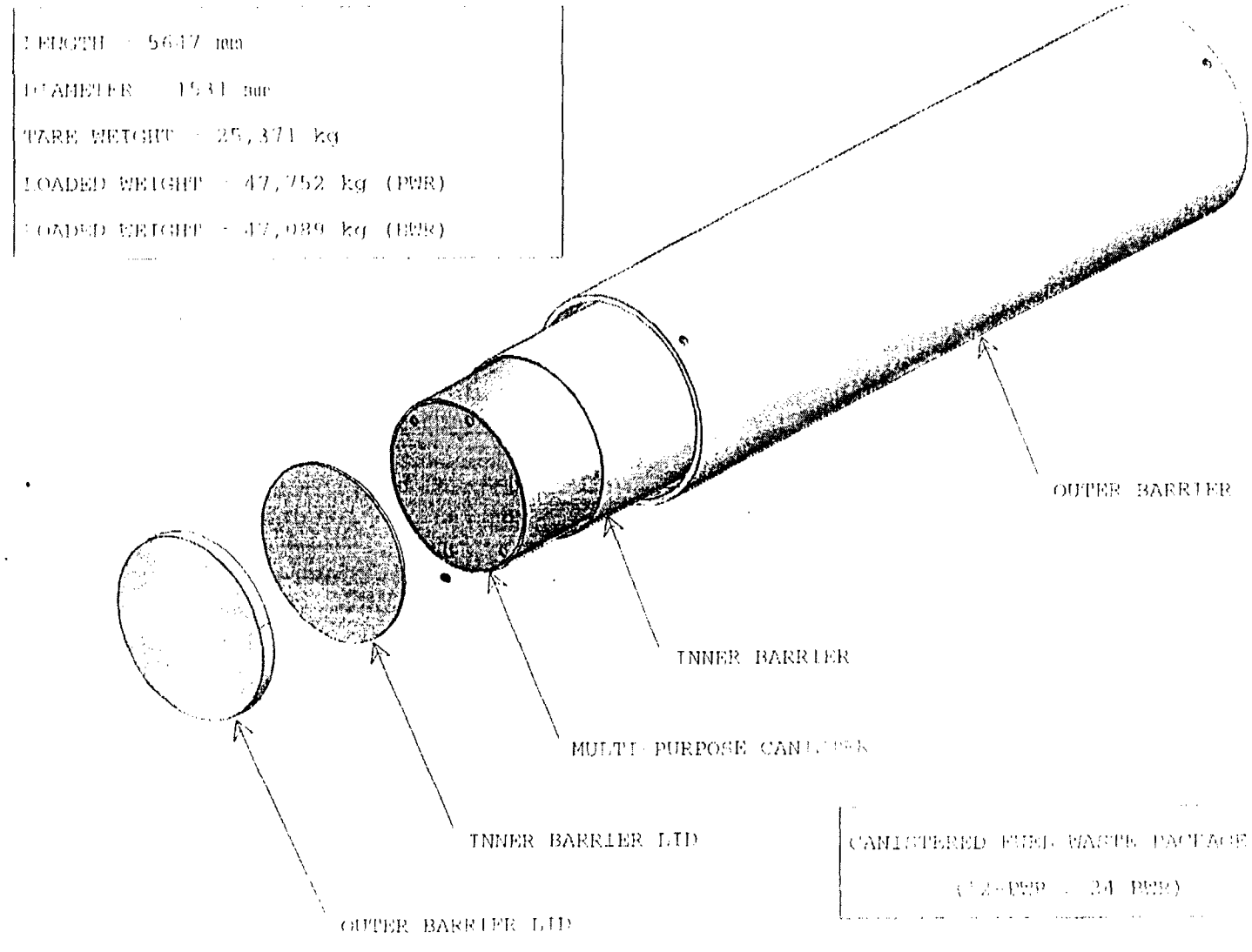

Figure 1-2. Small Canistered Fuel Waste Package Major Components

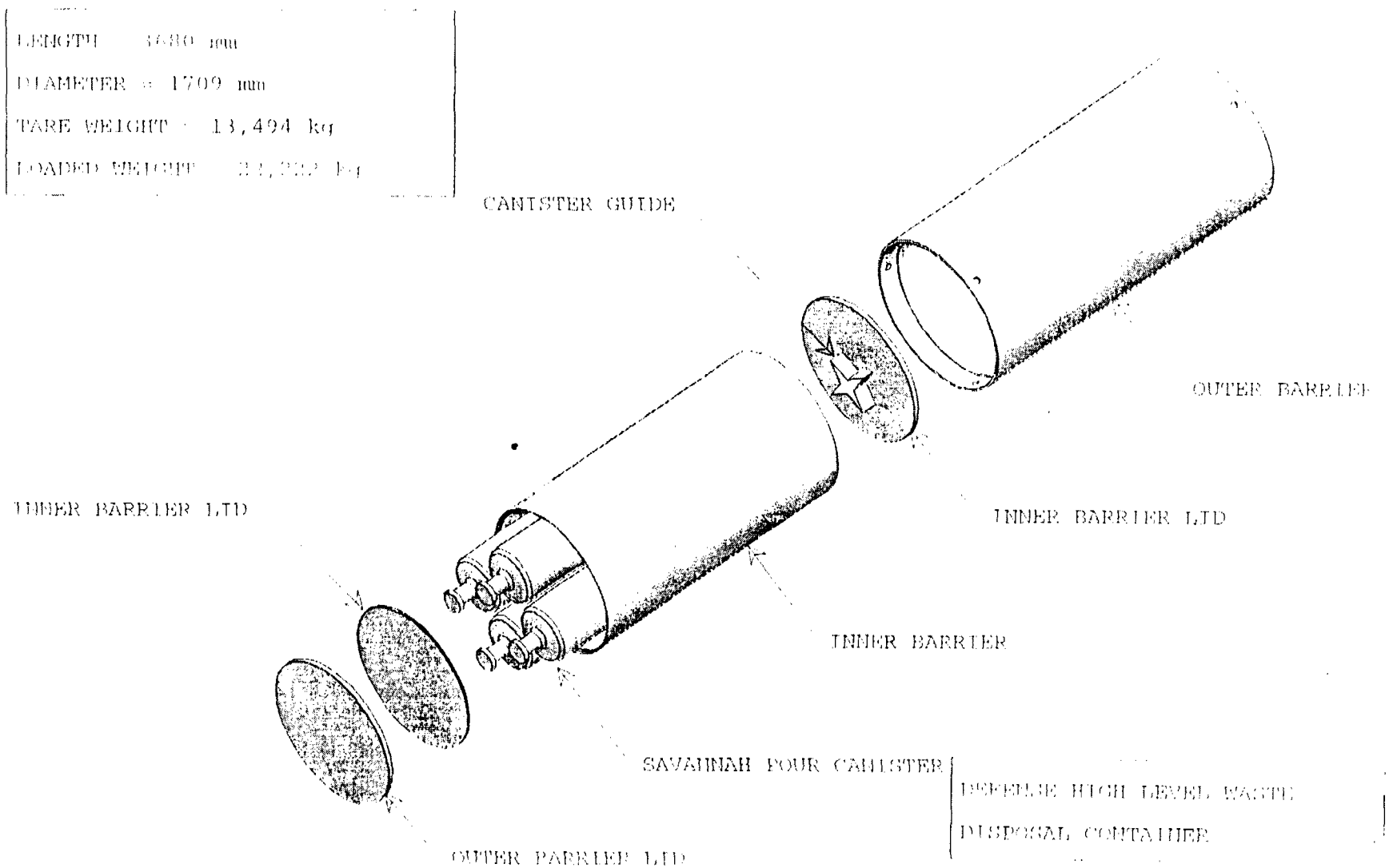

Figure 1-3. Defense High-Level Waste Package Major Components

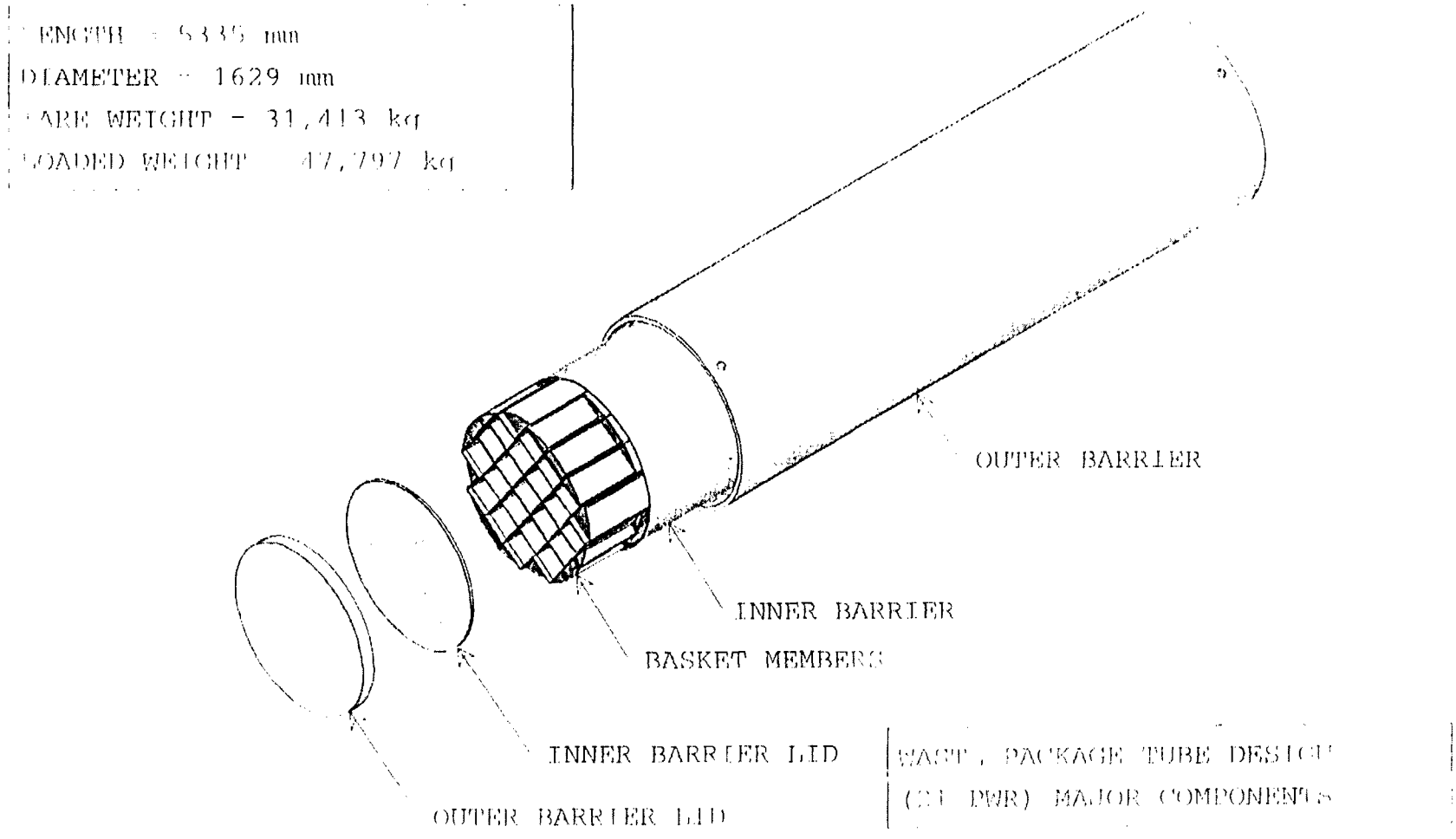

Figure 1-4. 21 Pressurized Water Reactor Uncanistered Fuel Waste Package Major Components

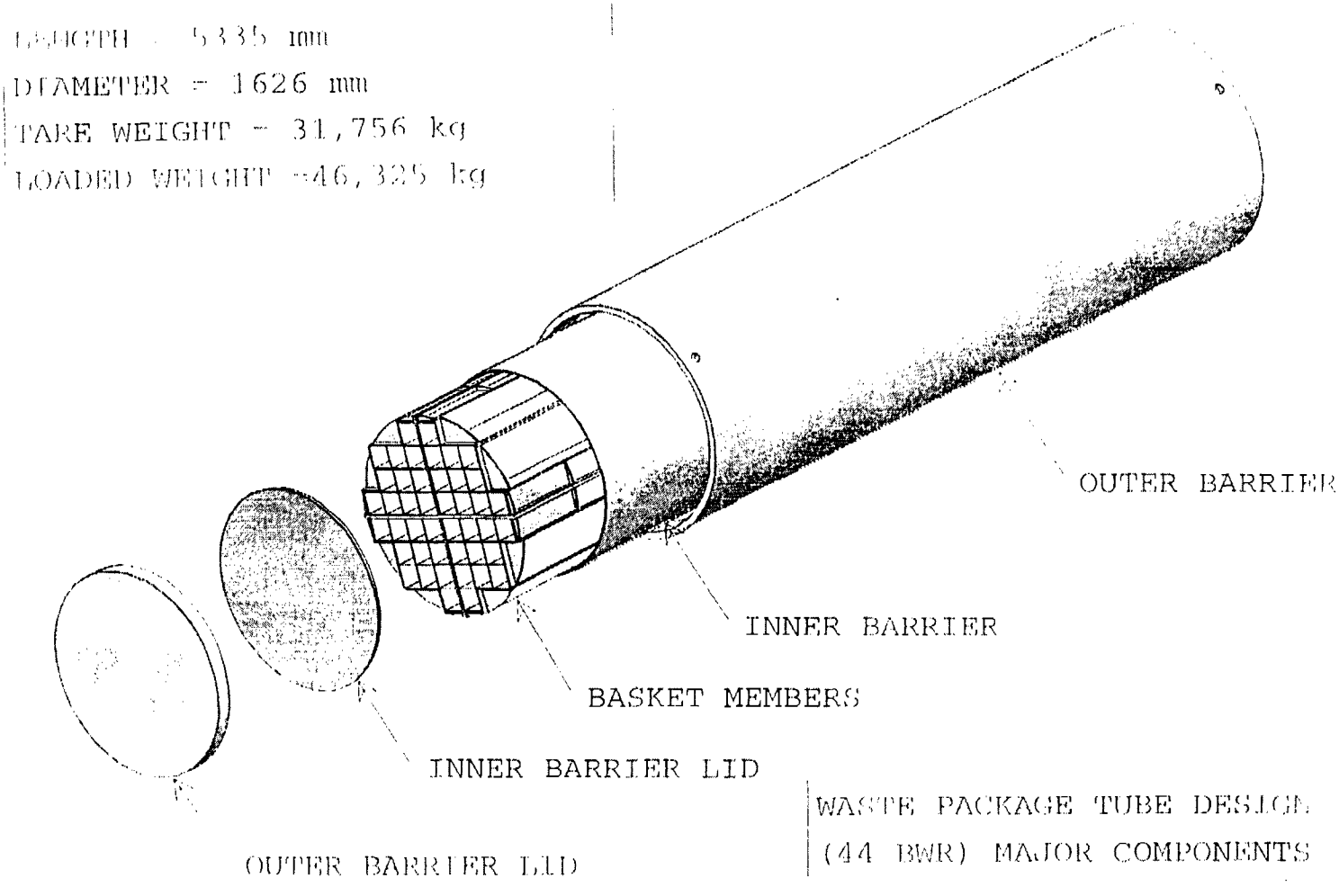

Figure 1-5. 44 Boiling Water Reactor Uncanistered Fuel Waste Package Major Components

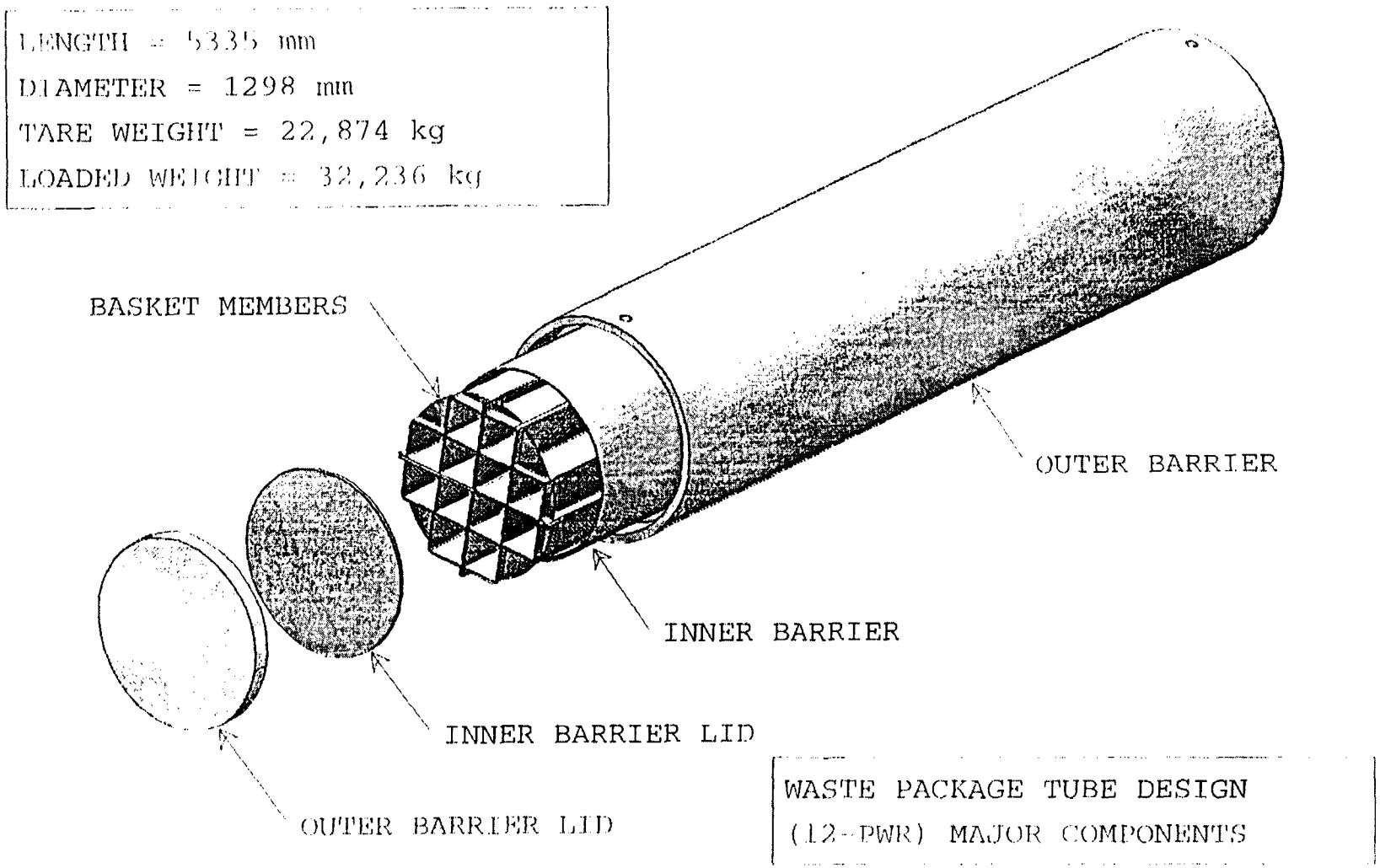

Figure 1-6. 12 Pressurized Water Reactor Uncanistered Fuel Waste Package Major Components

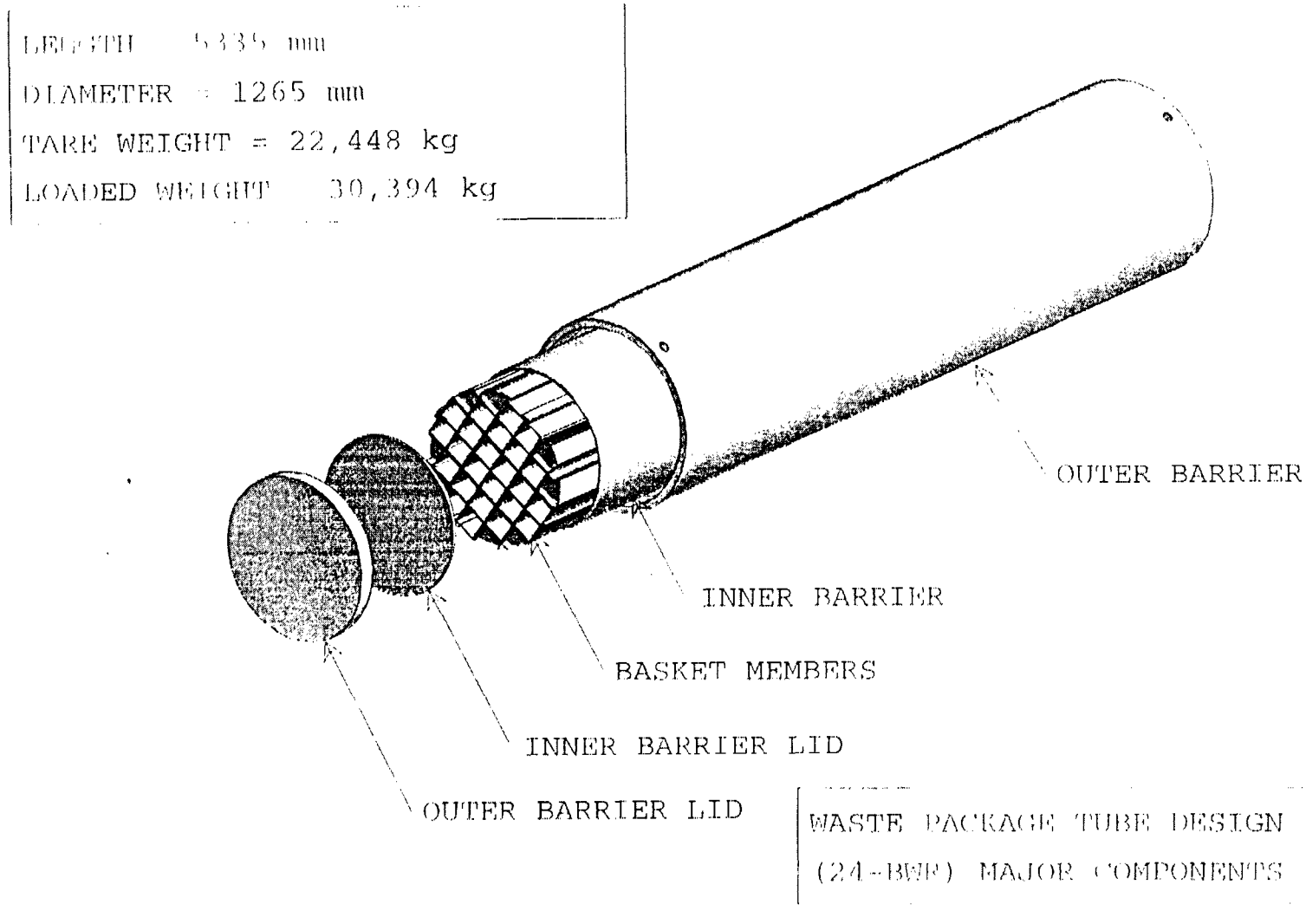

Figure 1-7. 24 Boiling Water Reactor Uncanistered Fuel Waste Package Major Components

The tube basket conceptual design is based on bundling together a number of identical full-length square tubes so as to form the necessary basket shape. The tubes would be formed by bending 316B6A alloy sheet into a square tubular shape, approximately 4.6 m long. A longitudinal weld would be used to close the tube longitudinal seam.

Use of several interlocking aluminum plates is also introduced into the tube type conceptual design as shown in Appendix B (examples of component sketches may be seen in Figures B.1-3, B.1-17, B.1-27, and B.1-39) to provide an enhancement of heat transfer paths to the WP shell (thermal shunts). The plates also serve to separate the tubes into clustered assemblies, as shown in Figures B.1-14 and B.1-25.

Standard design UCF waste containers that are based on the design basis fuels discussed in Section 5.2 will accommodate 96 to 97 percent of the 63,000 metric tons of uranium of spent commercial nuclear fuel that is planned for emplacement in the repository. The remaining 3 to 4 percent of the fuel not bounded by the design basis fuels (having more nuclear reactivity or higher thermal output than the design basis fuels, or are oversized) would be accommodated by other means, primarily using waste packages designed specifically to accommodate those SNF assemblies not bounded by the design basis fuels.

Various angle-iron guides and other structural forms (made of the same material as the inner barrier, Alloy 825) are installed within the interior of the disposal container shell to provide basket structural support and to guide installation of the basket.

1.3 NONSTANDARD WASTE PACKAGE

The SNF and other HLW that do not fall within the waste form categories defined in the *Engineered Barrier Design Requirements Document* will be classified as nonstandard. Nonstandard waste will be packaged to meet all of the same licensing requirements as standard SNF and HLW glass. If possible, nonstandard waste will be packaged in standard WPs. If this is not possible, nonstandard WP designs will be developed to accommodate the waste stream. During conceptual design, specifics of the nonstandard waste stream(s) were investigated and performance evaluation of various designs were initiated to support the unique features of the nonstandard waste.

2. DESIGN REQUIREMENTS AND STANDARDS

This section provides an overview of the requirements, codes, and standards that specify the general design and performance requirements related to the Engineered Barrier Segment.

The Engineered Barrier Segment and Waste Package (WP) development are being conducted in accordance with the *Quality Assurance Requirements and Description* document (DOE 1995c) through the use of a structured system of approved quality assurance implementing documents. The procedures used in the design process are described in this section.

The design requirements of the Engineered Barrier Segment are provided in the *Engineering Barrier Design Requirements Document* (YMP 1994b). This document provides all of the requirements which the Engineered Barrier Segment design will meet at this stage of design and the basis upon which the design acceptance will be evaluated. The source of requirements guiding the *Engineered Barrier Design Requirements Document* is the *Mined Geological Disposal System (MGDS) Requirements Document* (DOE 1995a). The *MGDS Requirements Document* is one of four System Requirements Documents that obtain requirements directly from the *Civilian Radioactive Waste Management System (CRWMS) Requirements Document* (DOE 1995e). The *CRWMS Requirements Document* is the top level source of requirements used in the Design Requirements Documents and System Requirements Documents. The *CRWMS Requirements Document* obtains its requirements from the Code of Federal Regulations, Department of Energy Orders, Nuclear Waste Policy Act.

2.1 QUALITY ASSURANCE

This document is developed under applicable M&O Quality Assurance (QA) implementing procedures which translate the requirements from the *Quality Assurance Requirements and Description* document. The determination that this document is applicable to the *Quality Assurance Requirements and Description* requirements, and the selection of the applicable implementing procedures, is documented in a Quality Administrative Procedure, QAP-2-0, *Control of Activities* evaluation (CRWMS M&O 1995bb). Much of the subject matter contained in this document pertains to designs of items on the Yucca Mountain Site Characterization Project (YMP) *Q-List* (YMP 1994c). QAP-2-3, *Classification of Permanent Items,* analyses are used in determining the classification of permanent items and their subsequent inclusion in the *Q-List.* At this time, a QAP-2-3 analysis of the Waste Package/Engineered Barrier Segment, that contains these items, has not been performed to determine its QA classification. However, these items are in the *Q-List* by direct inclusion by the U.S. Department of Energy (DOE). The contents of this volume represent conceptual design information that has been developed to date, and may be used to guide future design activities. However, use of any data from this document for input into documents supporting construction, fabrication, or procurement is required to be controlled in accordance with Nevada Site Administrative Procedure, NLP-3-15, *To Be Verified (TBV) and To Be Determined (TBD) Monitoring System.* Since this report utilized unqualified inputs and data, the entire report will be treated as unqualified.

Since information contained in this volume of the MGDS ACD will not be used as design inputs for manufacture, procurement, construction, or further design development by organizations outside the MGDS, the information does not require verification in accordance with QAP-3-2, *Design Verification*. Accordingly, the inputs associated with this document do not require verification and are not required to be controlled as TBV/TBD in accordance with NLP-3-15.

Figures contained in this document are preliminary in nature, are intended to convey conceptual information only, and as such, are not developed under the M&O Quality Administrative Procedures for controlled drawings. This document has been prepared according to the appropriate procedures identified in the applicable QAP-2-0 evaluation and in accordance with the Technical Document Preparation Plan (CRWMS M&O 1995l) developed for this purpose. To identify that this document is developed in accordance with the QA program, it contains the QA designator QA:L.

2.2 ENGINEERED BARRIER SEGMENT DESCRIPTION

The MGDS program chose to define an Engineered Barrier Segment as presented in Section 6 of the *Engineered Barrier Design Requirements Document* in order to control the interfaces required for design in an efficient fashion. The Engineered Barrier Segment is a subset of the Engineered Barrier System which includes the WPs and other engineered features of the underground facility provided to support the Contain Waste and Isolate Waste functions (such as emplacement drift orientation, geometry, layout, and depth; backfill; or invert). This volume deals with the design of the Engineered Barrier Segment specifically.

2.3 ENGINEERED BARRIER DESIGN REQUIREMENTS DOCUMENT REQUIREMENTS

The criteria applicable to Engineered Barrier Segment design provided in the *Engineered Barrier Design Requirements Document* (YMP 1994b). Many of these requirements are inherently a part of the WP designs presented in Appendix B. Material requirements are addressed in Section 4. Waste acceptance interface requirements are addressed in Sections 5 and 6.1. Repository interface requirements are addressed in Section 6.2. WP thermal requirements are addressed in Sections 6.3.3, 6.4.1, and 6.5.1. WP structural requirements are addressed in Sections 6.3.4, 6.4.2, and 6.5.2. WP criticality requirements are addressed in Sections 6.3.5, 6.4.3, and 6.5.3. WP shielding requirements are addressed in Sections 6.3.6, 6.4.4, and 6.5.4. Probabilistic methods to be used for long term performance assessment and WP/Engineered Barrier Segment performance assessment are addressed in Sections 7 and 8. Finally, the WP engineering development and manufacturing requirements are addressed in Section 9. The related program assumptions are provided in Section 3.2 and are closely related to the program requirements presented in this section.

A complete listing of requirements are provided in the referenced document. Table 2-1 is composed of a subset of those requirements that are applicable to this stage of design.

Table 2-1. Applicable Program Requirements

EBDRD Section	Para.	Applicable Requirement [and Source]
3.2.1 Performance Characteristics	A.	Mission Requirement. The design of the Engineered Barrier Segment, as part of the Mined Geologic Disposal System, shall provided for the disposal of SNF and civilian and defense high-level waste such that the public health and safety and the environment are protected.
	B.	Modes. The Engineered Barrier Segment will operate in the following seven modes, related to the eight modes described for the operations of the Mined Geologic Disposal System. These modes are applicable after the Nuclear Regulatory Commission approval and licensing process. 1. Construction 2. Emplacement 3. Caretaker 4. Retrieval 5. Decommissioning and closure 6. Post-closure 7. Off-normal The Engineered Barrier Segment can operate in more than one mode at a time. For example, the Mined Geologic Disposal System construction mode will begin after Nuclear Regulatory Commission issuance of the repository construction license. According to the current concept, only a portion of the emplacement locations will have been finished when the Mined Geologic Disposal System begins emplacement operations. The Mined Geologic Disposal System will continue to expand the underground facility in parallel with waste emplacement operations. This is a unique program aspect in that construction becomes part of the operation of the Mined Geologic Disposal System.
3.2.1.2 Emplacement Mode Requirements	C.	The Engineered Barrier Segment, together with the Repository Segment, shall be designed so that until permanent closure has been completed, radiation exposures, radiation levels, and releases of radioactive materials to unrestricted areas will at all times be maintained within the limits specified in 10 CFR 20 and applicable environmental standards for radioactivity established by the Environmental Protection Agency, as listed in Section 3.2.2. [10 CFR 60.111(a)]
3.2.1.3 Caretaker Mode Requirements		When the Mined Geologic Disposal System has reached its legislated or physical capacity for waste disposal, it will be in the caretaker mode. The option to retrieve any and all emplaced waste will be preserved for 50 years from the time emplacement is authorized. Performance confirmation continues during this mode. The Engineered Barrier Segment, together with the Repository Segment, shall be designed so that until permanent closure has been completed, radiation exposures, radiation levels, and releases of radioactive materials to unrestricted areas will at all times be maintained within the limits specified in 10 CFR 20 and applicable environmental standards for radioactivity established by the Environmental Protection Agency, as listed in Section 3.2.2. [10 CFR 60.111(a)]

Table 2-1. Applicable Program Requirements (Continued)

EBDRD Section	Para.	Applicable Requirement [and Source]
3.2.1.4 Retrieval Mode Requirements	A.	The Engineered Barrier Segment shall be designed and constructed to allow the retrieval of any SNF and HLW emplaced in the repository, during an appropriate period of operation of the facility, as specified by the Secretary of Energy.
	B.	The waste packages shall be designed to preserve the option of waste retrieval throughout the period during which wastes are being emplaced and, thereafter, until the completion of a performance confirmation program and Nuclear Regulatory Commission review of the information obtained from such a program. To satisfy this objective, the geologic repository shall be designed so that any or all of the emplaced waste could be retrieved on a reasonable schedule starting at any time up to 50 years after waste emplacement operations are initiated, unless a different time period is approved or specified by the Nuclear Regulatory Commission. 10 CFR 60.111(b)(3) gives guidance for developing the schedule. [10 CFR 60.111(b)(1)]
	C.	The Engineered Barrier Segment, together with the Repository Segment, shall be designed so that until permanent closure has been completed, radiation exposures, radiation levels, and releases of radioactive materials to unrestricted areas will at all times be maintained within the limits specified in 10 CFR 20 and applicable environmental standards for radioactivity established by the Environmental Protection Agency, as listed in Section 3.2.2. [10 CFR 60.111(a)]
3.2.1.6 Post-closure Mode Requirements	A.	The Engineered Barrier Segment shall support and facilitate a post-permanent closure monitoring program in accordance with the application to amend the license for permanent closure. [10 CFR 60.51(a)(1)]
	B.	The Engineered Barrier Segment shall be designed so that the Engineered Barrier Segment together with the Repository Segment, provides a reasonable expectation (TBD), based upon performance assessments, that the cumulative releases of radionuclides to the accessible environment for 10,000 years after disposal from all significant processes and events that may affect the disposal system shall have a likelihood of less than one chance in 10 of exceeding the quantities calculated according to Table 1 of Appendix A of 40 CFR 191. In addition, is shall have a likelihood of less than on chance in 1,000 of exceeding ten times the quantities calculated according to Table 1 of Appendix A of 40 CFR 191 (TBR). [40 CFR 191.13(a)(TBR)]
3.2.1.7 Off-normal Mode Requirements	A.	The Engineered Barrier structures, systems, and components important to safety shall be designed to maintain control of radioactive waste and radioactive effluents, and permit prompt termination of operations and evacuation of personnel during an emergency in the GROA. [10 CFR 60.131(b)(4)(I)]
	B.	The Engineered Barrier Segment structures, systems, and components important to safety shall be designed to satisfactorily perform their safety functions during and after credible fire or explosion conditions in the GROA. [10 CFR 60.131(b)(3)(I)]

Table 2-1. Applicable Program Requirements (Continued)

EBDRD Section	Para.	Applicable Requirement [and Source]
3.2.2 Radiological Protection 3.2.2.1 General Requirements	A.	The Engineered Barrier Segment, together with the Repository Segment, shall to the extent practicable, be designed and constructed with engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are as low as is reasonably achievable (ALARA). ALARA principles shall be based on the applicable sections of the NRC Regulatory Guides 8.8 and 8.10. [10 CFR 20.1101(b)]
	C.	The Engineered Barrier Segment, together with the Repository Segment, shall provide adequate shielding from radioactive components and high ambient temperatures.
3.2.2.6 Criticality Protection	A.	The Engineered Barrier Segment shall be designed to ensure that a nuclear criticality accident is not possible unless at least two unlikely, independent, and concurrent or sequential changes have occurred in the conditions essential to nuclear criticality safety. Each system shall be designed for criticality safety under normal and accident conditions. The calculated effective multiplication factor must be sufficiently below unity to show at least a five percent margin, after allowance for the bias in the method of calculation and the uncertainty in the experiments used to validate the methods of calculation. [10 CFR 60.131(b)(7)]
	B.	To mitigate the potential for nuclear criticality, the Engineered Barrier Segment shall be designed and constructed to comply with the nuclear criticality requirements specified by DOE Order 6430.1A, 1300-4.
	A.8	The Repository Segment will accommodate the emplacement concept (TBD) selected during Advanced Conceptual Design. [Together with the Engineered Barrier Segment] b. The Repository Segment design will prevent free-liquid-phase water from contacting the waste package during the period from package insertion until repository closure. [Together with the Engineered Barrier Segment] c. The Repository Segment layout will be designed so that a combination of characteristics will assist in keeping liquid water from contacting the waste packages for the first 300 to 1000 (TBV) years after closure. [Together with the Engineered Barrier Segment] d. The Repository Segment layout will also ensure that the design limit temperatures (TBD) for waste forms are not exceeded. [Together with the Engineered Barrier Segment]

Table 2-1. Applicable Program Requirements (Continued)

EBDRD Section	Para.	Applicable Requirement [and Source]
3.2.2.6 Criticality Protection (Continued)	A.11	The Repository Segment will be designed to preclude the potential for nuclear criticality of the stored waste at any time after being received at the Mined Geologic Disposal System.
	A.14	The orientation, geometry, layout, and depth of the underground facility, and the design of any engineered barriers that are part of the underground facility will contribute to the containment and isolation of radionuclides. b. Underground facility configuration (access and drift location, diameter, orientation, geometry, separation and depth will contribute to , or not detract from, the isolation capability of the site. (TBD) [10 CFR 60.133(a)(1)
	A.17	The underground facility will be designed so that the performance objectives will be met taking into account the predicted thermal and thermomechanical response of the host rock, and surrounding strata, and ground water system. [10 CFR 60.133(I)]
	B.3	The Engineered Barrier Segment, together with the repository design and operations, shall include provisions for controlling doses such that when approved operational procedures are followed, the exposure doses specified in 10 CFR 20.1201 for occupational doses, and 10 CFR 20.1301 for individual members of the public, are not exceeded. [10 CFR 20.1201; 10 CFR 20.1301]
	B.4	The Engineered Barrier Segment shall be able to withstand shock (TBD) and vibration (TBD) levels characteristic of handling, emplacement, retrieval, and seismic environments, without adverse impacts on waste containment and isolation capacity.
3.2.3.4 Engineered Barrier Segment - Waste Acceptance Interface Requirements	A.	Waste Acceptance will provide standard HLW meeting the following criteria: 1. Canistered HLW, which is borosilicate glass sealed inside an austenitic stainless steel canister with a concentric neck and lifting flange (TBV). a. Total Length: 3.00 meters (_0.005, -0.020 m)(TBV). b. Diameter: 61.0 cm (+1.5, -1.0 cm)(TBV). c. Weight: Up to 2500 kg (TBV). d. Fill Height: Equivalent to at least 80% of the volume (TBV) of the empty canister. e. Total heat generation rate: Up to 1500 W per canister (TBV) at the year of shipment. f. Waste temperature: Will not have exceeded 400°C (TBV) during transit to ensure the glass transition temperature was not exceeded.

Table 2-1. Applicable Program Requirements (Continued)

EBDRD Section	Para.	Applicable Requirement [and Source]
3.2.3.4 Engineered Barrier Segment - Waste Acceptance Interface Requirements (Continued)		2. Inert cover gas leak rate of the outermost closure will be less than $(10)^{-4}$ atm-cc/sec. 4. Estimated inventory of radionuclides will be (TBD). 5. Canistered waste forms will have a maximum surface gamma dose rate of up to $(10)^{5}$ rem/hr (TBV) and a maximum surface neutron dose rate of up to 10 rem/hr (TBV), at the year of shipment to the Mined Geologic Disposal System.
	C.	WA will provide standard SNF meeting the following criteria: 1.a. Total length: 14 feet, 11 inches or less for the boiling water reactor (BWR) assembly and 14 feet, 10 inches or less for the pressurized water reactor (PWR) assembly (TBV). [10 CFR 961.11, Appendix E, B.1] 1.b. Cross-section: 6 inches x 6 inches or less for BWR and 9 inches by 9 inches or less for PWR (TBV). [10 CFR 961.11, Appendix E, B.1] 1.c. Nonfuel Components: Nonfuel components including, but not limited to, control spiders, burnable poison rod assemblies, control rod elements, thimble plugs, fission chambers, and primary and secondary neutron sources, that are contained within the fuel assembly, or BWR channels that are an integral part of the fuel assembly, which do not require special handling, may be included as part of the SNF delivered for disposal. **Note:** Fuel that does not meet these specifications shall be classified as Nonstandard Fuel-Class NS-2. [10 CFR 961.11, Appendix E, B.2] 1.d. Cooling: The minimum cooling time for fuel is five years. **Note:** Fuel that does not meet this specification shall be classified as Nonstandard Fuel-Class NS-3. [10 CFR 961.11, Appendix E, B.3] 1.e Non-Light water reactor (LWR) Fuel: Fuel from other than LWR power facilities shall be classified as Nonstandard Fuel-Class NS-4. Such fuel may be unique and require special handling, storage, and disposal facilities. [10 CFR 961.11, Appendix E, B.4]

Table 2-1. Applicable Program Requirements (Continued)

EBDRD Section	Para.	Applicable Requirement [and Source]
3.2.3.4 Engineered Barrier Segment - Waste Acceptance Interface Requirements (Continued)		1.f. Consolidated Fuel Rods: Fuel that has been disassembled and stored with the fuel rods in a consolidated manner shall be classified as Nonstandard Fuel Class NS-5. [10 CFR 961.11, Appendix E., B.5] 1.g. Weight: Up to (TBD) kilograms. 2. Temperature will not have exceeded 350°C (TBV) during storage under inert gas. 4. Estimated inventory of radionuclides will be (TBD). 5. Dose rate at shipment will be up (TBD).
3.2.4.6 Design Basis Events	A	An Engineered Barrier Segment design objective shall be to ensure that conservatively estimated consequences of normal operations and credible accidents are limited in accordance with requirements contained in DOE Order 6430.1A, Section 1300-1.4, Guidance on Limiting Exposure of the Public. Events to be considered based on site function and licensing requirements are (TBD).
	B	The probable consequence of design basis accidents involving internally generated missiles or blast effects shall be considered. Such design basis accidents typically involve failure of high-speed rotating machinery, cranes, experimental facilities, high-energy fluid system components, or explosives. Structures required to function following such accidents must be designed to withstand these design basis accidents.
3.7 Engineered Barrier Segment Major Component Characteristics/ requirements	C.	The Engineered Barrier Segment shall be designed so that assuming anticipated processes and events: (A) containment of HLW will be substantially complete (TBD) during the period when radiation and thermal conditions in the Engineered Barrier Segment are dominated by fission product decay (TBD); and (B) any release of radionuclides from the Engineered Barrier Segment shall be a gradual process which results in small fractional releases (TBD) to the geologic setting over long times. [10 CFR 60.113(a)(1)(I)]
	D.	The Engineered Barrier Segment shall be designed, assuming anticipated processes and events, so that containment of radioactive material within the waste packages will be substantially complete (TBD) for a period to be determined by the Nuclear Regulatory Commission, but not less than 300 years nor more than 1,000 years (TBR) after permanent closure of the geologic repository. NRC Staff Position 60-001 [NRC 1990a] indicates that this is a minimum performance requirement, and that credit can be taken in Engineered Barrier Segment and overall repository system performance assessments for containment, provided by design, in excess of 1000 years. [10 CFR 60.113(a)(1)(ii)(A)]

Table 2-1. Applicable Program Requirements (Continued)

EBDRD Section	Para.	Applicable Requirement [and Source]
3.7 Engineered Barrier Segment Major Component Characteristics/ requirements (Continued)	E.	The Engineered Barrier Segment shall be designed, assuming anticipated processes and events, so that the release rate of any radionuclide from the Engineered Barrier Segment following the containment period shall not exceed 1 part in 100,000 per year of the inventory of that radionuclide calculated to be present at 1,000 years following permanent closure, or such other fraction of the inventory as may be approved or specified by the Nuclear Regulatory Commission. This requirement does not apply to any radionuclide which is released at a rate less than 0.1 percent of the calculated total release rate limit. The calculated total release rate limit shall be taken to be 1 part in 100,000 per year of the inventory of radioactive waste, originally emplaced in the underground facility that remains after 1,000 years of radioactive decay. [10 CFR 60.113(a)(1)(ii)(B)]
	F.	The Engineered Barrier Segment shall maintain performance under rock-induced loading (TBD).
	G.	To limit the predicted thermal and thermomechanical response of the host rock and surrounding strata, and groundwater system, the Engineered Barrier Segment configuration and loading shall: 1. Limit borehole wall temperature (if a borehole is used) to 275°C (TBV) 2. Limit the maximum temperature 1 meter into the rock to 200°C (TBV) 3. Limit the TSw3 (vitrophyre tuff) maximum temperature to 115°C (TBV) 4. Limit the maximum ground surface temperature change in the vicinity of the repository to 6°C (TBV) 5. Limit the emplacement drift maximum temperature to 100°C (TBV) 6. Limit the access drift maximum temperature to 50°C (TBV) [10 CFR 60.133(I)]
3.7.1 Waste Package Subsystem Requirements	A.	Packages for SNF and HLW shall be designed so that the in situ chemical, physical, and nuclear properties of the waste package and its interactions with the emplacement environment do not compromise the function of the waste packages or the performance of the underground facility or the geologic setting. [10 CFR 60.135(a)(1)]
	B.	The design of waste packages shall include, but not be limited to, consideration of the following factors: solubility, oxidation/reduction reactions, corrosion, hydriding, gas generation, thermal effects, mechanical strength, mechanical stress, radiolysis, radiation damage, radionuclide retardation, leaching, fire and explosion hazards, thermal loads, and synergistic interactions. [10 CFR 60.135(a)(2)]

Table 2-1. Applicable Program Requirements (Continued)

EBDRD Section	Para.	Applicable Requirement [and Source]
3.7.1 Waste Package Subsystem Requirements (Continued)	C.	The waste packages shall not contain explosive or pyrophoric material or chemically reactive materials in an amount that could compromise the ability of the underground facility to contribute to waste isolation or the ability of the geologic repository to satisfy the performance objectives. [10 CFR 60.135(b)(1)]
	D.	The waste package shall not contain free liquids in an amount that could compromise the ability of the waste package to achieve the performance objectives relating to containment of radioactive waste (because of chemical interactions or formation of pressurized vapor) or that could result in spillage and spread of contamination in the event of a waste package perforation during the period through permanent closure. [10 CFR 60.135(b)(2)]
	E.	Waste packages shall be designed to maintain waste containment during transportation, emplacement, and retrieval. [10 CFR 60.135(b)(3)]
	G.	Processes specified for the fabrication, assembly, closure, and inspection of waste packages shall be based on technology reasonably available at the time of final design. These processes need not be reduced to commercial practice in all applicable details and shall not require significant extensions of the technology.
	H.	The waste package shall contribute to limiting the dispersal of radioactive waste materials in the event of accidents or other dynamic effects.
	I.	Containment of radioactive material within the waste packages shall be substantially complete (TBD) for a period of years (TBD) after permanent closure of the geologic repository. [10 CFR 60.113(a)(1)(ii)(A)]
	J.	The waste package shall meet the following criteria: 1. External dimensions shall be (TBD), with a tolerance of (TBD). 2. Weight shall not exceed (TBD). 3. Surface finish shall be (TBD)
	K.	The waste package shall be transportable, emplaceable, and retrievable by the repository equipment.
3.7.1.1 Waste Form Requirements	A.	All such radioactive wastes shall be in solid form. [10 CFR 60.135(c)(1)]
	B.	Radioactive waste shall be placed in sealed containers. [10 CFR 60.135(c)(1)]
	C.	Particulate waste forms shall be consolidated (for example, by incorporation into an encapsulated matrix) to limit the availability and generation of particulates. [10 CFR 60.135(c)(1)]
	D.	All combustible radioactive wastes shall be reduced to a noncombustible form unless it can be demonstrated that a fire involving the waste packages containing combustibles will not compromise the integrity of other waste packages, adversely affect any structures, systems, or components important to safety, or compromise the ability of the underground facility to contribute to waste isolation. [10 CFR 60.135(c)(1)]

Table 2-1. Applicable Program Requirements (Continued)

EBDRD Section	Para.	Applicable Requirement [and Source]
3.7.1.1 Waste Form Requirements (Continued)	E.	The waste form shall be capable of sustaining normal handling and packaging operational loads without impairing the waste packages's ability to prevent loss of containment.
	F.	The canistered waste form shall be capable of withstanding a drop of (TBD) onto a flat, essentially unyielding surface without breaching.
	G.	Seismic loads (TBD) shall not decrease the postclosure performance of the waste form.
	H.	The waste form shall remain solid during handling, emplacement, and retrieval modes.
3.7.1.2 Waste Container Requirements	A.	The container shall contain the radioactive waste materials during all normal handling and emplacement operations and, in the event of accidents or other dynamic effects, contribute to limiting dispersal of the waste. The container shall also have the mechanical integrity to sustain routine handling and transportation loads (TBD).
	B.	The container shall contribute to the waste package such that containment of the enclosed radionuclides is substantially complete (TBD) during the containment period of not less than 300 to 1,000 years (TBR) after permanent closure of the geologic repository. [10 CFR 60.113(a)(1)(ii)(A)]
	C.	The container shall contribute (TBD) to controlling the release of radionuclides during the period of isolation. [10 CFR 60.135(a)(1)]
	D.	The container shall be designed to limit the amount of liquid water (TBD) allowed to contact the enclosed waste form.
	F.	The container shall maintain lifting and handling capabilities through the loading, emplacement, and retrieval phases. 1. All of the emplaced containers shall remain intact (TBV) and have sufficient strength through the period of retrievability so that the waste package may be removed. 2. The handling and lifting fixtures on all (TBV) of the containers shall maintain their performance to the end of the period of retrievability.

Table 2-1. Applicable Program Requirements (Continued)

EBDRD Section	Para.	Applicable Requirement [and Source]
3.7.1.2 Waste Container Requirements (Continued)	G.	The container shall be designed so that neither its in situ chemical, physical, and nuclear properties, nor its interactions with the waste form and the emplacement environment, compromise the function of the waste package or the performance of the natural barriers or engineered barriers. [10 CFR 60.135(a)(1)]
	H.	The container shall be designed to meet the following criteria: 1. The surface finish shall be suitable for ease of decontamination. 2. The maximum weight shall be (TBD). 3. The envelope shall be (TBD). 4. The acceptance criteria for weld flaws is (TBD).
	I.	The container shall be designed for ease of attaching and sealing the lid to the body of the container after the waste forms are installed.
	J.	If the design requires an inert gas barrier, the container shall be filled with the inert gas (e.g., Ar-He mixture) before final closure. The inert gas shall meet requirements (TBD) pertaining to such issues as fill temperature and pressure, gas purity and closure tightness. Inspection and test requirements shall be (TBD).
3.7.1.3 Internal Structure Requirements	A.	The internal structure shall provide separation of the waste forms such that nuclear criticality shall not be possible unless at least two unlikely, independent, and concurrent or sequential changes have occurred in the conditions essential to nuclear criticality safety. The calculated effective multiplication factor (k_{eff}) must be sufficiently below unity to show at least a five percent margin after allowance for the bias in the method of calculation and the uncertainty in the experiments used to validate the methods of calculation (TBD). [10 CFR 60.131(b)(7)]
	B.	The internal structure of the waste package shall be configured to accommodate the spent fuel waste form, provide mechanical stability of the waste form, and facilitate loading of the waste form into the waste package.
	C.	The internal structure(s) shall be designed to accommodate each configuration of the canisterized and uncanisterized SNF waste form and the canisterized HLW form.
	D.	The internal structure shall be capable of withstanding handling, emplacement, and retrieval loads (TBD).

Table 2-1. Applicable Program Requirements (Continued)

EBDRD Section	Para.	Applicable Requirement [and Source]
3.7.1.3 Internal Structure Requirements (Continued)	E.	The internal structure shall not degrade the performance of components in the waste package which have long term containment requirements.
	F.	The material used for the internal structure shall not cause adverse galvanic reactions inside the waste package.
	G.	The internal structure shall maintain functionality under the thermal and chemical conditions generated by the waste form.
3.7.2 Backfill Requirements	A.	Backfill material shall be chemically compatible with waste package containment and not impair its capability.
	C.	The transfer of heat from the waste package to the geologic setting by the backfill material shall not have an adverse effect on the long term performance of the WP.
	F.	The chemical and mechanical stability of the backfill when subjected to the maximum thermal environment anticipated in the repository shall not have an adverse effect on the long term performance of the waste package.

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3. DESIGN BASIS ASSUMPTIONS AND DEVELOPMENT

This section provides a summary of the plans, assumptions, operational processes, available data, environmental scenarios, and design goals which were used, along with the requirements and standards in Section 2, in the development of the Engineered Barrier Segment configuration items and design bases. The Engineered Barrier Segment performs the containment and isolation function within the confines of the emplacement drift. These configuration items and design bases, which address waste package loading, closing, handling, emplacement, materials selection, criticality control, thermal goals, structural integrity, and performance assessment, are required to design an operational Waste Package Subsystem within the Engineered Barrier Segment.

The Engineered Barrier Segment must meet the applicable regulatory requirements for safe disposal of spent nuclear fuel and vitrified high-level waste in a geologic repository. The Configuration Item architecture for the Engineered Barrier Segment is shown in Figure 3-5. Functional allocation to Engineered Barrier Segment Subsystems is provided in Section 3.3.

As a consequence of historical usage, the term waste package is used frequently throughout this report; however, this usage refers to the Waste Package Subsystem Configuration Item, and not the 10 CFR 60.2 generic definition (both definitions may be found in the glossary in Appenedix A). Waste container is a generic term for a Waste Package Subsystem without the Shielding Subsystem Element and the Packing and Absorbent Materials Subsystem Element (i.e., a waste container is a sealed disposal container with the uncanistered or canistered waste form placed therein [and possibly filler material]) (see Figure 3-5).

The specific requirements for the Engineered Barrier Segment are contained in the *Engineered Barrier Design Requirements Document* (YMP 1994b) as discussed in Section 2 of this volume. In addition, the *Controlled Design Assumptions Document* (CDA) (CRWMS M&O 1995n) provides direction on issues that are either not addressed specifically by the Engineered Barrier Design Requirements Document (EBDRD) or for which program direction has changed and the requirements document has not yet been updated to reflect the additional information.

Using a systems engineering approach, the WPs will be designed as multibarrier systems that meet the regulatory requirements with sufficient margin. The development of a WP design and the associated performance assessment of the Engineered Barrier Segment that meets the regulatory and other design requirements will be accomplished by following the process steps shown in the Engineered Barrier Segment Development Strategy chart (Figure 3-1). This process will be followed for all designs.

The process begins with the development of the design basis, which consists of inputs from many technical and non-technical areas. This includes the definition of environmental scenarios and the definitions of the performance functions, measures, and parameters for each Engineered Barrier Segment barrier. The process will permit the design of one or more options for each barrier of the Engineered Barrier Segment. This initial design provides the basis for setting performance measures

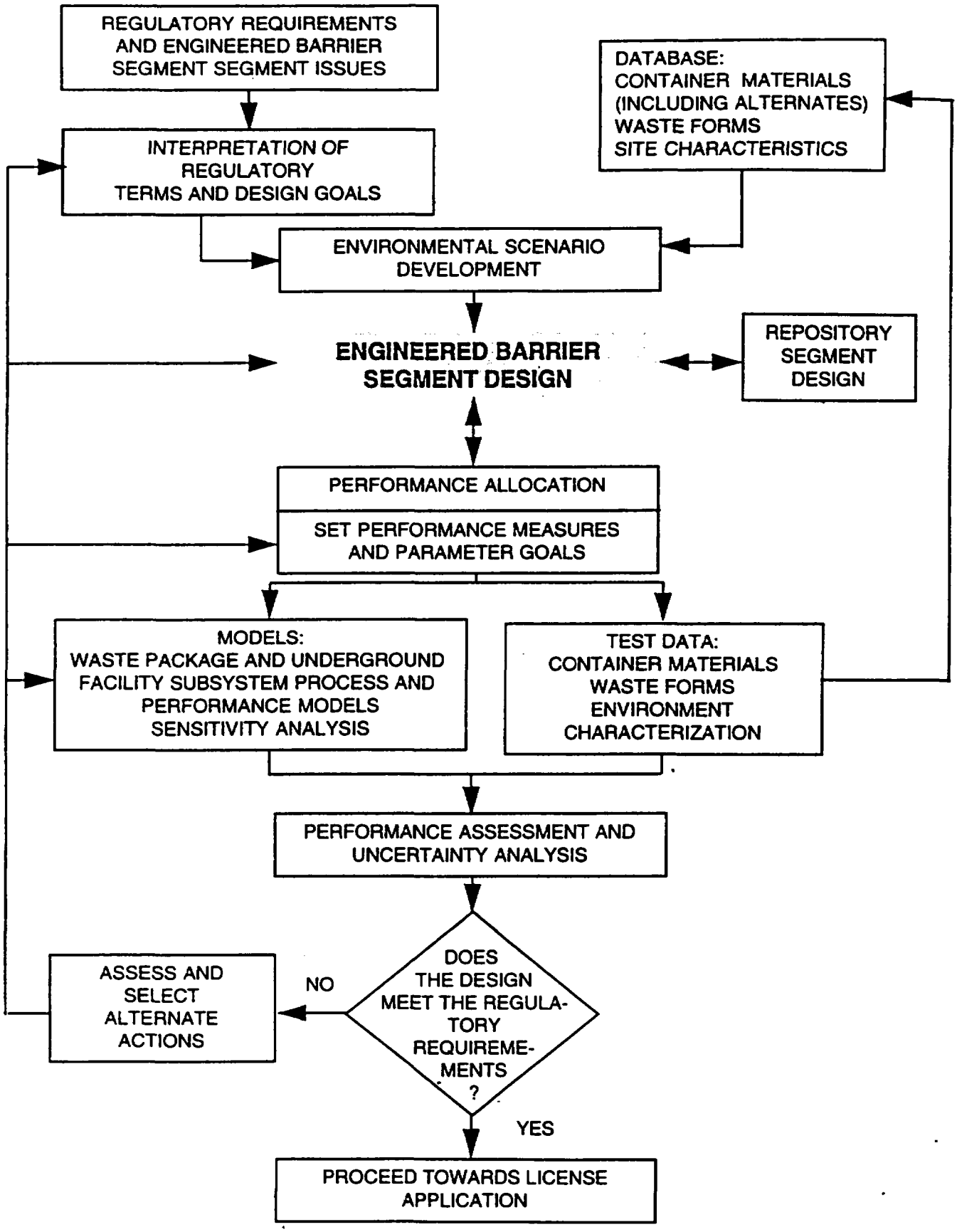


Figure 3-1. Engineered Barrier Segment Development Strategy

and performance parameter goals needed to refine the allocated performance. This in turn determines the test data and models needed to perform an assessment of performance of the WP and Engineered Barrier Segment. The process is interactive and iterative, and is repeated until at least one reference and one alternative design are developed that will meet or exceed the regulatory requirements. The final development of the License Application Design permits license application to proceed.

The concepts presented in this section are in support of the *Waste Package Implementation Plan* (YMP 1995) and are based on the *Engineered Barrier Design Requirements Document* (YMP 1994b). The evaluations and results support the goals of the WP designs and process as defined in the plan and requirements documents. The CDA Document (CRWMS M&O 1995n) provide the MGDS Advanced Conceptual Design (ACD) teams with a common list of assumptions. The CDA Document is an accumulation of engineering assumptions and key assumptions compiled by the CRWMS M&O organization. The EBDRD describes the functions to be performed by, establishes the requirements for the Engineered Barrier Segment, and provides the design requirements that affect Engineered Barrier design features. In addition, specific design goals have been established to provide quantitative design guidance and to minimize the potential risk to the public. The design requirements and design goals are discussed in the following sections.

3.1 PLANS FOR WASTE PACKAGE DESIGN DEVELOPMENT

WP development has been accomplished in accordance with four plans, which are summarized in this section.

The four plans are WP program plans covering the period from program initialization through the completion of the License Application Design and the submission of a repository license application to the NRC. Technical Implementation Plans were prepared yearly and described the planned activities for the WP program in next fiscal year (FY). For FY 1996, these planned activities included WP coordination and planning, WP requirements, WP design (includes multi-purpose canister [MPC] design support and ACD and performance analysis), WP materials testing and modeling, the WP Closure Weld development task, and waste form testing and modeling. All of these WP program activities support YMSCO objectives for FY 1996.

The four plans are scheduled to be integrated in the Yucca Mountain Site Characterization Project (YMP) System Engineering Management Plan which will then be the controlling plan.

As part of the FY 1996 effort to consolidate plans, a Waste Package Development Strategy document will be prepared that reflects the consolidation of the *Waste Package Plan* (YMP 1993) and the *Waste Package Implementation Plan* (YMP 1995). The Waste Package Development Strategy document will be consistent with the *Civilian Radioactive Waste Management Program Plan* (DOE 1994a), which represents the most current baseline of program objectives and activities, and the YMP Work Breakdown Structure 1.2.2 Long Range Plan (CRWMS M&O 1994a). The following defines the plans.

3.1.1 Waste Package Plan

Waste Package Plan, YMP/90-62 (YMP 1993)

This plan describes "...the waste package program of the YMP and (establishes) the technical plan against which overall progress can be measured. The plan provides guidance for execution and describes the essential elements of the WP program, including the objectives, technical plan and management approach. The plan covers the time period up to the submission of a repository license application to the Nuclear Regulatory Commission."

3.1.2 Waste Package Implementation Plan

Waste Package Implementation Plan, YMP/92-11Q (YMP 1995)

This plan is a detailed WP materials and component testing, modeling, and design implementation plan that supports the YMP *Waste Package Plan*. The strategy for the implementation of these activities is to use an interactive and iterative approach with performance assessment to determine whether the design meets the requirements with sufficient margin. Thus, the plan provides the bases for the design and performance assessment of the WP, and the requirements for the Engineered Barrier System that will demonstrate that they meet or exceed the regulatory requirements.

This plan prescribes work that affects items on the YMP *Q-List* (YMP 1994c) and changes to this plan shall be controlled in accordance with applicable YMP quality assurance procedures. The plan will be revised as necessary to reflect changes in upper-tier documents from the YMP, including the *Waste Package Plan*; and from the Office of Civilian Radioactive Waste Management; and to reflect requirements of the *Engineered Barrier Design Requirements Document* (YMP 1994b). The plan also interfaces with other YMP documents. The plan covers the period up to the submission of a repository license application to the NRC.

3.1.3 Waste Package Engineering Interface Plan

Waste Package Engineering Interface Plan (CRWMS M&O 1993k)

This document presents the purpose and requirements of the *Waste Package Engineering Interface Plan*, followed by a description of the WP conceptual design and WP performance requirements. The Waste Package Development Department activities are described, followed by a listing of organizations with which the Design Department will conduct interfacing activities. The final section of the document describes the established and anticipated interface activities.

The purpose of the *Waste Package Engineering Interface Plan* is to outline the interface requirements between the Waste Package Development Department and other affected CRWMS M&O organizations and affected outside organizations. This plan identifies the interface relationships and type of information expected to pass between the department and the interface organizations, and the form such information should take. Information forms may

be interface drawings, interface specifications, technical information reports, technical evaluation reports, computer software codes, databases/libraries, etc. The Waste Package Development Department will issue Interface Drawings and Specifications to the affected organizations.

Development and control of the Interface Drawings and Specifications which will be derived from this Interface Plan will be quality-affecting work and will be subject to the requirements of the DOE OCRWM *Quality Assurance Requirements and Description* (DOE 1995c). These requirements are delineated in the CRWMS M&O Quality Administrative Procedures and associated Implementing Line Procedures.

3.1.4 Waste Package Engineering Development Task Plan

Waste Package Engineering Development Task Plan (CRWMS M&O 1993b)

The purpose of the Waste Package Engineering Development program is to develop nuclear waste disposal container designs that the NRC will find acceptable and will license for disposal of SNF and vitrified DHLW within a tuff repository.

This *Waste Package Engineering Development Task Plan* satisfies the requirement specified in the *Waste Package Implementation Plan* (YMP 1995) for a description of planned activities associated with Work Breakdown Structure 1.2.2.4.2.

The purpose of this document is to provide a detailed WP materials and component testing, modeling, and design implementation plan that supports the YMP *Waste Package Plan* (YMP 1993). The strategy for the implementation of these activities is to use an interactive and iterative approach with performance assessment to determine whether the design meets the requirements with sufficient margin. Thus, the plan provides the bases for the design and performance assessment of the WP and the requirements for the Engineered Barrier System that will demonstrate that they meet or exceed the regulatory requirements. The principal regulatory requirements are the technical requirements for repository operation of the NRC as given in 10 CFR 60, *Disposal of High-Level Radioactive Wastes in Geologic Repositories*, and the environmental standards of the Environmental Protection Agency that limit offsite releases as given in 40 CFR 191, *Environmental Standards for the Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes*. [40 CFR 191 has been remanded. The revised environmental standards will include input from the National Academy of Sciences as mandated by the Comprehensive National Energy Policy Act of 1992.]

3.2 ASSUMPTIONS

The WP/Engineered Barrier Segment Subsystem designs are based on assumptions documented in the CDA Document (CRWMS M&O 1995n) in addition to the EBDRD requirements discussed in Section 2. The CDA Document is a tool designed to facilitate the development of the Mined Geologic Disposal System (MGDS) ACD. The CDA Document is a key element in the ACD approach, which uses management decisions and/or assumptions, as necessary, based on the best available information or engineering judgment, to advance the design. The CDA document contains

these assumptions, as well as rationale for the assumptions and references to plans and schedules to substantiate the assumptions. The assumptions are considered to be of indeterminate quality and therefore will require substantiation to validate, qualify, or determine their sufficiency for meeting overall system requirements.

The assumptions have been given two-part identifiers. If the assumption is not a requirement assumption, the first part of the identifier categorizes the assumption and the second part provides a sequence number within the category. Key assumptions are identified by the word "Key" and a number. If the assumption is a requirement assumption, then the first part identifies the source document for the requirement (e.g., EBDRD) and the second part is the paragraph identifier as found in the requirement document (e.g., 3.7.G., 3.2.4.6.A). Design concept assumptions are identified by the abbreviation "DCWP" and a number.

3.2.1 Key Assumptions

An important subset of the assumptions is identified as key assumptions. The basic rationale for identifying an assumption as a key assumption is whether the assumption is a highly controversial issue lacking a clear consensus among DOE officials and program participants, or whether the assumption involves more than one program element.

This section provides a list of key assumptions that are either already requirements in the *Engineered Barrier Design Requirements Document* (YMP 1994b) or are to be considered for the development of requirements in that document. The Assumptions Rationale Sheets for the key assumptions are provided in Section 6.1 of the CDA document (CRWMS M&O 1995n). Other requirements refer to the *Mined Geologic Disposal System Requirements Document* (MGDS-RD) (DOE 1995a). Only those assumptions that directly apply to the Engineered Barrier Segment/WP Subsystem design are listed here, the reader is referred to the reference for the complete list (CRWMS M&O 1995n).

Key 001 <u>Cask Arrival Scenario</u>

> The cask arrival scenario at the MGDS is as indicated in Table 3-3 in the MGDS-RD. Rail shipments total approximately 5,000 (MPC = 4,400, HLW = 600) with a maximum of three train cars per SNF shipment and a maximum of five train cars per HLW shipment, with one transportation cask per rail car. Truck shipments total approximately 1,000; all uncanistered SNF.

The cask arrival schedule will affect the heat produced at emplacement for the SNF assemblies. However, this effect is captured in the assumption of average SNF characteristics in CDA assumption Key 004. Therefore, this cask arrival schedule is indirectly used in the calculation of WP emplacement thermal response of Section 6.2.1.1. This assumption is utilized as an uncanistered SNF WP design consideration.

Key 002 Waste Form Arrival Scenario

The waste form arrival scenario at the MGDS is as indicated in Table 3-3 in the MGDS-RD.

The waste form arrival schedule will affect the heat produced at emplacement for the SNF assemblies. However, this effect is captured in the assumption of average SNF characteristics in CDA assumption Key 004. Therefore, this cask arrival schedule is indirectly used in the calculation of WP emplacement thermal response of Section 6.2.1.1. This assumption is an interface assumption that is utilized as a uncanistered SNF WP design consideration.

Key 003 Waste Package Emplacement Scenario

The WP emplacement scenario at the MGDS for the reference thermal load is as indicated in the table included in Key 003, which is compatible with Table 3-3 in the MGDS-RD. The table is compatible with the tables in Key Assumptions 001 and 002 for higher thermal loads. Total commercial SNF – 63,000 MTU in about 9,000 MPC and about 200 uncanistered fuel WPs.

The WP emplacement rate scenario may impact the ability to achieve low thermal loads. The thermal loading range to be considered is dictated by CDA assumption Key 019. This assumption is an interface assumption which is utilized as an UCF WP design consideration.

Key 004 Average SNF Characteristics

The average SNF characteristics upon receipt at the Repository and based on the oldest-fuel-first acceptance strategy, no Monitored Retrievable Storage, deferred dry storage, derated canisters, and four truck sites:

26.4 years old with 39.65 GWd/MTU burnup and 3.68 weight percent initial enrichment (PWR).

26.1 years old with 31.19 GWd/MTU burnup and 2.97 weight percent initial enrichment (BWR).

The table in the CDA provides the total repository emplacement decay heat by WP type as a function of time.

This assumption affects the WP heat output in the emplacement scale thermal evaluation described in Section 6.2.1.1. However, at the time the emplacement thermal evaluation was performed, a previous revision of the CDA was in effect. In the previous revision, assumption Key 004 called for average PWR SNF characteristics of 22 years old with 42.2 GWd/MTU burnup based on a youngest-fuel-first acceptance strategy (instead of the current oldest fuel first). The oldest-fuel-first waste stream is slightly cooler than youngest-fuel-first and would result in slightly lower temperatures. Future calculations of emplacement scale temperatures will use the current CDA characteristics. WP design calculations are based upon the design basis fuel selection as specified in Section 5.

Key 005 Total HLW

Total HLW: 7,000 MTU equivalent in about 13,000 commercial and DHLW canisters containing an immobilized waste form (e.g., vitrified glass). Potential sources include waste resulting from reprocessing at the Savannah River Site, Hanford Reservation, Idaho National Engineering Laboratory, and the West Valley Site.

The Savannah River/West Valley design of HLW glass canisters are the basis for the geometry, as used in Section 6, and for the DHLW WP design analysis presented in Section 6.4. The Savannah River pour canisters, as indicated in Appendix B, provide the basis for the DHLW WP design.

Key 008 No Rod Consolidation

Rod consolidation will not be performed at the MGDS.

This assumption is inherent in the WP designs as presented in Appendix B. Intact PWR and BWR SNF assemblies are assumed in the WP design evaluations of the CF WP and uncanistered SNF WPs in Sections 6.3 and 6.5 and in the determination of SNF assembly effective thermal conductivity in Section 6.2.1.2.

Key 009 Burnup Credit

Will receive credit for burnup.

This assumption is inherent in the WP designs as presented in Appendix B. Burnup Credit is addressed in Sections 6.2.2.1, 6.2.2.3, 6.3.5 and 6.5.3.

Key 011 Horizontal In-drift Emplacement

WPs will be emplaced in-drift in a horizontal mode.

This assumption is inherent in the WP designs as presented in Appendix B and has made the consideration of larger waste packages possible as discussed in Section 6. This assumption is also addressed in Section 4 to address material selection concerns and is inherently part of the WP design analysis presented in Section 6.2, 6.3, 6.4, and 6.5.

Key 013 No Human Entry in Emplacement Drifts Containing Waste Packages

No human entry is planned in an emplacement drift while WPs are present. The waste emplacement/retrieval equipment may use robotics and/or remote control features to perform operations and monitoring within the emplacement drifts. Under off-normal conditions, human entry will be considered if protection to the workers can be provided.

The absence of humans in the loaded emplacement drifts averts any requirement to maintain temperatures suitable for human exposure in the emplacement drifts and reduces the radiation shielding requirements of any WP design. This assumption is addressed in Sections 6.2, and 6.3.6, 6.4.4, and 6.5.4.

Key 016 Retrievability Period

> The repository will be designed for a retrievability period of up to 100 years after initiation of emplacement.

This assumption is an interface assumption which is utilized as a WP design consideration. This assumption is inherent in the WP designs as presented in Appendix B and is discussed in Sections 6, 7, and 5.

Key 017 Reasons for Retrieval

> Retrieval of emplaced waste may be performed for the following reasons:
>
> Failure in site, WP, or some other system causing an unreasonable risk to public health and safety. The determination that recovery of valuable resources from the SNF is necessary.

This assumption is an interface assumption which is utilized as a WP design consideration and is a companion to Key 016. This assumption is inherent in the WP designs as presented in Appendix B and is discussed in Sections 6, 7, and 5.

Key 019 Thermal Load Range

> Surface, subsurface and waste package designs will be based on a reference thermal load of 80-100 MTU/acre. The reference thermal load for the revised MGDS ACD is 83 MTU/acre.

Thermal loadings consistent with this range have been selected for the thermal evaluations of CF concepts, DHLW, and uncanistered SNF WPs throughout Section 6. The thermal loadings considered were 25, 83, and 100 MTU/acre. Rationale for these selections are given in Sections 6.2.1.1.1 and 6.2.1.1.2. Section 4 provides a discussion of the effects of near-field relative humidity as it relates to material selection.

Key 022 Repository Horizon

> For the reference thermal loading of 80-100 MTU per acre, the repository horizon will be located mainly in the TSw2 geologic unit within the primary area.

The repository horizon is assumed to lie within the TSw2 geologic unit in the emplacement scale thermal evaluation presented in Section 6.2.1.1. This assumption is inherently part of all of the Engineered Barrier Segment/WP analysis presented in Section 6.

Key 023 Subsurface Fault Stand off

To the extent practical, repository openings will be located to avoid Type 1 faults. For unavoidable Type 1 faults that intersect emplacement drifts, allow a 15 m standoff from the edge of the fault zone to the nearest waste package.

Avoidance is assumed to be adequate by using a 60 m offset from the main trace of a fault at the repository level. (Exception: 120 m stand off should be used on the west side of the Ghost Dance Fault because the Exploratory Studies Facility Topopah Spring Main drift will be excavated before the Ghost Dance Fault characteristics are fully investigated.)

Where avoidance cannot be achieved, for Type I faults that intersect emplacement drifts, allow 15 m standoff distance of emplaced waste packages from the edges of the fault zone.

Any stand off of WPs from faults will affect the local thermal loading of the repository in that region. However, no calculation of this affect on thermal loading is included in this volume. Future evaluations will consider the effects of non-homogeneous spacing of the WPs in the emplacement drifts. Also, consideration of this assumption is also discussed in Section 7 concerning the development of the probabilistic approach to determine rock drop accident scenarios.

Key 031 Waste Package Containment Barrier Shielding

Waste Package containment barriers will provide sufficient shielding for protection of WP materials from radiation enhanced corrosion.

Individual waste packages will not provide additional shielding for personnel protection.

Additional shielding for personnel protection will be provided on the subsurface transporter and in surface and subsurface facilities.

This assumption is inherent in the WP designs as presented in Appendix B. WP designs were driven mainly by performance assessment concerns, structural strength, thermal goal, and criticality concerns and therefore this assumption is inherently part of the WP design analyses presented in Section 6.2, 6.3, 6.4, and 6.5. Transporter shielding issues and analysis is provided in Sections 6.3.6, 6.4.4, and 6.5.4.

Key 036 Substantially Complete Containment (10 CFR 60.113)

Substantially complete containment, as referred to in 10 CFR 60.113, will not be defined quantitatively.

This assumption is discussed and inherently part of the analysis presented in Sections 6, 7, and 8. In each case the term is not quantitatively defined.

Key 037 Expected Waste Package Lifetime

Mean waste package lifetime will be well in excess of 1,000 years.

Sections 4, 7, and 8 consider the material selection, probabilistic degradation analysis, and performance assessment issues which will impact the WP design in order to meet the stated assumption. The analysis provided in Section 6 uses this assumption based upon the discussions in Sections 4, 7, and 8 to assume that the WP will maintain its geometry and material characteristics for 1,000 years.

Key 038 Waste Packages Breached at 1,000 Years

The fraction of WPs breached at 1,000 years shall be less than 1 percent.

This assumption is an assumption concerning design and performance goals of the WP and is utilized as a WP design consideration. This assumption is a companion to Key 037 and, likewise, is discussed in Sections 4, 7, and 8.

Key 039 Criticality Control Period

Criticality Control Period is indefinite or until trends indicate that risk will continue to decrease with time out to 1,000,000 years.

Isotopic concentration time effects are investigated out to 1 million years in support of this assumption in the criticality analysis presented in Sections 6.3.5, 6.4.3, and 6.5.3. Discussion of the method for a probabilistic approach to determine degraded WP configurations for criticality analysis is provided in Section 7. The criticality analysis method is still being developed (CRWMS M&O 1995av).

Key 046 Backfill in Emplacement Drifts

Current design assumes no backfill in emplacement drifts. Options for backfill will be considered based on ongoing and future backfill studies.

No backfill was assumed in the emplacement drifts for the base-case emplacement scale thermal evaluations of Sections 6.2.1.1.1 and 6.2.1.1.2. However, the effect of drift backfill of varying thermal conductivities was investigated in Section 6.2.1.1.3. The performance assessment aspects of backfill is discussed in Section 8.

Key 054 Normal Waste Handling Building Capability - (No Filler Material)

The Design for the Waste Handling Buildings MPC standard handling operations is based on no capability to add filler material to MPCs at the Repository. The addition of filler material to first procurement MPCs will be performed as an off-normal operation per Key Assumption 053.

(Key Assumption 053 is listed in Table 4-1 of Section 4, Volume 2 of this report.) The addition of filler material is discussed in Section 9.5.4.2. Specific analysis for WP which have had filler material added has not been performed as part of the conceptual design, but will be considered as part of future design analysis.

3.2.2 Requirements Assumptions (Engineered Barrier and Repository Design Requirements Documents)

The requirement assumptions include requirements in the *Engineered Barrier Design Requirements Document* (YMP 1994b) and the *Repository Design Requirements Document* (YMP 1994d) that currently carry a TBV or TBD qualifier and that were reviewed for impact on ACD. The Assumption Rationale Sheets for these assumptions are provided in Sections 6.2 of the CDA document (CRWMS M&O 1995n). Only those assumptions which directly apply to the Engineered Barrier Segment/WP design are listed here, the reader is referred to the reference for the complete list (CRWMS M&O 1995n).

EBDRD 3.2.3.4.C.1.g Spent Nuclear Fuel Weight

Waste Acceptance will provide standard SNF meeting the following criteria:

Spent fuel waste form will be the high-level radioactive waste and any encapsulating or stabilizing matrix, such as cladding that is associated with spent fuel.

Weight: Up to 887 kg per PWR assembly and 332 kg per BWR assembly.

The assumption is inherent in the WP designs presented in Appendix B, but, are discussed in Sections 6.3.4 and 6.5.2. The values quoted here are considered maximum values, the structural analysis presented in this volume has used more representative values.

EBDRD 3.2.3.4.C.4 Spent Nuclear Fuel Reference Radionuclide Inventory

Inventory of radionuclides will be based on the Characteristics Data Base (ORNL 1993b).

This assumption is used to support performance assessment, which is discussed in Section 8, and to calculate SNF heat loads for the thermal analysis presented in Section 6.

EBDRD 3.2.3.5 Engineered Barrier/Transportation Physical Interfaces

The Engineered Barrier System interfaces with transportation are with the MPC portion of the Transportation System element.

The CF disposal design and interface issues are discussed in Section 6.5.

EBDRD 3.2.4.5 Shielding

Shielding criteria shall be as defined in Key Assumption 031.

This assumption is addressed in the same manner as Key Assumption 031 in the previous section.

EBDRD 3.2.4.6.A Engineered Barrier Segment Design Objective

An Engineered Barrier Segment design objective shall be to ensure that conservatively estimated consequences of normal operations and credible accidents are limited in accordance with requirements contained in DOE Order 6430.1A, Section 1300-1.4, Guidance on Limiting Exposure of the Public.

This assumption is inherent in the WP designs presented in Appendix B and strongly influences the WP/Engineered Barrier Segment design. The subject is not addressed definitively in this volume, much of these requirements will be addressed as the WP designs approach the final design stage. However, the performance criteria and evaluation methods are discussed in Sections 6, 7 and 8.

EBDRD 3.2.5.1.2.B.1 Engineered Barrier Segment Reliability

Reliability of the Engineered Barrier Segment shall be as follows:

Waste Package - The probability of failure of an individual WP during the preclosure phase should be demonstrated to be less than 10^{-6}, based on credible hazards.

This assumption is inherent in the WP designs presented in Appendix B and strongly influences the WP/Engineered Barrier Segment design. The subject is not addressed definitively in this volume, much of these requirements will be addressed as the WP designs approach the final design stage. However, the performance criteria and evaluation methods are discussed in Sections 7 and 8.

EBDRD 3.2.5.4 Engineered Barrier Segment Maintainable Preclosure Service Life

Engineered Barrier Segment structures, systems, and components shall be designed for a maintainable preclosure service life of at least 150 years following first emplacement of waste or the period of time authorized by the license granted by the NRC in accordance with the provisions of 10 CFR 60.3.

This assumption is inherent in the WP designs presented in Appendix B and strongly influences the WP/Engineered Barrier Segment design. The subject is not addressed definitively in this volume, many of these requirements will be addressed as the WP designs approach the final design stage. However, the performance criteria and evaluation methods are discussed in Sections 7 and 8.

EBDRD 3.7.D Engineered Barrier Segment Minimum Performance

> The Engineered Barrier Segment shall be designed, assuming anticipated processes and events, so that containment of radioactive material within the WPs will be substantially complete for 1,000 years (with less than 1 percent of the WPs failing within 1,000 years after permanent closure of the geologic repository) and with a mean WP lifetime well in excess of 1,000 years.

This assumption is a companion to Key Assumption 037 and Key Assumption 038, is inherent in the WP designs presented in Appendix B, and strongly influences the WP/Engineered Barrier Segment design. The subject is not addressed definitively in this volume; many of these requirements will be addressed as the WP designs approach the final design stage. Sections 4, 7 and 8 consider the material selection, probabilistic degradation analysis, and performance assessment issues which will impact the WP design in order to meet the stated assumption. The analysis provided in Section 6 uses this assumption based upon the discussions in Sections 4, 7 and 8 to assume that the WP will maintain its geometry and material characteristics for 1,000 years.

EBDRD 3.7.F Rock-Induced Waste Package Loading

> The WP must be able to withstand a uniform external pressure of 0.50 MPa and a dynamic load of 50 kN and still maintain structural integrity.

This assumption is addressed in Sections 6.3, 6.4, and 6.5.

EBDRD 3.7.G.2 Vertical Borehole Rock Mass and In-Drift Wall Temperatures

> To limit the predicted thermal and thermomechanical response of the host rock and surrounding strata and groundwater system, the Engineered Barrier Segment configuration and loading shall:
>
> Keep emplacement drift wall temperatures less than 200°C.

Emplacement drift wall temperatures are calculated and compared to this thermal goal in Section 6.2.

EBDRD 3.7.G.3 TSw3 Temperature Limit

> To limit the thermal and thermomechanical response of the host rock and surrounding strata and groundwater system, the Engineered Barrier Segment configuration and loading shall:
>
> Limit the TSw3 (basal vitrophyre) maximum temperature to less than 115°C.

TSw3 rock unit temperatures are calculated and compared to this thermal goal in Section 6.2.

EBDRD 3.7.G.4 Ground Surface Temperature Rise Limit

To limit the thermal and thermomechanical response of the host rock and surrounding strata and groundwater system, the Engineered Barrier Segment configuration and loading shall:

Limit the maximum ground surface temperature change in the vicinity of the repository to 2°C.

Repository scale temperatures are calculated in Section 6.2.1.1, however, the ground surface is considered a constant temperature boundary condition in this analysis. Ground surface temperature changes were not calculated in this volume. Calculations of this thermal response will be discussed in future volumes.

EBDRD 3.7.G.6 Access Drift Temperature Limit

To limit the thermal and thermomechanical response of the host rock and surrounding strata and groundwater system, the Engineered Barrier Segment configuration and loading shall:

Limit the access drift wall rock maximum temperature to 50°C during preclosure.

Access drift temperatures were not calculated for this volume, and they do not affect the design of the WP.

EBDRD 3.7.1.I Waste Package Substantially Complete Containment

Containment of radioactive material within the WPs shall be substantially complete for at least a thousand years after permanent closure of the geologic repository (i.e., fewer than 1 percent of the WPs shall be breached within the first 1000 years after permanent closure of the geologic repository).

This assumption is a companion to Key Assumption 037 and Key Assumption 038, is inherent in the WP designs presented in Appendix B, and strongly influences the WP/Engineered Barrier Segment design. The subject is not addressed definitively in this volume; many of these requirements will be addressed as the WP designs approach the final design stage. Sections 4, 7 and 8 consider the material selection, probabilistic degradation analysis, and performance assessment issues which will impact the WP design in order to meet the stated assumption. The analysis provided in Section 6 uses this assumption based upon the discussions in Sections 4, 7 and 8 to assume that the WP will maintain its geometry and material characteristics for 1,000 years.

EBDRD 3.7.1.J.1 and 3.7.1.2.H.3 External Dimensions for Waste Package

The WP shall meet the following criteria:

External dimensions shall not exceed:

Outer Diameter:	1,850 mm
Outer Length:	5,850 mm

The designs presented in Appendix B meet this design goal. Section 9 also provides the external dimensions for the WP designs.

EBDRD 3.7.1.J.2 Waste Package Maximum Weight

The WP shall meet the following criteria

Weight shall not exceed 69,000 kg (this excludes filler material).
The internal WP/MPC filler material, if required, is estimated to add a maximum of 24,000 kg.

This assumption is discussed in Sections 6 and 9. Also, the structural analysis presented in this volume is based upon the masses provided in Section 6.1.

EBDRD 3.7.1.1.F Waste Package Drop Tolerance

The WP, after being sealed, shall be capable of withstanding a 2 meter drop onto a flat, essentially unyielding surface without breaching.

This design goal is addressed in 6.

EBDRD 3.7.1.1.G Post Closure Seismic Loads on Waste Package

WP seismic design for postclosure (through the substantially complete containment phase) performance shall be compatible with Design Basis Earthquake for surface and subsurface repository. Potential Repository peak accelerations for design are currently estimated at 0.75 g for surface and 0.4 g for subsurface.

The subject is not addressed definitively in this volume, many of these requirements will be addressed as the WP designs approach the final design stage. However, an indication of the structural performance under the indicated loads is presented in Section 6

EBDRD 3.7.1.2.A Waste Package Handling and Transportation Loads

The WP shall have the mechanical integrity to sustain static loads of 25 kN during routine handling and transportation.

This design goal is addressed in Section 6. Dynamic structural analyses have been performed in which the loads exceeded the 25 kN value. The analyses showed that no internal structural deformation resulted.

EBDRD 3.7.1.2.B Container Substantially Complete Containment

> The container shall contribute to the WP such that containment of the enclosed radionuclides is substantially complete for 1,000 years (with less than 1 percent of the WPs failing within 1,000 years after permanent closure of the geologic repository) and with a mean WP lifetime well in excess of 1,000 years.

This assumption is a companion to Key Assumption 037, Key Assumption 038, and EBDRD 3.7.1.I, is inherent in the WP designs presented in Appendix B, and strongly influences the WP/Engineered Barrier Segment design. The subject is not addressed definitively in this volume; many of these requirements will be addressed as the WP designs approach the final design stage. Sections 4, 7 and 8 consider the material selection, probabilistic degradation analysis, and performance assessment issues which will impact the WP design in order to meet the stated assumption. The analysis provided in Section 6 uses this assumption based upon the discussions in Sections 4, 7 and 8 to assume that the WP will maintain its geometry and material characteristics for 1,000 years.

EBDRD 3.7.1.2.C Controlled Release During Period of Isolation

> The container shall contribute to controlling the release rate of radionuclides during the period of isolation.

The assumption is addressed in the discussion of material selection in Section 4 in that corrosion resistance is considered. Corrosion resistance is important in determining the release rate of radionuclides during the period of isolation. Barrier life is also discussed in Section 8 to address performance assessment issues.

EBDRD 3.7.1.2.D Container to Limit Liquid Water Contact

> The container shall be designed to limit the amount of liquid water allowed to contact the enclosed waste form consistent with the requirement that at most, 1 percent of the WPs will be breached at 1,000 years and that the mean time to breaching is well in excess of 1,000 years.

The possibility of liquid water contact with the WP will depend on the local thermal conditions around each WP. Thermal evaluations resulting in the possibility of water contacting the WP surface are discussed in Section 6. No thermo-hydrological calculations were performed in this volume. The assumption is also addressed in the material selection discussion in Section 4. Corrosion resistance is vital to determining the lifetime of the WP and the fraction of WPs breached at 1000 years. Barrier life is also discussed to address performance assessment issues.

EBDRD 3.7.1.2.H.2 Disposable Container Maximum Weight

> The waste package disposal container shall meet the following criteria:
>
> Weight of empty container shall not exceed 32,000 kg.

This design goal is addressed in Section 6.1. Tables of the tare (empty) mass for all of the WP designs are provided.

EBDRD 3.7.1.3.A Criticality Control Requirement

> The internal structure shall provide separation of the waste forms such that nuclear criticality shall not be possible unless at least two unlikely independent, and concurrent or sequential changes have occurred in the conditions essential to nuclear criticality safety.
>
> The calculated effective multiplication factor (k_{eff}) must be sufficiently below unity to show at least a 5 percent margin after the allowance for the bias in the method of calculation and the uncertainty in the experiments used to validate the method of calculation.

This design requirement is addressed by the criticality analyses reported in Section 6. The criticality evaluation method is currently being developed and is discussed in Section 7 and *Disposal Criticality Analysis Technical Report* (CRWMS M&O 1995av).

EBDRD 3.7.1.3.D Waste Package Internal Structure Loads

> For ACD, the loads imposed on the internal structure are similar to the WP loads.
>
> Static load of 25,000 N, handling.
> Withstand a drop of 2 meters.

This assumption is a companion to EBDRD 3.7.1.2.A and EBDRD 3.7.1.1.F. These design goals are addressed in Section 6. Dynamic structural analyses have been performed in which the loads exceeded the 25 kN value. The analyses showed that no internal structural deformation resulted.

EBDRD 3.7.2.B Backfill Permeability

> (In the case backfill is used)
> The requirements for backfill placed in other underground openings in the repository are addressed in the RDRD (YMP 1994d).
>
> B. The backfill shall not have an adverse effect on the long-term performance of the WP.

The effect of backfill permeability on long-term performance of the WP is not definitively addressed in this volume. However, drift backfill is discussed in Section 8 including issues concerning performance assessment and the effect on temperatures of drift backfill having varying thermal conductivities is investigated in Section 6.

EBDRD 3.7.3 Emplacement Hardware Requirements

> The emplacement hardware requirements are for hardware used to support and protect the emplaced WPs. Examples of emplacement hardware are a pedestal under the WP for the in-drift emplacement concept and a carriage and rail system for the horizontal opening concept.

Emplacement hardware does not include ground support hardware, which is part of the Repository Segment. Emplacement hardware requirements will be added during and after ACD.

Emplacement hardware around the emplaced WP were not modeled in the thermal or structural evaluations in this volume. To be conservative, the conductance of pedestals, rail cars, or other supports were neglected for the thermal evaluations. Future analyses will consider the impact of WP emplacement hardware on near-field temperatures and will be incorporated into the structural evaluations as the hardware is designed.

RDRD 3.2.1.6.C Disposal System Postclosure Performance (40 CFR 191)

The disposal system shall be designed to provide a reasonable expectation, based on PAs, that the cumulative releases of radionuclides to the accessible environment for 10,000 years after disposal from all significant processes and events that may affect the disposal system shall have a likelihood of less than one chance in 10 of exceeding the quantities calculated according to Table A-1 of Appendix A of 40 CFR 191; and have a likelihood of less than one chance in 1,000 of exceeding 10 times the quantities calculated according to Table A-1 of Appendix A of 40 CFR 191.

[MGDS-RD 3.2.1.7.C][40 CFR 191.13(a)]

This requirement is not definitively addressed in this volume. However, its impact and method of evaluation is discussed in Sections 7 and 8 including performance assessment issues and probabilistic evaluation methods.

RDRD 3.2.3.2.2.A.11a Repository Layout to Limit Waste Package - Water Contact

The repository layout shall be designed so that a combination of characteristics will limit the amount of liquid water allowed to come into contact with the WPs consistent with the requirement that at most, 1 percent of the WPs will be breached at 1,000 years and that the mean time to breaching is well in excess of 1,000 years.

This assumption is a companion to Key Assumption 037, Key Assumption 038, EBDRD 3.7.1.2.B, and EBDRD 3.7.1.2.D, is inherent in the WP designs presented in Appendix B, and strongly influences the WP/Engineered Barrier Segment design. The subject is not addressed definitively in this volume; many of these requirements will be addressed as the WP designs approach the final design stage. However, the possibility of liquid water coming in contact with the WP will depend on the local thermal conditions around each WP. Thermal evaluations which result in the possibility of water contacting the WP surface are discussed in Section 6. No thermo-hydrological calculations were performed for this volume. The assumption is also discussed in Section 4, Materials Selection, with regard to corrosion resistance requirements. Corrosion resistance is important in determining the lifetime of the WP and the fraction of WPs breached at 1000 years. Section 7 discusses the probabilistic degradation analysis methods which are being developed to evaluate WP performance in order to meet the stated assumption. Finally, barrier life is discussed in Section 8, including performance assessment issues.

3.2.3 Waste Package Design Concept Assumptions

The design concept assumptions include identification of design judgments and/or decisions made to move forward with the design. The design concept assumptions included in the CDA document typically are those that have insufficient technical data to support a final decision on the design. Only those assumptions which directly apply to the Engineered Barrier Segment/WP design are listed here, the reader is referred to the reference for the complete list (CRWMS M&O 1995n).

DCSS 006 Maximum Excavation Extraction Ratio

Maximum excavation extraction ratio for emplacement drifts: 30 percent.

Drift spacings assumed for the emplacement scale thermal evaluations of Section 6 are within the maximum excavation extraction ratio for the drift diameter used. (In a horizontal plane through the widest part of repository excavations, the area of excavation divided by the total area is the excavation extraction ratio.)

DCSS 023 Maximum Preclosure Rock- Surface Temperature

Maximum allowable preclosure rock surface temperature in:
Shafts: 35°C - unventilated
Ramps: 35°C - unventilated
Mains: 50°C
Emplacement Drifts: 200°C

Emplacement drift wall temperatures are calculated and compared to this thermal goal in Section 6. The emplacement drift thermal goal is equivalent to the one specified in CDA assumption EBDRD 3.7.G.2.

DCSS 025 Maximum Temperature in CHn

Maximum allowable temperature within CHn: 115°C.

The CHn rock unit is below the TSw3 rock unit which is limited to 115°C by CDA assumption EBDRD 3.7.G.3. CHn temperatures cannot exceed the thermal goal if TSw3 temperatures do not. TSw3 rock unit temperatures are calculated and compared to the thermal goal in Section 6.

DCSS 030 Limit Ground Surface Uplift

Limit surface uplift to less than 0.5 cm/yr and relative motion of the top of TSw1 to less than 1 m with no intact rock failure and no continuous joint slip.

Surface uplift of the top of the TSw1 rock unit were not calculated in the thermal evaluations, but are to be addressed as the design progresses.

DCSS 031 Maximum Temperature in PTn

Limit temperatures in PTn (Upper Paint Brush non-welded): 115°C.

PTn rock unit temperatures were not compared to this thermal goal in emplacement scale thermal evaluations conducted, however, in all cases evaluated, TSw3 rock unit temperatures were much higher than PTn rock unit temperatures. TSw3 rock unit temperatures are calculated and compared to the thermal goal in Section 6.

DCWP 001 Limit Fuel Cladding Temperature

Limit the fuel cladding temperature to less than 350°C.

SNF cladding temperatures are calculated and compared to this thermal goal in Section 6.

DCWP 002 Limit High-Level Waste Glass Temperature

Limit the temperature of the HLW glass to less than 500°C.

Peak HLW glass temperatures are calculated and compared to this thermal goal in Section 6.

DCWP 003 Alternate Waste Package Container Materials

Alternate waste package disposal container materials:

Corrosion allowance materials:
UNS G10200 (ASTM A 516 Grade 55)
UNS J02501 (ASTM A 27 Grade 70-40)
UNS K21590 (ASTM A 387 Grade 22)

Moderately corrosion resistant materials:
UNS C70600 (ASTM B 171)
UNS C71500 (ASTM B 171)
UNS N04400 (ASTM B 127) (Monel 400)

Corrosion resistant materials:
UNS N06022 (ASTM B 575, AWS ERNiCrMo-10)
UNS N06030 (ASTM B 582, AWS ERNiCrMo-11)
UNS N06455 (ASTM B 575)
UNS N06625 (AWS ERNiCrMo-3)
UNS N06985 (ASTM B 582, AWS ERNiCrMo-9)
UNS N08065 (AWS ERNiFeCr-1)
UNS N08221 (ASTM B 424) (not in commercial production)
UNS N08825 (ASTM B 424)
UNS R53400 (ASTM B 265 Grade 12, AWS ERTi-12)
Titanium Grade 16

This assumption is inherent in the WP designs presented in Appendix B. A discussion of these materials is presented in Sections 4 and 8. Material selection is discussed in Section 4 and the WP design analysis using the assumed materials is provided in Section 6. The Monel 400 was not analyzed in based upon the discussion concerning the material selection requirements, the engineering development/manufacturing requirements, and the cost estimate of using the material in the WP design.

TDSS 002 Ground Surface Temperature, Rock Thermal Gradient

Rock temperature at ground surface:	18.7°C.
Thermal Gradient in rock:	0.01968°C/m for depth 0 to 150 m
	0.018°C/m for depth 150 to 400 m
	0.030°C/m for depth 400 to 541 m

These thermal gradients are existing (pre-repository)gradients and are not repository limits. The ground surface temperature and thermal gradients were assumed in the emplacement scale thermal evaluations of Section 6.

TDSS 005 TSw2 Thermal Conductivity

Thermal conductivity of in situ rock mass - TSw2: 2.1 W/m·K.

This thermal conductivity was assumed in the emplacement scale thermal evaluations of Section 6.

TDSS 006 TSw2 Heat Capacitance

Heat capacitance of in situ rock - Tsw2:

2.0324×10^6 J/m^3·K @ 25°C	2.0065×10^6 J/m^3·K @ 115°C
2.1280×10^6 J/m^3·K @ 50°C	2.1114×10^6 J/m^3·K @ 155°C
2.2638×10^6 J/m^3·K @ 94°C	2.1912×10^6 J/m^3·K @ 195°C
10.7683×10^6 J/m^3·K @ 95°C	2.2692×10^6 J/m^3·K @ 235°C
10.4690×10^6 J/m^3·K @ 105°C	2.3410×10^6 J/m^3·K @ 275°C
10.1984×10^6 J/m^3·K @ 114°C	

These thermal capacitances were assumed in the emplacement scale thermal evaluations of Section 6.

3.3 FUNCTION ALLOCATION AND ARCHITECTURE

This section describes the allocation of functions first to the MGDS, then the MGDS functions assigned to the Engineered Barrier Segment, and finally, the Engineered Barrier Segment functions allocated to the Waste Package Subsystem and the Underground Facility Subsystem. The functions and functional allocations presented in this section are based on Chapter 4 of the CDA Document. The functional definitions, performance requirements, and Configuration Item allocations currently

being completed under the MGDS Functional Analysis project will be reviewed and incorporated into future designs.

This section further defines the Engineered Barrier Segment by providing information on the interrelationships of the functions, defining the top-level architecture of the Engineered Barrier Segment, and allocating the identified functions to subsystem elements. The repository architecture is described in Volume II, and repository design requirements are listed in Table 3-3, Volume II.

3.3.1 Mined Geologic Disposal System Element Functions

The top-level function of the MGDS, Dispose of Waste, is shown in context with the other CRWMS functions in the Manage Waste Disposal function flow diagram, Figure 3-2. The Dispose of Waste function includes, as subfunctions: Characterize Site, Construct MGDS, Operate MGDS (including retrieval, if necessary), Isolate Waste, Evaluate System Performance, and Close MGDS. These functions are depicted in the Dispose of Waste functional flow block diagram, Figure 3-3.

Table 3-1 provides a list of Dispose of Waste subfunctions and identifies references to functional flow blocks in the *MGDS Requirements Document* (DOE 1995a). The following sections define the major subfunctions of Dispose of Waste that pertain to the Engineered Barrier Segment.

3.3.1.1 Dispose of Waste - Function 1.4

Function 1.4, Dispose of Waste, is the final subfunction of 1.0, Manage Waste Disposal, and is the top-level function of the MGDS.

Dispose of Waste includes functions and activities required to emplace SNF and HLW in a geologic medium (while preserving an option for retrieval during the operational phase of the repository) and to isolate such wastes from the accessible environment. This includes the functions required to characterize the site to determine site suitability and support the process of designing and licensing the repository using the regulatory requirements specified in 10 CFR 60 and the guidelines of 10 CFR 960.

The term "disposal" is defined as the isolation of radioactive wastes from the accessible environment. Disposal means the emplacement in a repository of high-level radioactive waste, spent nuclear fuel, or other highly radioactive material with no foreseeable intent of recovery, whether or not such emplacement permits the recovery of such waste.

Dispose of Waste contains three subfunctions that are allocated to the Engineered Barrier Segment, Function 1.4.3, Operate MGDS; Function, 1.4.4 Evaluate System Performance and Function 1.4.6, Isolate Waste (Postclosure).

3.3.1.2 Operate Mined Geologic Disposal System - Function 1.4.3

This function includes MGDS activities conducted following Construct MGDS, but prior to close MGDS and Isolate Waste (Postclosure). It begins when the first waste form is received at the Repository and ends when an authorization of permanent closure is received. Operate MGDS

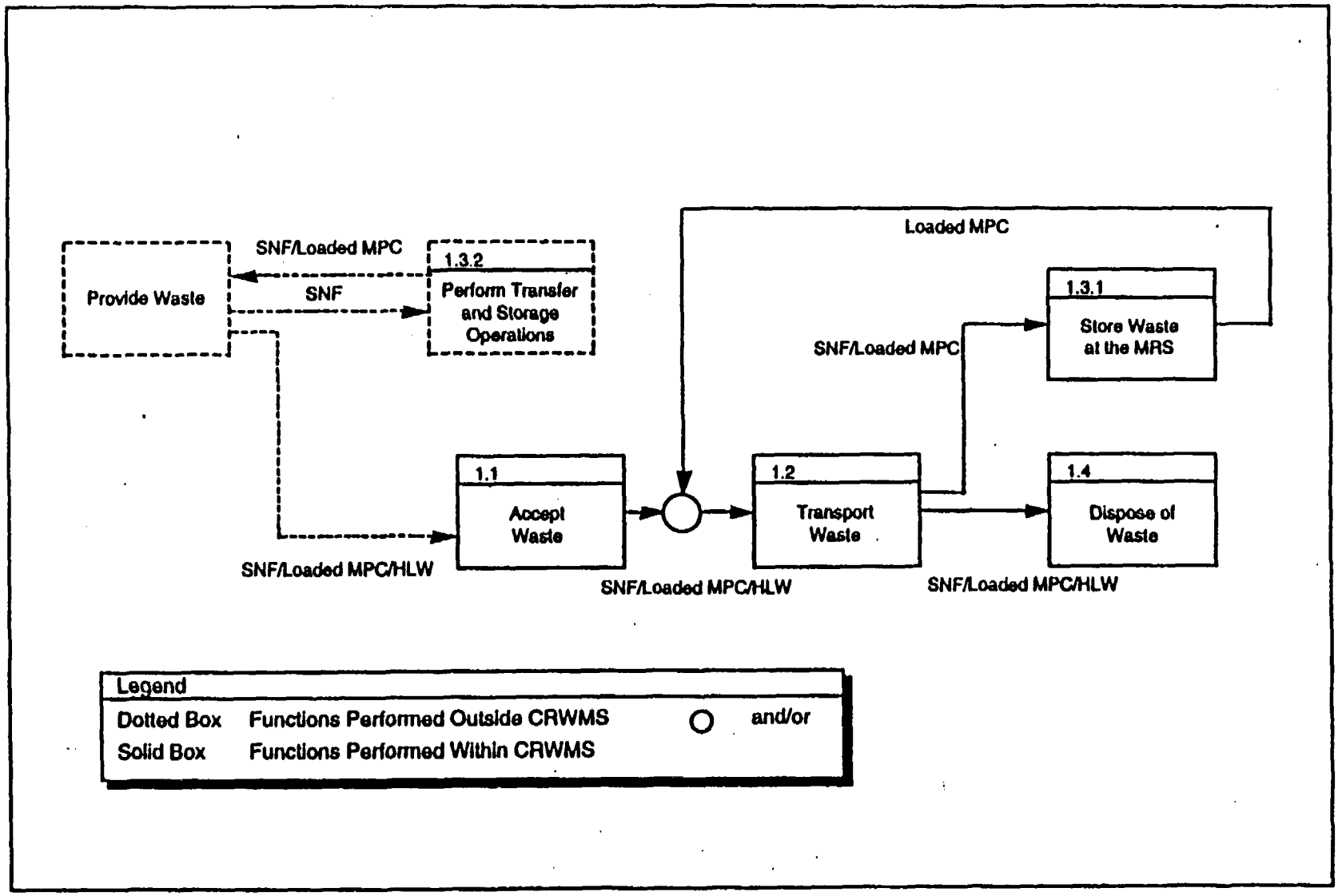
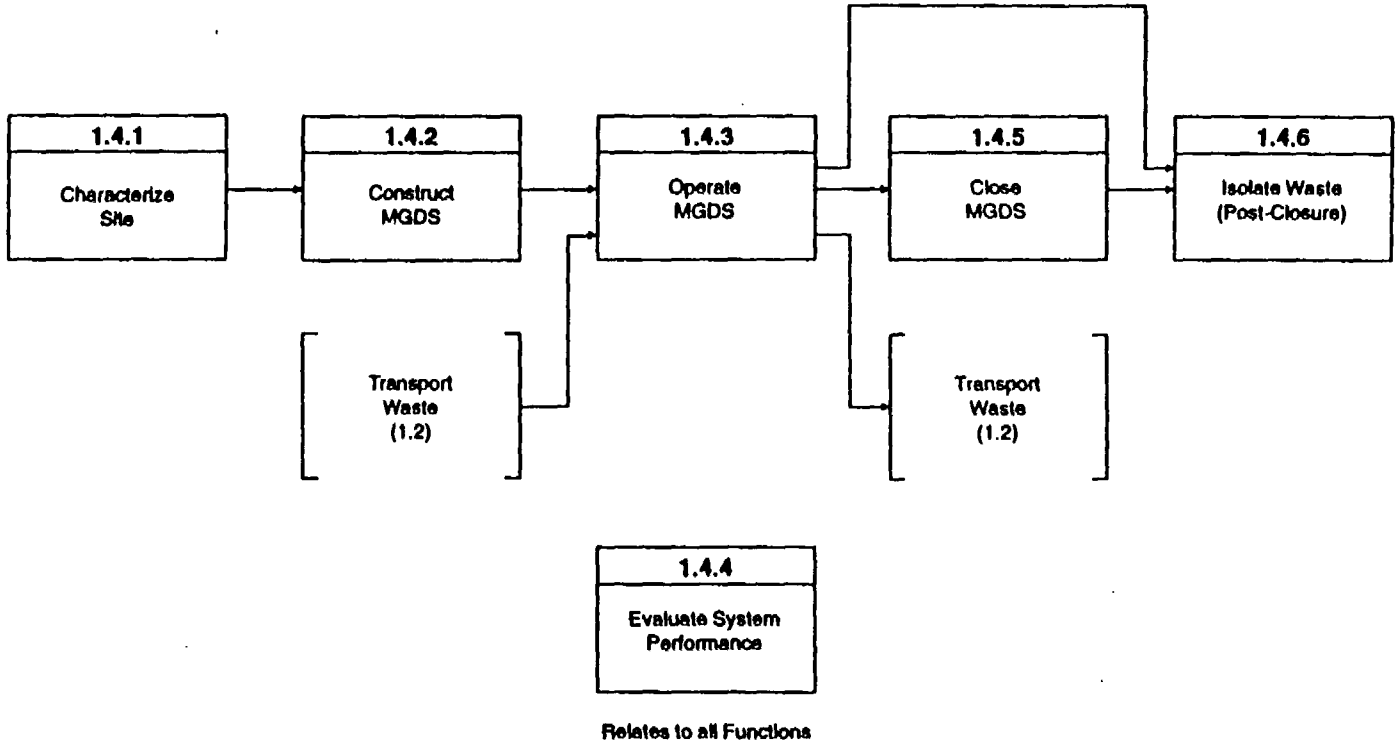


Figure 3-2. Manage Waste Disposal Functional Flow (1.0)



(1.4) Dispose of Waste

Figure 3-3. Dispose of Waste Functional Flow (1.4)

Table 3-1. Mined Geologic Disposal System Functions Allocated to Segments

MGDS Function Number	Function Title	Site Segment	Repository Segment	Engineered Barrier Segment
1.4	Dispose of Waste	X	X	X
1.4.1	Characterize Site	X		
1.4.1.1	Conduct Surface Investigations	X		
1.4.1.2	Conduct Subsurface Investigations	X		
1.4.1.3	Assess Data	X		
1.4.1.4	Close ESF	X		
1.4.2	Construct MGDS		X	
1.4.3	Operate MGDS		X	X
1.4.3.1	Support MGDS Operations		X	
1.4.3.2	Handle Waste		X	
1.4.3.3	Develop Underground (UG) Openings		X	
1.4.3.4	Contain Waste (Preclosure)			X
1.4.4	Evaluate System Performance		X	X
1.4.5	Close MGDS		X	
1.4.5.1	Close UG Openings		X	
1.4.5.2	Decommission Surface Facilities		X	
1.4.5.3	Establish Institutional Barriers		X	
1.4.5.4	Reclaim Site		X	
1.4.6	Isolate Waste (Postclosure)		X	X
1.4.6.1	Contain Waste (Postclosure)			X
1.4.6.2	Limit Egress of Radionuclides			X
1.4.6.3	Limit Release of Radionuclides to Accessible Environment		X	X
1.4.6.4	Monitor Postclosure Performance (if required)		X	X

includes the operation of surface and subsurface facilities and continued development of the underground drifts necessary to emplace radioactive wastes in the Repository. In addition to operational support and maintenance, principal activities include receiving, transferring, packaging, emplacing, and, if required, retrieving radioactive wastes. This function also included administrative subfunctions of receiving, interpreting, and distributing external permits, regulations, procedures, etc., to other Dispose of Waste functions.

The Operate MGDS function allocated to the Engineered Barrier Segment is 1.4.3.4, Contain Waste (Preclosure) and is discussed in Section 3.3.2.1.

3.3.1.3 Evaluate System Performance - Function 1.4.4

This function applies to all activities associated with Dispose of Waste (1.4). Evaluate System Performance includes all activities related to performance confirmation that are required to assess the ability of the repository system to comply with regulations governing its preclosure and postclosure performance objectives and its effects on the environment; using the estimates in updates to compliance documents and in support of the continuing development of the system; conducting the performance confirmation and environmental monitoring, in situ and laboratory testing, and performance modeling and analysis programs needed to supply data for the estimates; and assessment of the need for postclosure monitoring and any planning for postclosure monitoring of the system. The function starts during site characterization and continues until permanent closure.

3.3.1.4 Isolate Waste (Postclosure) - Function 1.4.6

This function follows Operate MGDS and Close MGDS and is an objective of Dispose of Waste. Isolate Waste includes inhibiting the transport of radionuclides in the MGDS so that amounts and concentrations entering the accessible environment will be kept within prescribed limits. Isolate Waste (Postclosure) begins when closure of the underground openings is complete and continues indefinitely. It includes initial containment of waste within a WP; then, together with other parts of the Engineered Barrier Segment, retardation of the movement of any uncontained radionuclides from the Engineered Barrier Segment to the geologic setting; and retardation and isolation of radionuclides through the geologic setting to the accessible environment.

The Contain Waste (Postclosure) and Limit Egress of Radionuclides functions are allocated to the Engineered Barrier Segment alone. The Limit Release of Radionuclides to Accessible Environment and Monitor Postclosure Performance functions are allocated to both the Repository Segment and the Engineered Barrier Segment. The Limit Release of Radionuclides to Accessible Environment function is also allocated to the natural barrier.

3.3.2 Engineered Barrier Segment Functions

This section defines the Engineered Barrier Segment by identifying and defining the functions the Engineered Barrier Segment will perform, providing information on the interrelationships of the functions, defining the top-level architecture of the Engineered Barrier Segment, and allocating the identified functions to the architecture. It also describes the major considerations and assumptions used in specifying the requirements contained in subsequent sections.

As indicated in Table 3-2, the Engineered Barrier Segment functions for Dispose of Waste are Isolate Waste, Operate MGDS and Evaluate System Performance and the subfunction, Contain Waste (Preclosure). The Isolate Waste (Postclosure) function includes, as subfunctions, Contain Waste (Postclosure), Limit Egress of Radionuclides and Limit Release of Radionuclides to Accessible Environment. Of these, the subfunction, Contain Waste (Postclosure), is a function of the Engineered Barrier Segment and the subfunction Limit Release of Radionuclides is a shared function of the Engineered Barrier Segment and the geologic setting. In addition, the Engineered Barrier Segment design will facilitate maintenance and retrieval functions, which will be performed by the Repository Segment. The functions, 1.4.4 Evaluate System Performance and 1.4.6.4 Monitor Post Closure Performance (if required) have been allocated to the Engineered Barrier Segments, however they are not included in Table 3-2 because there has been no allocation to Engineered Barrier Segments Subsystems.

3.3.2.1 Contain Waste (Preclosure) - Function 1.4.3.4

The Contain Waste (Preclosure) function is the containment of radionuclides prior to final closure of the MGDS. The function is allocated to the Waste Package Subsystem because a subsystem element within the Waste Package Subsystem will provide the waste form containment boundary. Function 1.4.3.4 results from function 1.4.3.2, Handle Waste Operations, beginning with the sealing of disposal containers, and continues through emplacement and terminates at MGDS closure; waste containment then continues within function 1.4.6, Isolate Waste (Postclosure).

3.3.2.2 Contain Waste (Postclosure) - Function 1.4.6.1

The Contain Waste (Postclosure) function confines the waste form and limits release of materials from an established boundary. The function is allocated to the Waste Package and Underground Facility Subsystems. A subsystem element within the Waste Package Subsystem will provide the waste form containment boundary. The Underground Facility Subsystem will assist in containment by limiting ingress of water. The function begins upon closure of the MGDS and continues until the disposal container degrades to a state that provides no degree of containment.

3.3.2.3 Limit Egress of Radionuclides - Function 1.4.6.2

The Limit Egress of Radionuclides function allocated to the Waste Package Subsystem because a subsystem element within the Waste Package Subsystem will provide the waste form containment boundary. The function is also allocated to the Underground Facility Subsystem because subsystem elements will retard the transport of radionuclides out of the Engineered Barrier Segment. The function begins upon closure of the MGDS and continues until the Engineered Barrier Segment degrades to a state that provides no control of egress.

3.3.2.4 Limit Release of Radionuclides to Accessible Environment - Function 1.4.6.3

The Limit Release of Radionuclides to Accessible Environment function is allocated to the Engineered Barrier and Repository Segments as well as the natural barrier. The function is allocated to the Waste Package Subsystem because a subsystem element within the Waste Package Subsystem will provide the waste form containment boundary. The function is also allocated to the

Table 3-2. Functions Allocated to Engineered Barrier Segment Subsystems (TBV)

Function Title	Engineered Barrier Segment Function Number	Waste Package Subsystem	Underground Facility Subsystem
Operate MGDS	1.4.3	X	X
Contain Waste (Preclosure)	1.4.3.4	X	
Isolate Waste (Postclosure)	1.4.6	X	X
Contain Waste (Postclosure)	1.4.6.1	X	X
Limit Egress of Radionuclides	1.4.6.2	X	X
Limit Release of Radionuclides to Accessible Environment	1.4.6.3	X	X

Underground Facility Subsystem because subsystem elements will retard the transport of radionuclides out of the Engineered Barrier Segment into the natural barrier. The natural barrier will function as the final barrier. The function begins upon closure of the MGDS and continues for the lifetime of the MGDS.

3.3.3 Mined Geologic Disposal System Architecture Description

The MGDS architecture, shown in Figure 3-4 as defined in Section 3, Requirements, of the MGDS Requirements Document, consists of the physical segments that will be developed to accomplish the top-level MGDS Dispose of Waste functions listed in Section 3.3.1. These functions will be implemented by the Site Segment, the Repository Segment, and the Engineered Barrier Segment. The allocation of functions to the MGDS segments is shown in Table 3-1.

The Site, Repository, and Engineered Barrier segments will provide the primary facilities, hardware, software, and processes that comprise the physical means to accomplish all MGDS functions.

3.3.4 Engineered Barrier Segment Architecture Description

The Engineered Barrier Segment consists of the Waste Package and Underground Facility subsystems. The allocation of functions to the Engineered Barrier Segment subsystems is shown in Table 3-2 and Figure 3-5 provides an illustration of the revised architecture. The architecture presently described in the EBDRD has been updated for this report to illustrate the progress which has been made in identifying subsystem element configuration items. This updated architecture will be documented in the next revision of the EBDRD.

3.4 OPERATIONAL PROCESSES

The CRWMS Manage Waste Disposal function will be performed by the four system elements: Waste Acceptance, Transportation, Storage, and MGDS. These system elements work together to fulfill a variety of functional and performance requirements intended to make the storage, transportation, and permanent disposal of waste in a geologic medium environmentally safe. CRWMS will provide appropriately documented tracking of the waste from initial acceptance to final MGDS closure. A program-level capability will be established to manage and control CRWMS

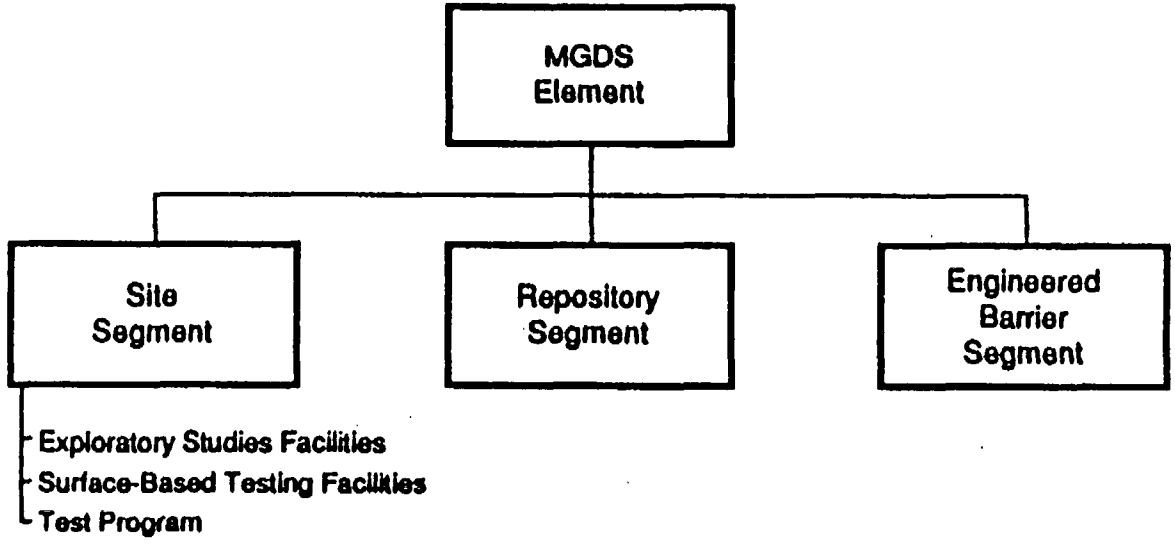


Figure 3-4. MGDS Architecture

ENGINEERED BARRIER SEGMENT CONFIGURATION ITEM ARCHITECTURE

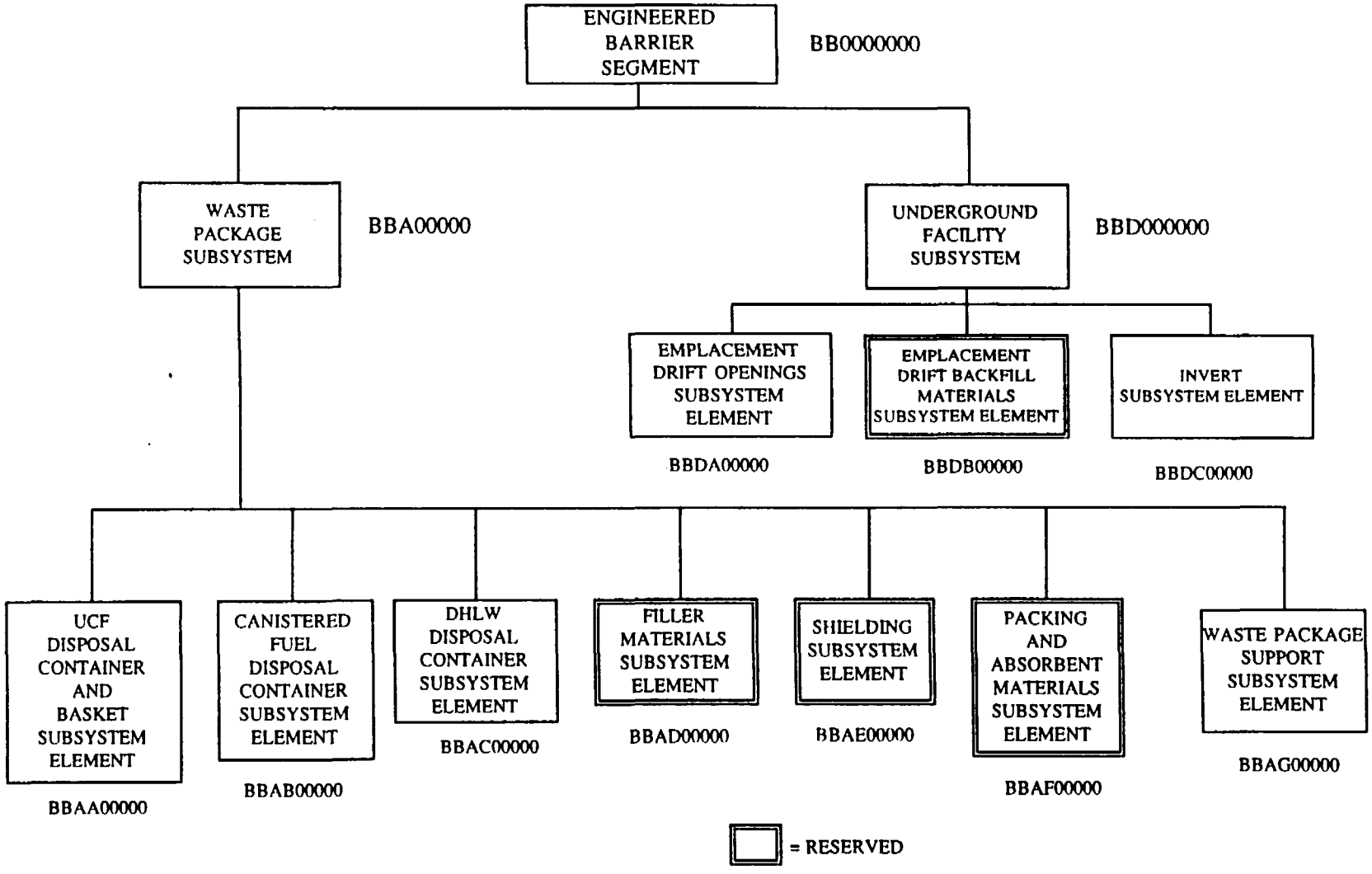


Figure 3-5. Engineered Barrier Segment Architecture

operation, such as communications management, scheduling, and other interactions and management functions among system elements. It will be functionally and operationally above the system elements as part of the overall CRWMS.

Detailed information regarding operational processes is provided in Section 5 of Volume 2 of this report.

3.5 DATABASE INFORMATION

The database information needed for design includes an understanding of waste form materials, container materials, and the WP environment. This type of information has been collected since the inception of the YMP and is not yet complete. The waste form data has been collected into the *Preliminary Waste Form Characteristics Report* (LLNL 1994c), while the container material data has been largely collected in the Survey of *Degradation Modes of Candidate Materials for High-Level Radioactive Waste Disposal Containers* (LLNL 1988) and *Engineered Materials Characteristics Report for the Yucca Mountain Site Characterization Project* (LLNL 1995h). The information on the WP environment has been collected into the *Preliminary Near-Field Environment Report* (LLNL 1993a). Design information is being collected into the Reference Information Base (YMP 1994a).

3.6 ENVIRONMENTAL DESIGN BASIS

No controlled design assumptions have been developed to specify the particular environments that are to be used to analyze waste package performance. Part of the difficulty in specifying an environment is the large number of characteristics that must be specified: water flux, water chemistry, pH, humidity, temperature, and radiation field form a minimal set. The environment will vary from the center to the edge of the repository because of heat transfer effects. It may vary from one waste package to another if localized fracture flow is significant. It will vary with time. If microbiological effects are significant, there may even be local variations depending on whether a microbial colony is present. Because of the complexity of the near-field environment, different investigators have made different assumptions, according to their needs. The conditions assumed have generally been deemed to be either realistic or conservative, according the investigator's aims.

Drift-emplaced WPs will be affected by the atmosphere surrounding them, the water that could potentially come in contact with them, and the movement of rock that could potentially impact them. During emplacement and early in the postclosure period, the atmosphere surrounding the WPs is expected to be hot air. Water vapor will be present, but its partial pressure will be well below that for saturation. The host rock surrounding the WPs will dry out and water vapor will be driven out spatially to locations where the temperature is low enough to permit condensation. As the near-field repository rock cools below the boiling point of water, moisture may be able to condense within the near field. The amount of water that could return to the emplacement openings will be a function of the thermal loading of the repository, the thermal profile around the WPs as a function of time, the imbibition into the rock matrix, and the active, available flow paths. Higher thermal loadings will increase the duration of dryout and increase the time required for water to return to the emplacement openings; thus the dryout time is a function of WP Subsystem/Engineered Barrier Segment design and thermal load. The water entering the emplacement openings could potentially

contact the WPs. This contact will be a function, once again, of the design of the WP Subsystem/Engineered Barrier Segment in that the emplacement openings may be backfilled to retard the contact of the water with the WPs or to drain the water away from them. If water does contact the WPs, it could occur by the wet-drip or moist continuous scenarios. The Engineered Barrier Segment design will be influenced by the scenarios' possibilities. Lastly, it has been suggested that the repository could be flooded by either the upwelling of water or a series of surface storms. The former event is deemed to be incredible; the latter event is considered to be possible, but not long lasting. Hence, the repository horizon may see water via fracture flow for brief episodic periods. However, the repository and Engineered Barrier System design could preclude such waters from entering emplacement locations.

Rock movement may impart thermal and mechanical loads to the emplaced WPs, particularly during the early postclosure period when the rock temperature is increasing. The rock load could be a result of rock expansion and rock fall or by rock instability caused by a phase transformation in the rock. For example, the transformation of the mineral cristobalite, from the alpha to beta structure at about 225°C, causes a volume expansion. Cristobalite (a form of silicon dioxide) is found in the TsW2 region of the potential repository horizon. Drift-emplaced WPs have a greater exposure to rock fall even though the rock is likely to be cooler and go through a slower thermal cycle. However, the wall thickness for drift-emplaced WPs is much thicker than the previous thin-walled borehole-emplaced reference design, so that the consequences of rock fall is reduced. In addition, drift-emplaced packages can be inspected during the preclosure period and steps can be taken to mitigate further rock fall. Table 3-3 contains environmental design bases affecting the design of the Engineered Barrier Segment.

3.7 OTHER INPUTS TO THE DESIGN DEVELOPMENT BASIS

Repository design is a vital input into WP design, and surface and subsurface facility design decisions must be interfaced closely with the WP design decisions. These include the repository surface facility limitations and the devices required for the emplacement operation. The decision on emplacement mode and the separation or commingling of spent fuel and HLW glass WPs will be made based upon the recommendation of the MGDS development team, consisting of the WP Subsystem/Engineered Barrier Segment and the repository subsurface design teams.

Table 3-3 Environmental Design Bases Affecting the Design of the Engineered Barrier Segment

1. Rock Induced Waste Package Loading	• Uniform external pressure of 0.50 MPa and a dynamic load of 50 kN	• Stress calculations based on the expected load from a rock falling onto the waste package
2. Limit of Fuel Cladding Temperature	• Less than 350°C	• SCP thermal goal, reconfirmed in 1993 analysis
3. Limit of HLW Glass Temperature	• Less than 500°C	• SCP thermal goal, reconfirmed in 1993 analysis
4. Repository Horizon	• For the reference thermal loading of 80-100 MTU per acre, the repository horizon will be located mainly in the TSw2 geologic unit within the primary area	• Provides a basis for thermal conductivity values and other physical properties of near-field materials
5. Backfill in Emplacement Drifts	• Current design assumes no backfill in emplacement drifts. Options for backfill will be considered based on ongoing and future backfill studies.	• Waste Package will be designed to withstand expected rockfall during substantially complete containment.
6. In-Drift Wall Temperatures Backfill in Emplacement Drifts	• Limit the predicted thermal and thermomechanical response of the host rock and surrounding strata and groundwater system, the Engineered Barrier Segment configuration and loading shall Keep emplacement drift wall temperatures less than 200°C.	• Based on recommendations from *Site Characterization Plan Thermal Goals Reevaluation* (CRWMS M&O 1993i)
7. TSw3 Temperature Limit	• Limit the TSw3 (basal vitrophyre) maximum temperature to less than 115°C.	• Based on recommendations from *Site Characterization Plan Thermal Goals Reevaluation* (CRWMS M&O 1993i)
8. Ground Surface Temperature Rise Limit	• Limit the maximum ground surface temperature change in the vicinity of the repository to 2°C.	• Based on recommendations from *Site Characterization Plan Thermal Goals Reevaluation* (CRWMS M&O 1993i)

4. MATERIALS SELECTION

The Waste Package (WP) design effort is focused on metallic, multibarrier disposal containers for large and small multipurpose-based canisters and other canistered fuel, uncanistered spent fuel, and high-level waste (HLW) glass canisters. This design incorporates an outer corrosion-allowance metal barrier over an inner container made of corrosion-resistant metal. The two distinctly different materials are selected to reduce the probability that a single environment will cause rapid failure of both. The WP is being designed for emplacement in a horizontal drift located in the unsaturated zone. The corrosion-allowance barrier, which will be thicker than the corrosion-resistant barrier, is being designed to corrode slowly, thus providing the inner container protection from the potential repository environment for a prolonged service life. Selection of suitable materials, therefore, exerts a significant influence on the resistance of these containment barriers to all pertinent forms of environmentally induced degradation.

The focus of this section is on WP materials. The Engineered Barrier Segment also includes the Emplacement Drift Backfill Material Subsystem Element and the Invert Subsystem Element. To date, much less emphasis has been placed on materials for these Subsystem Elements. A general discussion of these Subsystem Elements, their functions, and materials for constructing them is found in Section 8.1. In the balance of this section, materials selection for Subsystem Elements is discussed. For this purpose, both Subsystem Elements and portions thereof are referred to as components. The program for testing and modeling of these materials is described in Section 4.3.5.

4.1 MATERIALS SELECTION PROCESS

The WP materials selection process operates in a parallel, iterative manner with the design activities. The selection process draws heavily on previous work performed for the *Site Characterization Plan* waste package conceptual design (LLNL 1990b). The activities associated with the selection process, shown in Figure 4-1, are as follows:

- Definition of component functions, performance requirements, design requirements, and environments.
- Establishment of selection criteria and weighting factors.
- Identification of candidate materials.
- Collection of information/test data.
- Application of collected information/data to selection criteria and ranking.
- Selection and review.
- Confirmatory tests of selected materials.

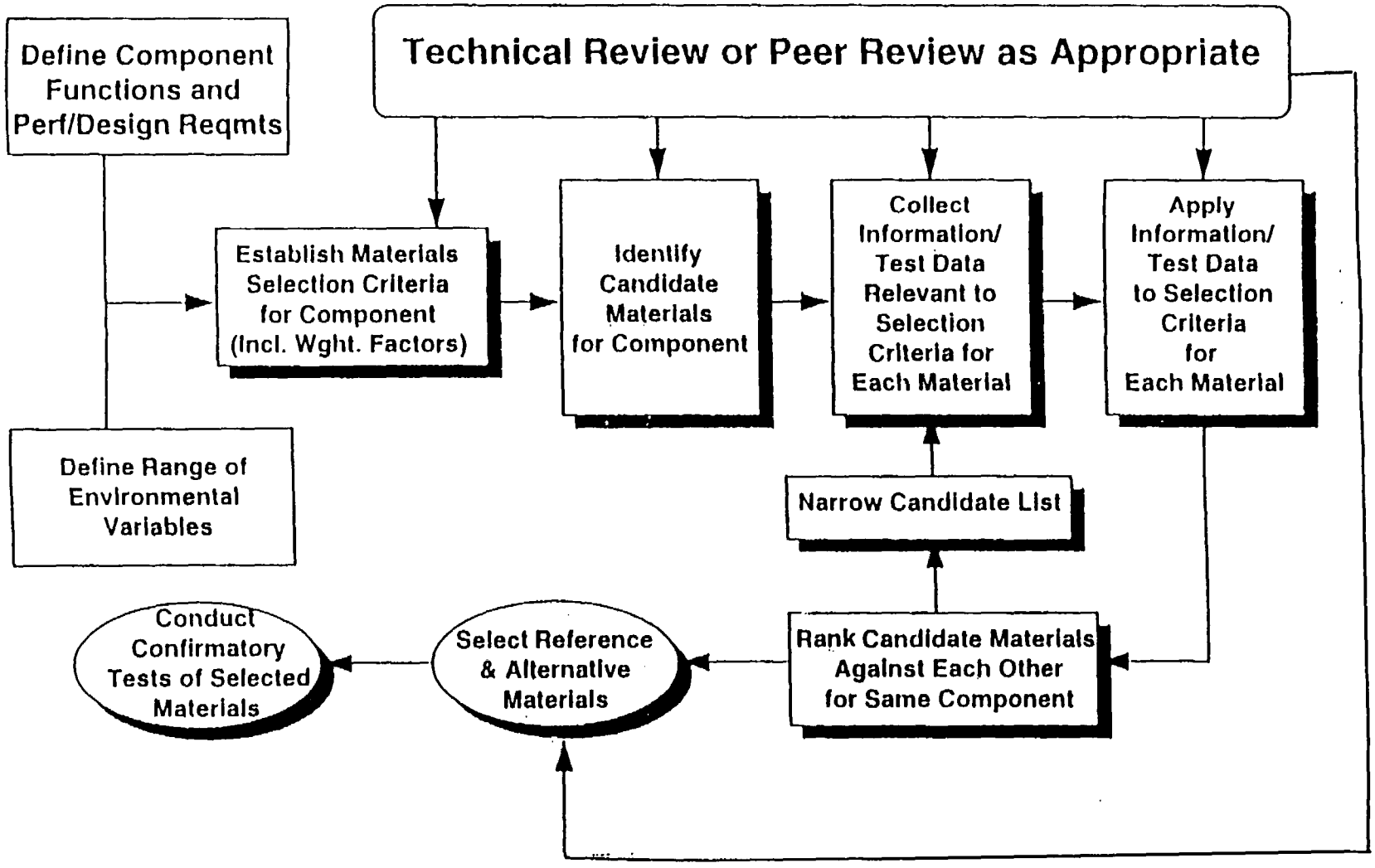


Figure 4-1. WP Material Selection Process

The criteria for materials selection should consider the WP design and service environment conditions. For metallic, multibarrier designs presented, the selection criteria are essentially a composite of how a material for a specific component performs within a WP system, and how well it meets the performance and design requirements. The anticipated functions and performance and design requirements of different components of a metallic multibarrier WP have been identified, as described in Tables 4-1 through 4-7.

Table 4-1. Postclosure Functions and Performance and Design Requirements of Metallic Outer Containment Barrier(s) in Air Environments*

Component	Function	Performance Requirements	Design Requirements
Outer Containment Barrier(s)	Contain Radionuclides	Cumulative fraction of WPs breached: 0 - 1,000 years $\leq 10^{-2}$ (to be verified [TBV]) (CRWMS M&O 1995n, Key 038)	Proven and reliable material, fabrication, closure and inspection technology (YMP 1994b, sec. 3.7.1.G) Reduce residual stress to a reasonably low level (YMP 1994b, sec. 3.7.1.B) Minimize strength reduction due to microstructural instability (YMP 1994b, sec. 3.7.1.2.G) Capability of withstanding seismic loading (YMP 1994b, sec. 3.7.1.1.G) Limit long-term penetration rate due to dry oxidation to 0.001 mm/year (TBV) (YMP 1994b, sec. 3.7.1.B)
	Limit radionuclide egress after container breach	Fractional release rate of radionuclides $< 10^{-5}$ per year of 1000 year inventory (YMP 1994b, sec. 3.7.E)	Degraded as well as intact containment barriers provide for retardation of radionuclide release (CRWMS M&O 1995n; YMP 1994b, sec. 3.7.1.2.C)

*Low to moderate relative humidity; other environmental variables as anticipated.

Table 4-2. Postclosure Functions and Performance and Design Requirements of Metallic Inner Containment Barrier in Air Environments*

Component	Function	Performance Requirements	Design Requirements
Inner Containment Barrier	Contain radionuclides	Cumulative fraction of WPs breached: 0 - 1,000 years ≤ 10^{-2} (TBV) (CRWMS M&O 1995n, Key 038)	Proven and reliable material, fabrication, closure and inspection technology (YMP 1994b, sec. 3.7.1.G) Reduce residual stress to a reasonably low level (YMP 1994b, sec. 3.7.1.B) Minimize strength reduction due to microstructural instability (YMP 1994b, sec. 3.7.1.2.G) Capability of withstanding seismic loading (YMP 1994b, sec. 3.7.1.1.G)
	Limit radionuclide egress after container breach	Fractional release rate of radionuclides < 10^{-5} per year of 1000 year inventory (YMP 1994b, sec. 3.7.E)	Degraded as well as intact containment barriers provide for retardation of radionuclide release (CRWMS M&O 1995n; YMP 1994b, sec. 3.7.1.2.C)

*Low to moderate relative humidity; other environmental variables as anticipated.

Table 4-3. Postclosure Functions and Performance and Design Requirements of Metallic Outer Containment Barrier(s) in Wet Environments*

Component	Function	Performance Requirements	Design Requirements
Outer Containment Barrier(s)	Contain radionuclides	Cumulative fraction of WPs breached: 0 - 1,000 years ≤ 10^{-2} (TBV) (CRWMS M&O 1995n, Key 038)	Proven and reliable material, fabrication, closure and inspection technology (YMP 1994b, sec. 3.7.1.G) Reduce residual stress to a reasonably low level (YMP 1994b, sec. 3.7.1.B) Minimize strength reduction due to microstructural instability (YMP 1994b, sec. 3.7.1.2.G) Capability of withstanding seismic loading (YMP 1994b, sec. 3.7.1.1.G) Compatibility of outer container material with inner container material so that inner container is galvanically protected (YMP 1994b, sec. 3.7.1.B) Limit long-term general (uniform) corrosion rate to 0.001 mm/year (TBV) (YMP 1994b, sec. 3.7.1.B) Limit long-term penetration rate due to localized (pitting and crevice) corrosion to 0.001 mm/year (TBV) (YMP 1994b, sec. 3.7.1.B) Limit long-term penetration rate due to microbiologically influenced corrosion to 0.01 mm/year (TBV) (YMP 1994b, sec. 3.7.1.B) Limit long-term penetration rate due to dry oxidation 0.001 mm/year (TBV) (YMP 1994b, sec. 3.7.1.B)
	Limit radiolytic effects		Provide sufficient wall thickness to reduce radiolytic effects to acceptable levels (CRWMS M&O 1995n, Key 031)
	Limit radionuclide egress after container breach	Fractional release rate of radionuclides < 10^{-5} per year of 1000 year inventory (YMP 1994b, sec. 3.7.E)	Degraded as well as intact containment barriers provide for retardation of radionuclide release (CRWMS M&O 1995n; YMP 1994b, sec. 3.7.1.2.C)

*High relative humidity or aqueous environments with variables as anticipated.

Table 4-4. Postclosure Functions and Performance and Design Requirements of Metallic Inner Containment Barrier in Wet Environments*

Component	Function	Performance Requirements	Design Requirements
Inner Containment Barrier	Contain radionuclides	Cumulative fraction of WPs breached: 0 - 1,000 years ≤ 10^{-2} (TBV) (CRWMS M&O 1995n, Key 038)	Proven and reliable material, fabrication, closure and inspection technology (YMP 1994b, sec. 3.7.1.G) Reduce residual stress to a reasonably low level (TBD) (YMP 1994b, sec. 3.7.1.B) Control strength reduction due to microstructural instability (TBD) (YMP 1994b, sec. 3.7.1.2.G) Capability of withstanding seismic loading (YMP 1994b, sec. 3.7.1.1.G) Compatibility of inner container material with basket material so that corrosion potentials are very close to each other (YMP 1994b, sec. 3.7.1.B) Limit long-term penetration rate due to localized (pitting and crevice) corrosion to 0.001 mm/year (TBV) (YMP 1994b, sec. 3.7.1.B) Limit long-term general (uniform) corrosion rate to 0.0001 mm/year (TBV) (YMP 1994b, sec. 3.7.1.B) Threshold stress intensity factor for stress corrosion cracking ≥ 50% (TBV) of critical plane strain stress intensity factor (YMP 1994b, sec. 3.7.1.B)
	Limit radionuclide Egress After Container Breach	Fractional release rate of radionuclides < 10^{-5} per year of 1000 year inventory (YMP 1994b, sec. 3.7.E)	Degraded as well as intact containment barriers provide for retardation of radionuclide release (CRWMS M&O 1995n; YMP 1994b, sec. 3.7.1.2.C)

*High relative humidity or aqueous environments with variables as anticipated

Table 4-5. Postclosure Functions and Performance and Design Requirements of Spent Nuclear Fuel Baskets in Air Environments*

Component	Function	Performance Requirements	Design Requirements
Basket (Criticality Control)	Control criticality by providing neutron absorption	Neutron multiplication factor (k_{eff}) ≤ 0.95 for period of waste isolation (YMP 1994b, sec. 3.7.1.3.A)	Proven and reliable material, fabrication, and inspection technology (YMP 1994b, sec. 3.7.1.G) Keep k_{eff} ≤ 0.95 for design basis fuel (YMP 1994b, sec. 3.7.1.3.A) Reduce residual stress to a reasonably low level (YMP, 1994b, sec. 3.7.1.B)
Basket (Structural)	Enhance heat transfer	Temperature of cladding ≤ 350°C (TBV) after closure (CRWMS M&O 1995n, DCWP 001)	

*Low to moderate relative humidity; other environmental variables as anticipated.

Table 4-6. Postclosure Functions and Performance and Design Requirements of Spent Nuclear Fuel Baskets in Wet Environments*

Component	Function	Performance Requirements	Design Requirements
Basket (Criticality Control)	Control criticality by providing neutron absorption	Neutron multiplication factor (k_{eff}) ≤ 0.95 for period of waste isolation (YMP 1994b, sec. 3.7.1.3.A)	Proven and reliable material, fabrication, and inspection technology (YMP 1994b, sec. 3.7.1.G) Keep k_{eff} ≤ 0.95 for design basis fuel (TBV) (YMP 1994b, sec. 3.7.1.3.A) Limit chemical degradation of criticality control material to 10% (TBV) loss of boron over a period of waste isolation (YMP 1994b, sec. 3.7.1.3.A) Limit neutronic degradation of criticality control material to 10% (TBV) loss of boron over a period of waste isolation (YMP 1994b, sec. 3.7.1.3.A) Reduce residual stress to a reasonably low level (YMP 1994b, sec. 3.7.1.B)
Basket (Structural)	Enhance heat transfer	Temperature of cladding ≤350°C after closure (CRWMS M&O 1995n, DCWP 001)	

*High relative humidity or aqueous environments with variables as anticipated.

Table 4-7. Preclosure Functions and Performance and Design Requirements of Metallic Containment Barrier(s)

Component	Function	Performance Requirements	Design Requirements
Containment Barrier(s)	Contain radionuclides	Containment of radionuclides under normal operating conditions (YMP 1994b, sec. 3.7.1.2.A)	Proven and reliable material, fabrication, closure and inspection technology (YMP 1994b, sec. 3.7.1.G)
	Maintain structural stability	Contain radionuclides under design-basis loads (YMP 1994b, sec. 3.7.1.1.F)	Capability of preventing plastic deformation due to design basis drop of 2 meters (CRWMS M&O 1995n; YMP, 1994b, sec. 3.7.1.1.F) Capability of sustaining 50 kN without breaching (TBV) (CRWMS M&O 1995n; YMP 1994b, sec. 3.7.1.1.F)

The selection criteria were classified into two major categories: (1) those related to the performance of candidate material in the anticipated repository environment, and (2) engineering-related aspects dealing with cost, engineering experience, and practical considerations of fabrication, closure, and material availability. The selection criteria under the first category may consist of several topical areas such as mechanical performance, chemical performance, predictability of performance, and compatibility with other materials. Use of the criteria in selecting materials was based on engineering judgment. The selection criteria are as follows:

Performance-Related Factors

- Chemical Performance
  - Resistance to general corrosion (dry oxidation and uniform aqueous corrosion)
  - Resistance to localized corrosion (pitting and crevice)
  - Resistance to environmentally assisted cracking (stress corrosion cracking and hydrogen embrittlement)
  - Resistance to microbiologically influenced corrosion
- Mechanical Performance
  - Strength
  - Fracture toughness (resistance to crack growth)
  - Phase stability

- Predictability of Performance

  - Existence of predictive methods to explain and predict degradation phenomena and to extrapolate existing performance data to repository time scales and conditions, or ability to develop such methods

  - Existence of long-term performance data

  - Ability to generate required data

  - Acceptance for use in other nuclear applications

- Compatibility with Other Materials

  - Interactions among materials of different components
  - Interactions with waste form

Engineering-Related Factors

- Fabricability

  - Fabricability of container body
  - Ability to close and seal the container
  - Inspectability of closure
  - Postclosure damage tolerance

- Cost

- Previous Experience with the Material

  - Previous engineering experience with the material
  - Previous engineering standards for the material
  - Strategic availability of material

In a previous analysis of the *Site Characterization Plan* design (LLNL 1993b), engineering judgment was used to identify materials that have the desired properties and generally favorable attributes relative to the selection criteria, and container materials were selected from the candidates. A similar process will be used for the current multibarrier design. A quantitative rating will be given to each candidate material for each selection criterion. This rating will be based on available test data, the degradation mode surveys, and other relevant data in both performance-related and engineering-related categories. The weighting factor for each criterion will be multiplied by these ratings, and the results summed for each material to establish an overall material rating. These ratings will then be used to rank the candidate materials for each component. Materials will be selected for each component, followed by confirmatory testing.

4.2 MATERIALS RECOMMENDED FOR DESIGN AND TESTING

4.2.1 Container Materials

A test plan encompassing a large number of candidate materials from families of corrosion allowance and corrosion-resistant metals would be both time-consuming and costly. In view of this, a few alloys have been chosen that are representative of major classes of engineering materials and that are expected to offer favorable combinations of price and performance. The *Report on Preliminary Selection of Waste Package Materials* describes which materials were tentatively selected and how they were selected (CRWMS M&O 1995p). A summary of the results in the report is given below. The precise environments of the WPs are not yet known, so it was decided that both dry and wet (aggressive) environments need to be considered in designing the test matrix.

Based on a degradation modes survey conducted during conceptual design (LLNL 1988), Alloy 825 was found to possess the best combination of metallurgical and corrosion properties. However, the conceptual design used for the survey had a single metal containment barrier, which is significantly different from the multibarrier packages currently being considered. It was nevertheless concluded that Alloy 825 should be the primary material for a corrosion-resistant inner barrier of a multibarrier WP emplaced in a horizontal drift, under both extended dry and wet environmental conditions. With respect to the outer containment barrier, American Society for Testing and Materials A 516 (ASTM 1990), a plain carbon steel, was selected as the corrosion-allowance material. There is some evidence that actively corroding ferrous metals will promote degradation of HLW glass (PNL 1984). As a result, the primary outer barrier material for HLW glass is currently specified as UNS C71500. The overall list of materials recommended for testing is presented in Table 4-8. Primary materials recommended for testing are summarized in Table 4-9; other materials are alternatives. Specifications for these materials are published by the American Society for Testing and Materials (ASTM 1990; ASTM 1992a; ASTM 1992b; ASTM 1993a; ASTM 1993b; ASTM 1993c; ASTM 1994; ASTM 1995a; ASTM 1995b) and the American Welding Society (AWS 1989; AWS 1990). UNS numbers are assigned by the Society of Automotive Engineers (SAE 1986).

4.2.2 Basket Materials

Austenitic stainless steel with boron is the material of choice for controlling criticality in disposal containers for uncanistered spent fuel. Should this prove to have insufficient durability for long-term criticality control, more costly and corrosion-resistant materials (such as zirconium-hafnium alloys) will be considered. Such materials might be more effective in disposable control rods than in a basket.

Additional materials that have been considered for use as criticality control materials include aluminum-boron alloys and Boral® (CRWMS M&O 1995a). Boral is produced by a powder metallurgical process in which boron carbide and aluminum powders are blended, sintered, and rolled. Boral has a sandwich structure, with a layer of boron-containing composite between layers of plain aluminum alloy. The inner layer is exposed at the cut edges of sheet or plate. Because of the very high boron carbide loading, the resulting composite has significant amounts of connected porosity, i.e., long, tubular pores that run from the surface deep into the material. The boron carbide particles are not well dispersed in that they are not separated by the aluminum matrix. In contrast,

aluminum-boron alloys are produced from a melt, have low porosity, and contain isolated aluminum borides (AlB_2) as the active neutron absorber.

The difference in microstructure between Boral and aluminum-boron alloys has important effects on corrosion behavior. In the aluminum-boron alloys, corrosion proceeds from the exposed surface through the material. As the aluminum boride particles are exposed, they may fall out, sift out of position, and became ineffective for criticality control. However, the particles that are still encapsulated in the aluminum matrix are still effective. In Boral, the connected porosity allows water to penetrate. Because the pores are connected, corrosion can occur not only on the apparent surface of the material, but also through the volume. As a result, much faster corrosion is expected for Boral than for aluminum-boron alloys.

Table 4-8. Materials Recommended for Testing

Component	Material
Outer Containment Barrier	UNS G10200 (ASTM A 516 Grade 55) UNS J02501 (ASTM A 27 Grade 70-40) UNS K21590 (ASTM A 387 Grade 22) UNS C70600 (ASTM B 171) UNS C71500 (ASTM B 171) UNS N04400 (ASTM B 127)
Inner Containment Barrier	UNS N06022 (ASTM B 575, AWS ERNiCrMo-10) UNS N06030 (ASTM B 582, AWS ERNiCrMo-11) UNS N06455 (ASTM B 575) UNS N06985 (ASTM B 582, AWS ERNiCrMo-9) UNS N08065 (AWS ERNiFeCr-1) UNS N08221 (ASTM B 424) (not in commercial production) UNS N08825 (ASTM B 424) UNS R53400 (ASTM B 265 Grade 12, AWS ERTi-12) Titanium Grade 16 (ASTM B 265 Grade 16)

Table 4-9. Primary Materials Recommended for Testing

Component	Material
Inner Containment Barrier	UNS N08825
Outer Containment Barrier (spent nuclear fuel)	UNS G10200
Outer Containment Barrier (high-level waste)	UNS C71500

4.3 MATERIALS TESTING AND MODELING

A testing and modeling program is required to support both the materials selection process associated with design (Section 4.1) as well as the performance analysis activities that are described in Section 8. Many of the design and performance requirements specified in the *Engineered Barrier Design Requirements Document* (YMP 1994b) and *Controlled Design Assumptions Document* (CRWMS M&O 1995n) will be verified through the results obtained under the testing and modeling program. This section provides an overview of the status and plans for a number of studies that are in progress to characterize the performance of candidate materials within a WP subsystem.

4.3.1 Container Material Degradation Studies

As discussed in Section 4.2, a number of materials have been recommended for testing as containment barriers. The alloys chosen are representative of major classes of engineering materials that are expected to meet both cost and performance considerations. However, because it is not possible to predict the precise environments in which the WPs must function, a series of studies have been initiated to bound the performance of the materials under a range of possible repository emplacement conditions. Material studies include literature surveys, laboratory studies, and model development activities.

4.3.1.1 Literature Surveys

The objectives of degradation mode surveys are to (a) compile relevant previously published information about a candidate material and its performance in a number of environments, (b) interpret this body of information in the context of a potential repository in Yucca Mountain, and (c) identify critical testing needs. In many instances, the degradation mode survey indicates the ways in which a material can degrade and serves to indicate the rate and kind of degradation in environments that have some similarity to what a metal barrier may experience in the Yucca Mountain setting. In other instances, the lack of information suggests what work will be required to determine the behavior of the candidate material in Yucca Mountain environmental conditions.

Degradation mode surveys have been completed for several candidate materials. The titanium survey (LLNL 1995a) was recently published and a draft degradation mode survey for nickel-copper alloys, particularly Alloy 400 (LLNL 1995g) is in review. The special role of welded microstructure degradation has recently been reported (LLNL 1995c), and additional degradation mode surveys are planned to study the galvanic effects between candidate materials for the WP container outer and inner barriers, and the galvanic effects between the container materials and other metallic components inside the package. Previous surveys, some dating back as far as 1988 (LLNL 1988), have dealt with copper-based alloys, austenitic stainless steels, high-nickel alloys (such as Alloy 825), high nickel-chromium-molybdenum alloys (including Alloy C-4 and C-22) (LLNL 1991b), and with carbon steels, alloy steels, and cast irons (LLNL 1995b) as corrosion allowance materials.

These surveys have ranked the performance of the evaluated materials under a range of environmental conditions. The early surveys examined materials for the thin-walled *Site Characterization Plan* waste package design utilizing a corrosion-resistant material. Of the materials evaluated for that design, Alloy 825 and CDA 715 were believed to have superior overall properties.

Titanium Grade 12 was added to this short-list of acceptable materials as a result of consideration of titanium-based alloys. These three alloys seem the last susceptible to localized corrosion mechanisms. The more recent surveys included nickel-chromium-molybdenum Alloys C-276, C-4, C-22 and Alloy 625. These alloys have excellent corrosion resistance in chloride environments such as seawater as well as more aggressive environments. With the design change to include an outer barrier of a thick-walled corrosion-allowance material, a survey was conducted on carbon and low alloy steels. It was found that plain carbon steels, low alloy steels, and cast irons and steels corrode at similar rates in an aqueous environment. Small additions of chromium and molybdenum increase aqueous corrosion resistance but, at higher levels, may increase susceptibility to localized corrosion.

On the basis of these surveys, as well as other input, the alloys identified in Table 4-8, plus some others, were included in the test matrix. The survey on welded microstructures provided insights into the impact of the welding process on performance and has provided feedback to the designers. Welded materials will be added to the text matrix. Testing will focus on particular areas relevant to expected conditions at Yucca Mountain, such as pitting of corrosion-resistance materials, where the available field and laboratory data are limited.

4.3.1.2 Laboratory Tests

A number of laboratory tests are planned or have been initiated to support the material degradation studies. These tests are designed to bound performance of candidate materials when subjected to degradation modes/environmental conditions associated with a potential Yucca Mountain repository. Specific test plan details are discussed in *Metal Barrier Scientific Investigation Plans* (LLNL 1995f). Additional discussion of candidate material properties and projected performance is presented in the *Engineered Materials Characterization Report* (LLNL 1995h) and the updated candidate materials report (LLNL 1995d).

Long-term corrosion studies will be initiated in early FY 1996 to determine comprehensive corrosion properties of metallic alloys being considered for constructing the multibarrier WP container. Three classes of materials are to be addressed: corrosion resistant, corrosion allowance, and intermediate. Corrosion properties to be assessed are general corrosion, pitting corrosion, crevice corrosion, intergranular corrosion, stress corrosion cracking, hydrogen embrittlement, and galvanic corrosion. This activity will provide kinetic and mechanistic information about the corrosion degradation of candidate materials. This information will support materials selection, performance analysis, and model development. Tests are conducted in "bounding environments" to capture the range of environmental conditions and water chemistries that are projected to evolve in the vicinity of the container surface over long periods of time. This comprehensive corrosion test is planned for a five-year period (or longer), with periodic removal and inspection of test specimens to measure corrosion degradation as a function of exposure time.

Thermogravimetric analysis studies are being used to determine the conditions under which atmospheric corrosion processes occur after emplacement of the WP. These conditions have special significance in an unsaturated zone repository, because the extent of degradation of the candidate materials becomes significantly greater when aqueous processes begin. The key parameters appear to be humidity, temperature, and surface conditions (roughness and composition) on the metal; the experimental work aims to determine the interrelationship among these parameters. Much of the

Yucca Mountain Site Characterization Project strategy for a robust WP, enduring for thousands of years, is predicated on maintaining low humidity levels around the WP. Recent thermogravimetric analysis studies (LLNL 1995e) indicate that carbon steel does not suffer from aqueous corrosion at relative humidities below 70 - 80 percent on clean surfaces. However, under repository conditions atmospheric corrosion may occur at somewhat lower humidities. These observations will be confirmed by long-term exposure in temperature and humidity-controlled chambers. The impact of the presence of contaminating surface films will also be evaluated.

Crack growth tests are being performed to determine the susceptibility of candidate WP materials to stress corrosion cracking under a variety of environmental, metallurgical, and mechanical stress conditions relevant to the repository, and to provide the experimental database for models capable of predicting the onset and growth rates for stress corrosion cracking in the WP environment. Metallurgical conditions include welded materials since stress corrosion cracks often emanate from welded areas. These tests are equally valid for determining the hydrogen embrittlement susceptibility of materials, if that degradation phenomenon is operating. Stress corrosion cracking and hydrogen embrittlement are important degradation modes that can affect corrosion-resistant and corrosion-allowance materials. Stress corrosion testing is focused on determining the critical stress intensity below which cracks will not propagate. This is important from a design point of view, to make sure that stresses on the container do not exceed this value and to place an upper value on defect sizes in the welded areas. Crack growth tests have been conducted on types 304L and 316L stainless steels and Alloy 825 in an earlier phase of the program. Current tests include titanium grade 12, Hastelloy C-14, Hastelloy C-22, and a new heat of Alloy 825.

A program of microbiologically influenced corrosion studies has been initiated to determine if corrosion is enhanced by the presence and propagation of microorganisms, particularly bacterial species. Metabolism products from these microorganisms can alter the chemical environment significantly, and this can occur on a localized level or over a wide area of the container surface. Different types of microorganisms attack different kinds of metals and alloys because of the chemical specificity of the corrosion process. Mutual interactions between electrochemical corrosion and bacterial cell growth are suspected but not well understood. For example, imposed galvanic protection potentials stimulate bacterial growth, and microbiologically influenced corrosion concentrates at welds where local galvanic currents are present. One aspect of the multibarrier container design is using "galvanic protection" from the sacrificial outer barrier to protect the inner barrier. The resistance of this inner barrier material to microbiologically influenced corrosion, and in particular in the welds of this material, takes on a great importance.

An experimental program has been initiated to gain a better fundamental understanding of microbiologically influenced corrosion in repository environments. Five factors have been identified as important to microbiologically influenced corrosion studies for the YMP:

- Sensitive electrochemical test procedures, including potentiostatic polarization
- Bacteria types and combinations that are most aggressive
- A range of pertinent alloys for study
- Weldments and galvanic effects
- Solute concentration and pH changes induced by man-made engineered barriers materials.

Plans are to use the well-known polarization resistance method to monitor corrosion rate changes during initiation and propagation of microbiologically influenced corrosion. Experimental cells have been assembled to culture various bacteria and measure effects on corrosion rates. Carbon steel, which is widely known to be susceptible to microbiologically influenced corrosion attack, was selected as the first test material. Corrosion rates in situ and in an external non-inoculated dummy cell are currently being measured.

4.3.1.3 Model Development

The objective of the model development activity is to derive predictive tools that will enable the use of experimental data and analyses to draw long-term assessments of the performance of candidate materials under Yucca Mountain conditions. This work will ultimately describe the performance of the multibarrier WP container, but as a first step in that direction the modeling work has focused on the pitting corrosion of corrosion-resistant material, such as one of the nickel-base or titanium-base alloys. While pitting is usually governed by electrochemical, chemical, and occasionally metallurgical parameters, an important aspect of pitting is "stochastic." Much of the present modeling work is aimed at developing the stochastic aspect of pitting within the electrochemical and chemical parameters. The pitting model accounts for three phenomena:

- The apparently random initiation of pits
- Growth of pits as a function of time
- The occasional permanent cessation of growth observed for "stable" macroscopic pits.

Experimental work was recently initiated in support of the pitting corrosion model. To date, this work has focused on electrochemical techniques to intentionally introduce pits by polarizing the test specimens to anodic values where breakdown of passive films readily occurs. This establishes a number of initiated and growing pits, and the depths (and other characteristics) of these pits are determined. These "polarization curves" also establish electrochemical potential ranges where pits initiate, propagate, or repassivate—and these will be used to establish test conditions for longer-term exposure tests to determine the fundamental terms in the pit growth model and to measure pit growth as a function of time. To date, only a limited number of electrochemical polarization tests have been conducted, mostly on austenitic stainless steels (where pit initiation is relatively easy) and on Alloy 825 in chloride solutions. Pits have not yet been generated on the most resistant materials, Alloy G-3, C-4, C-22, and titanium-Grade 12, because the environments so far tested have not been sufficiently aggressive.

It is expected that much of the same approach can be followed in modeling crevice corrosion, because both degradation modes show similar phenomenological dependencies for the corrosion-resistant materials. In many ways, an active crevice site is like the initiation stage of pitting.

4.3.2 Waste Package Basket Material Degradation Studies

Both the canistered fuel and the uncanistered fuel WP will incorporate internal "baskets" for fuel assemblies. These baskets are intended to provide structural support for the spent nuclear fuel (SNF) assemblies, to assist in heat transfer, and to assist in criticality control. Because of the emphasis on significantly larger, more robust waste packages, and on WP performance over a longer time period

than what was envisioned for the *Site Characterization Plan Conceptual Design Report* (SNL 1987b), evaluation of basket materials represents a relatively new activity in the materials testing effort.

Since the inception of the basket material degradation studies in FY 1995, the following have been accomplished:

- Bounding environments to be experienced by SNF baskets in the potential repository over 10,000 years were established.
- Selection criteria for basket materials, based upon their performance requirements and these bounding environments were developed.
- A preliminary list of candidate basket materials, making use of past practice in similar applications to the extent possible was generated (see below).
- Available data on the relevant properties of these candidates from literature sources and suppliers were compiled.
- Short-term scoping chemical environment, corrosion, and dissolution experiments to screen the candidates were performed (see below).

A preliminary list of candidate materials was developed. These materials were selected from the following product groups:

- Borated aluminum metal products
- Borated copper metal products
- Borated stainless steel products
- Zirconium metal products
- Ceramic and non-metal products.

These materials were subjected to a scoping corrosion test in a solution that, while aggressive in nature, is believed to be characteristic of the kind of environment that would develop inside the WP if the container is breached and radiolysis of incoming groundwater occurs. The aluminum- and copper-based materials were found to undergo considerable corrosion, as expected. The unwelded stainless steel based materials, as well as the zirconium-hafnium and the ceramic materials, performed well. The welded stainless steel suffered considerable corrosion on and adjacent to the weld.

Testing of promising candidate materials will continue, under a range of repository-relevant environmental conditions:

- To perform longer-term scoping chemical environment, corrosion, and dissolution experiments to screen the candidates.

- To measure the physical, thermal, and mechanical properties needed for design on the most promising candidates, if these data are not otherwise available.

- To model the corrosion and dissolution behavior of the most promising candidates over the long term.

- To recommend the most promising candidate basket materials to the designers of canistered fuel and uncanistered fuel waste packages.

4.3.3 Spent Fuel Oxidation Studies

The oxidation response of spent fuel is a degradation/alteration mode that can significantly increase the potential radionuclide release rate in the repository. This is because the UO_2 phase of spent fuel transforms to a U_4O_9 lattice (slight volume decrease) and then to U_3O_8 (approximately 30 percent volume increase), increasing the spent-fuel surface area exposed relative to the original pellet fragment/grain area. Also, the volume expansion accompanying the formation of U_3O_8 phase can split the zircaloy cladding open lengthwise. The oxidation response of spent fuel under repository conditions depends primarily on the temperature at the time when the spent fuel is exposed to atmospheric oxygen. Thus, this is an important input to design, particularly in keeping the temperature of the fuel below this oxidation threshold when the containment barriers are breached.

Dry-bath weight gain tests are in progress to determine spent fuel oxidation response. These are long-term weight gain tests conducted in a hot cell. These tests are primarily providing low-temperature (less than 200°C) oxidation response, but one dry bath recently was set to operate at 255°C to accelerate the oxidation rate. On the basis of information obtained from the dry-bath tests, thermogravimetric analysis tests were initiated at a higher range of temperature (250°C to 320°C). These two types of tests will provide temperature-time-phase response as UO_2 spent fuel oxidizes to U_4O_{9+x}, to U_3O_8, and finally to UO_3.

The dry-bath oven tests show an initial oxidation response of UO_2 by a U_4O_{9+x} phase front propagating into the UO_2 spent fuel grains. This U_4O_{9+x} phase has an oxygen-to-uranium metal ratio of approximately 2.4. The $UO_{2.4}$ is considered a metastable phase, and at low temperatures (less than 200°C), a plateau in the oxygen to uranium metal ratio vs. time curve is observed. To assess the stability time of the plateau domain, one dry bath is currently being operated at 255°C. The results from the dry bath at 255°C show no measurable plateau at the oxygen-to-uranium metal ratio of approximately 2.4. Detailed microscopic analyses of samples from the 255°C dry-bath tests have been difficult to perform because the samples are highly friable and tend to readily break into pieces. There are some preliminary indications of an amorphous structure, rather than an expected U_3O_8 lattice structure. These higher temperature dry-bath samples show that the U_4O_{9+x} plateau is unstable at the 255°C test temperature, and that the spent fuel degrades to a powdered state in relatively short times at this temperature.

The thermogravimetric analysis tests are complementary to the dry-bath tests and can provide higher temperature-time response data. Also, the thermogravimetric analysis tests provide a better weight gain versus time response data set, which, in conjunction with microscopy, can be used to gain a mechanistic understanding of the time-temperature kinetics and the transition from the $UO_{2.4}$ plateau phase to U_3O_8. The first test matrix in this activity addresses the data needs for the influences of temperature on the $UO_{2.4}$ plateau time interval before the initiation of the expected U_3O_8 phase

transition. Results of initial thermogravimetric analysis tests at 283°C are showing some sample-to-sample variability, where some samples do show a $UO_{2.4}$ plateau phase and some samples do not show this plateau. In studies of metastable phase transformations, some sample-to-sample variability would be expected. Thus, the task of determining time at temperature curves for spent fuel oxidation kinetics becomes more difficult.

4.3.4 Waste Form Dissolution Studies

The objective of these activities is to generate waste form (spent fuel and HLW glass) dissolution data for use in performance assessments and for direct use in licensing.

4.3.4.1 Spent Fuel Dissolution Testing and Modeling

As part of this task, saturated flow-through tests on spent fuel and uranium oxides are designed to measure their dissolution rates and dependence on several parameters, such as solution pH, temperature, oxygen fugacity, flow rate and solution anions, particularly carbonate species. The unirradiated uranium oxides represent new or zero burnup fuel. In addition, unsaturated drip tests are used to determine the rate of fuel alteration and release rates of different radionuclides under low-water-volume contact-rate conditions. Colloidal actinide species may form under these unsaturated conditions and be the major transport mode for radionuclides.

Several flow-through dissolution experiments on uranium oxides have been completed. Rates for spent fuel, UO_2, U_3O_8 and UO_3-H_2O have been measured now at several common conditions. As with all of the earlier experiments in these test series, these were conducted at 8 ppm dissolved oxygen (20 percent oxygen in the gas phase). The temperature (25 to 75°C), pH (8 to 10) and total carbonate concentration (2×10^{-4} to 2×10^{-2} molar) varied. The average dissolution rate of U_3O_8 (approximately 26 $mg{\cdot}m^{-2}{\cdot}d^{-1}$) was slightly more than UO_2 (approximately 18 $mg{\cdot}m^{-2}{\cdot}d^{-1}$) at the same experimental conditions. The dissolution rates from the same experiments with UO_3-H_2O were over three times higher at approximately 18 $mg{\cdot}m^{-2}{\cdot}d^{-1}$. Carbonate concentration has a particularly strong effect on the higher oxides.

Cesium-137 release studies of ATM-103 spent fuel indicate grain boundaries were exposed to water. This would indicate an increased total surface area exposed to water of up to 10 times the original area. However, the uranium dissolution rate did not confirm this interpretation. Flow-through dissolution tests were initiated with two oxidized (U_4O_{9+x}) and two unoxidized (UO_2) ATM-104 spent fuel specimens. One specimen of each fuel, oxidized and unoxidized, was in the form of multigrain particles approximately 1 mm in size. The other two specimens, one of each fuel, consisted of powders reduced to the size of individual grains and subgrains. The powders of both oxides appear to be dissolving at similar rates of 5 to 10 $mg{\cdot}m^{-2}{\cdot}d^{-1}$, similar to earlier results with ATM-105 fuel. However, the oxidized powder specimens are dissolving about ten times faster than the unoxidized particle samples.

The long-term unsaturated dissolution tests at 90°C continue with RJ-13 water and two types of spent fuel samples, ATM-103 and ATM-106. These tests have been in progress for over three years and sampled once during the past six months. The tests are conducted at two different water contact rates and with saturated water vapor. At the 750 day period, the ATM-103 fuel surface in the higher contact mode was dramatically altered. The surface was covered with a white yellow mat of material estimated to be about 0.5 mm thick. After the 920 day period, the ATM-106 fuel also had altered in a similar fashion. In the lower water contact mode tests, some surface alteration was observed,

but the extent was less than in the high contact mode tests. Some alteration was observed in the saturated vapor test. The experimental results indicate that colloids contain actinide phases and will be present in the leachates in significant quantities, an important outcome for performance assessment.

The development of a model for dissolution has focussed on the oxidative dissolution approach. Experiments have been conducted in a joint U.S.-Canadian program on powder dissolution to provide input parameters to the model. Experiments were conducted in sodium bicarbonate solution at a pH of 8 and a "standard" saline/bicarbonate solution, the latter without pH control. Samples were taken to examine and weigh the specimens. The dissolution rates were found to be consistent with those noted above, i.e., in the range of 4 to 6 $mg \cdot m^{-2} \cdot d^{-1}$.

4.3.4.2 High-Level Waste Glass Dissolution Testing and Modeling

These tests provide data on radionuclide release mechanisms and degradation/alteration rates of glass waste forms. These data are being used to constrain and guide on-going model development for glass corrosion. Additional glass waste form testing will be required to model glass degradation/alteration modes and the subsequent dissolution and release of radionuclides as soluble and colloidal species over the range of potential repository environments. This information is fed directly into repository performance assessment models, such as the Yucca Mountain Integrating Model to evaluate total release of radionuclides from the glass waste form.

Long-term unsaturated tests (drip tests) of two glass compositions (Defense Waste Processing Facility and West Valley ATM-10) continue in two test series labeled the N2 and N3 series. These tests are being used to determine the types and quantities of radionuclide elements released from waste glasses when subjected to an intermittent dripping water contact scenario. Both soluble and colloidal radionuclide releases of actinides and technetium are being measured. A 304L stainless steel sample holder is also present in these tests to simulate the presence of the pour canister material on glass waste form behavior.

The Defense Waste Processing Facility glass is being tested in the N2 test series which has been in progress for over 500 weeks (9.6 years). The tests are sampled roughly every six months. In addition to the analyses required by the test procedure, sequential filtering (removing successively finer suspended solids with a series of filters) of solution extracts were performed. Preliminary analytical electron microscopy of these solids has tentatively identified them as clays and iron silicates, some of which contain small amounts (0.1 to 1 weight percent) of uranium. These particles have been preserved for later analyses with alpha spectroscopy and other analytical techniques. Uranium in these tests solution was measured for the first time using inductively-coupled plasma-mass spectroscopy. These solutions were not amenable to analysis using alpha spectroscopy.

The West Valley glass (ATM-10) being tested in the N3 test series has been in progress for over 435 weeks (8.4 years). The tests were sampled on schedule. As with the N2 series, solution samples were subjected to sequential filtering and the particles are to be analyzed using analytical electron microscopy. For the first time, the visual examination of the reacted test components was documented using a video-microscope. The visual observations and verbal comments are thereby captured on videotape. Still photos from the videotape will be prepared for the records file and annual report.

Measurements made previously indicate that the release of plutonium and americium is controlled by the rate which the clay layers and corroded metal are transferred from the glass-metal assembly by the slowly dripping water. This rate has been substantially less than observed for lithium and boron, but the releases have been dominated by particulate material (colloids) that remain suspended in solution. During the last two sampling periods, a change in appearance of the reacting glasses was observed in some of the tests. The change was the appearance of apparently non-reacted glass which covered up to 40 percent of the glass surface, and an increase in the corrosion rate of the stainless steel components in the tests. A concomitant increase in the amount of particulate material was noted in the bottom of the test vessels.

Analyses of the test solution, including the solids, resulted in a sharp increase in the release of americium and plutonium. The releases of boron and lithium in these tests remained at nearly constant values, reinforcing the understanding of the manner in which the glass reacts. That is, a nearly congruent dissolution of the glass occurs at the solution/glass interface. Sparingly soluble elements are retained in a clay layer that is only loosely attached to the glass, while soluble elements are released to solution and rapidly rinsed from the glass-metal retainer. Pieces of clay spall from the retainer due to the flow of water over the glass surface. With time, much of the layer becomes detached from the glass and is released to solution. As these tests continue, it is anticipated that similar behavior will be observed in all the tests.

These results are being fed back into the modeling effort on glass corrosion and release of radionuclides. In this area, work continued on simplifying the current glass model to express the relationship between radionuclide release rate and environmental variables in terms of a minimum amount of parameters. This will aide the abstraction of the mechanistic model into the system model. The primary relationship involves the rate of dissolution of the glass as a function of temperature, pH, and dissolved silica concentration. Two additional empirical relationships were added to the current version of Yucca Mountain Integrating Model: (1) pH versus the amount of dissolved glass, and (2) the ratio of silica released to solution versus silica contained in alteration phases as a function of surface area-to-volume ratio in the test.

4.3.5 Engineered Barrier Materials Testing and Modeling

The program for the evaluation of engineered materials has just been initiated. These include potential backfill, packing, and invert materials, as well as materials for drip shields. The near-term objective of the effort is to characterize the potential engineered materials versus their function in the system. The evaluation will determine the benefits and detriments of these materials to system performance and develop a short list of materials to further characterize and recommend as a design element. Effort within the WP program and the system study is being integrated to assure efficient use of resources. The WP effort has focused on the chemical interactions of engineered barrier materials with the WP components, while the system study portion has examined the retardation and release of radionuclides from the Engineered Barrier Segment using the Yucca Mountain Integrating Model code. Longer-term WP work will include appropriate property determinations of the potential materials, engineered barrier/waste package material interactions, including corrosion products and colloids, utilizing the EQ3/6 code, and the development of predictive models.

handled in four or more ways: (1) the fuel may be blended so that the DBF is exceeded by only a small fraction of the assemblies in any single WP, (2) the package may not be fully loaded (called derating), (3) supplemental criticality control features may be added, or (4) another, more robust, waste package may be designed, to accommodate SNF that does not conform to DBF characteristics. From the definition of DBF, the amount of SNF requiring blending, derating, etc. will be less than the complement of the DBF percentile. Since this analysis will specify two DBFs, the percentage of fuel requiring such special consideration will generally be greater than the complement of either percentile. The actual percent will be a function of the dependence between the two DBF parameters, and will be indicated in the results section.

The DBFs are intended to satisfy requirements for all times after emplacement, but they are defined with respect to performance parameters at time of emplacement as a representative worst case. This simplification is based on the assumption that *the percentile of fuel covered by a DBF at time of emplacement will remain the same for all times at which design critical requirements are imposed.* This assumption is only an approximation for thermal performance beyond the first 100 years, since older fuel at time of emplacement will have a slower thermal decay than younger fuel. The inaccuracy of this approximation for the thermal DBF can be accepted because the significant consequences of high thermal output are limited to the first 100 years following emplacement. For criticality, this assumption will be verified by a family of deterministic criticality calculations at 10,000 to 20,000 years following emplacement because of the increase in reactivity during part of this time period. If the assumption is proven to be incorrect, an additional DBF (for long-term criticality) will have to be defined.

The following sections summarize the method and results; further details are found in the *Waste Package Design Basis Fuel Analysis* (CRWMS M&O, 1995c).

5.1 ASSUMPTIONS AND METHODOLOGY

Although the Characteristics Database contains information on over 85,000 metric tons of uranium (MTU), it is assumed that the repository will receive only the current statutory limit of 63,000 MTU and that the waste stream at the repository will be oldest-fuel- first. The oldest-fuel-first assumption was discussed in the introduction to this section.

The DBFs have been selected to represent a stated percent of SNF performance-based parameters, with statistics tabulated on a per-assembly basis. This captures the effects of variations in MTU per assembly, which is important since the WPs are designed to hold a specific number of assemblies, not a specific amount of MTU.

The DBF is specified in terms of three parameters: age, burnup, and initial enrichment. These are called the specification parameters. The effects of variations in other parameters, such as assembly geometry and cladding material, will have only a small effect on the performance parameters of present interest, so their specific value will be taken from standard engineering practice without the statistical analysis used here.

The performance parameters for which the statistics will be developed to select the DBF fall into two groups according to the type of stress they place on a WP design. The first group involves the

thermal and radiation (shielding) performance and is primarily determined by age and burnup. The second group involves the criticality performance and is primarily determined by initial enrichment and burnup (with a weak dependence on age). To represent both of these groups (thermal/shielding and criticality), we specify the DBF in terms of two sets of the specification parameter triplet (age, burnup, and initial enrichment).

The first step in compiling the statistics for the parameters of interest is to divide the range of each parameter into a number of intervals. These intervals constitute a set of bins for counting the number of assemblies having values of the specific parameter that falls within the limits of the interval, or bin. The next step is to read each batch record in the database and tabulate the numbers of assemblies into the appropriate bins. For tabulation into joint distributions, the bins become a two-dimensional array of intervals. Those parameters not read directly from the database records (primarily heat, k_∞) are calculated from the characteristics in the record (age, burnup, and initial enrichment). The methodologies for the tabulations for the individual parameters are given below.

5.1.1 Burnup

Since burnup is found directly in the Environmental Information Agency database records, the tabulation into percentile bins is straightforward. Assembly weights are accumulated in 50 bins, 1,000 megawatts days (MWd)/MTU wide, according to the value of burnup, ranging from 0 to 49,000 MWd/MTU for boiling water reactor (BWR) SNF. The percentages from bin to bin were then accumulated going from low to high, so that each percentile represented the percent of BWR SNF having burnup lower than the upper limit of the bin. For pressurized water reactor (PWR) fuel 75 bins were used, ranging from 0 to 74,000 MWd/MTU.

5.1.2 Heat at Emplacement

The thermal load on the WP (and consequently its temperature) is most directly determined by the rate of heat generation. Both heat generation rate and WP temperature change with time and with the total repository thermal loading. However, heat at time of emplacement is the single strongest determining parameter for peak temperature inside the WP, which is constrained by the need to avoid cladding failure (as a consequence of temperature-induced creep). The performance parameter, heat at emplacement, is primarily a function of age at emplacement and burnup.

The oldest-fuel-first scenario assumption determines which assemblies are included in the 63,000 MTU that go in the first repository. With this identification, the age at emplacement, burnup, and initial enrichment for each assembly can be read from the Environmental Information Agency database record. Heat at emplacement is computed by interpolation in the Characteristics Database table as a function of these three parameters.

Percentiles were developed by counting the number of assemblies in 70 bins with each bin having a width of 25 watts/assembly (so that the heat rates range from 0 through 1,725 watts per assembly).

5.1.3 Criticality

The effective neutron multiplication factor (k_{eff}) for an assembly in a WP is a function of age, burnup, initial enrichment, and the package geometry and contents. The situation is further complicated because burnup and initial enrichment are strongly correlated for most of the SNF. One general way to parameterize criticality as a function of burnup and enrichment is to compute k_∞ (the neutron multiplication factor for an infinite medium) for various values of these parameters using a suitable radioisotope inventory and neutron transport code. This has been done in a study by Cerne, Hermann, and Westfall, using the results of 210 computer runs (covering a typical range of values for age, burnup, and initial enrichment) using a set of programs and isotope tables known as the Standardized Computer Analysis for Licensing Evaluations (SCALE) (ORNL, 1987). These runs covered a range of five representative values for age, seven for burnup, and six for initial enrichment. The following formula was developed:

$$k_\infty = 1.06-(0.01\cdot b)-(0.002\cdot c)+(0.114\cdot a)+(0.00007081\cdot b^2)+(0.00007565\cdot c^2)-(0.007\cdot a^2)-(0.0002671\cdot b\cdot a)-(0.0001145\cdot b\cdot c)+(0.0002318\cdot c\cdot a)+(0.000009366\cdot b\cdot c\cdot a)$$

where a (assay) is initial enrichment in percent, b is burnup in GWd/MTU (gigawatt-days/MTU), and c (cooling time) is age in years.

Since k_∞ represents only the properties of the fuel as an infinite medium, any criticality analysis for a specific WP design must use k_{eff} to reflect the specific geometry of the fuel assembly and its environment. However, it is impractical to calculate k_{eff} for each possible SNF loading of a standard WP. Therefore, parameterization of k_∞ is used to represent k_{eff} by assuming that the burnup-enrichment pairs that make up the constant k_∞ isopleths corresponding to a specified percentile of all k_∞ values will be the same burnup-enrichment pairs that would make up a constant k_{eff} isopleth if the detailed, geometry-dependent values were calculated for all the assemblies in the database. In other words, all burnup-enrichment pairs having a single value for k_∞ will map into a single value for k_{eff}.

This formula for k_∞ was then applied to the batches in the database, and the resulting values of k_∞ were statistically tabulated in 70 bins of width 0.01, corresponding to values of k_∞ ranging from 0.75 through 1.44. The actual emplacement ages in the oldest-fuel-first scenario were used up to 20 years, but all assemblies older than 20 years at time of emplacement were assigned the age of 20, since the fit to the 210 SCALE runs of Cerne, et al. (ORNL, 1987) covered only age up to 20 years. The consequences of this assumption are expected to be minimal; the plan for checking this and making any necessary corrections is given in section 5.4 below.

Since the required series of SCALE runs are not yet available for BWR fuel, the criticality DBF for BWR fuel must be determined by using percentiles of initial enrichment. For this reason, the BWR criticality DBF will be provisional until the necessary parameterizing runs have been completed. The BWR DBF will not be changed when such a parameterization becomes available; instead, the percentage of fuel covered by the design basis will be changed.

5.1.4 Percentile Criteria for Specification of Design Basis Fuel

The container is required to satisfy regulatory requirements when loaded with fuel giving some specified heat output and having some specified k_∞. The tabulated statistics can then be used to translate these performance levels into coverage percentiles. This is the process of determining the coverage of expected fuel for a particular design.

Since the WP is designed on the basis of the DBFs (thermal/shielding and criticality) and will handle only the designated percentage of the fuel, packages handling the more stressing fuel will have to be derated to a smaller capacity or accommodated in another approved manner. With this derating, the design will meet regulatory requirements even for the more stressing fuel.

5.2 RESULTS

The results are presented separately for BWR and PWR SNF.

5.2.1 Pressurized Water Reactor Design Basis Fuel

The combined heat and criticality (k_∞) tabulation for PWR SNF is given in Table 5-1. Note that the combined percentages are tabulated from low parameter values to high so that the percentiles give the percentage of the fuel with less stressing values of the two parameters: heat generation rate and criticality potential.

In Table 5-1, the marginal percentages (single parameter) are given in the second row and the next to last column, for k_∞. The first row is heat output (watts per assembly), the second row is the percent of fuel having less than or equal values of watts per assembly, the last column is k_∞, the next to last column is percent of fuel having less than or equal value of k_∞, and the values in the rest of the table are the percentages of fuel having lower values of both heat and k_∞.

Preliminary design analysis has indicated the capability to design a 21 PWR package with sufficient internal thermal conductivity to handle a thermal load of 850 watts per assembly without exceeding the 350°C temperature limit on any of the fuel cladding. Preliminary analysis has also indicated the capability to apply sufficient internal neutron absorber material to reduce/translate a k_∞ of 1.13 to the required k_{eff} of 0.95. These values are, therefore, taken to be the design basis targets. They must then be mapped into the fuel characteristics—age, burnup, and enrichment—but first the percentage of coverage provided by these design basis targets should be noted.

Table 5-1 gives the percentage of fuel having a heat generation rate less than or equal to the design basis 850 watts per assembly (97.85 percent, second row, third column), and the percentage having a criticality potential less than or equal to the design basis k_∞ = 1.13 (98.31 percent, 13th row, 8th column). It also gives the coverage of the design basis with respect to both thermal and criticality together: 96.16 percent, 13th row, 3rd column.

Table 5-1. Percentiles for Combined Heat and Criticality for Pressurized Water Reactor Fuel

Heat Output (watts per assembly)								
900	875	850	825	800	775	750		
Percent of Fuel with ≤watts per assembly								
98.77	98.30	97.85	97.15	96.29	95.08	93.81		
Percentages of Fuel Having Lower Values of Both Heat and k_∞.							% with ≤ k_∞	k_∞
98.31	97.85	97.40	96.70	95.84	94.63	93.36	99.55	1.23
98.19	97.73	97.28	96.58	95.71	94.51	93.24	99.43	1.22
98.12	97.65	97.20	96.51	95.64	94.43	93.16	99.35	1.21
98.05	97.59	97.14	96.44	95.58	94.37	93.10	99.29	1.20
97.93	97.46	97.02	96.32	95.45	94.25	92.98	99.17	1.19
97.78	97.31	96.86	96.17	95.30	94.10	92.83	99.02	1.18
97.69	97.23	96.78	96.08	95.21	94.01	92.74	98.93	1.17
97.61	97.15	96.70	96.00	95.14	93.93	92.66	98.85	1.16
97.48	97.01	96.56	95.87	95.00	93.80	92.53	98.72	1.15
97.28	96.81	96.36	95.67	94.80	93.59	92.32	98.51	1.14
97.07	96.61	96.16	95.46	94.60	93.39	92.12	98.31	1.13
96.95	96.48	96.03	95.33	94.47	93.26	91.99	98.18	1.12
96.58	96.11	95.67	94.97	94.10	92.90	91.63	97.82	1.11
96.22	95.75	95.30	94.61	93.74	92.53	91.26	97.45	1.10
95.67	95.20	94.75	94.06	93.19	91.99	90.72	96.90	1.09

Before mapping the PWR heat per assembly into fuel characteristics, a mass per assembly must be selected. The value 0.464 MTU per assembly has been used for PWR waste package conceptual design, so that is the most appropriate for this purpose. To proceed with the mapping of heat per assembly into fuel characteristics, it should be noted that 850 watts per assembly can be produced by a range of values of age, burnup, and initial enrichment. For a given heat, the older fuel will have higher burnup and will decay more slowly (giving a higher integrated heat over the repository life); therefore, the more conservative design basis for the entire repository would have the older fuel at the higher burnup. However, most of the hot fuel will be close to 10 years in age, so the older fuel is not representative of the most stressing condition on an individual package basis. Therefore, it is conservative with respect to an individual WP to use 10 years age when selecting the burnup that will represent 850 watts per assembly. Since the thermal requirement is with respect to an individual package (peak internal and surface temperatures), the age (emplacement year minus discharge year) of 10 years is used. It should be noted that whatever age is used to specify DBF, the analysis used the actual ages of oldest-fuel-first to associate a percentile with 850 watts per assembly in the first place, and this is the essence of the DBF selection process.

Once age is selected, the process of burnup DBF specification is completed as follows. Take an initial estimate that the burnup giving 850 watts per assembly at 10 years age will be somewhat less than 50,000 MWd/MTU; then take an initial enrichment of 4.2 percent, which is slightly less than the average initial enrichment in the Characteristics Database table for 50,000 MWd/MTU (ORNL, 1993b). With age 10 years and initial enrichment 4.2 percent, interpolating in the Characteristics Database table finds that 48,086 MWd/MTU burnup will give 850 watts per assembly.

The criticality design point is selected to fall well within the range of those forecast SNF characteristics that give the target $k_\infty = 1.13$. In particular, the burnup was chosen so that the enrichment that gives $k_\infty = 1.13$ falls in the middle of the range of actual enrichment values for fuel having that burnup. For 10-year age, the enrichment-burnup pair that satisfies this condition is 3 percent, 20,000 MWd/MTU. These design basis recommendations are summarized in Table 5-2.

Table 5-2. Recommended Pressurized Water Reactor Design Basis Fuel for 96 Percent Combined Coverage

Parameter	Thermal/Shielding	Criticality
Burnup	48,086 MWd/MTU	20,000 MWd/MTU
Initial enrichment	4.2%	3%
Age	10 years	10 years
Oldest-fuel-first Coverage (96.16%)	97.85%	98.31%

It should be noted that although the DBFs specified in Table 5-2 are characterized in terms of age, burnup, and initial enrichment, the percentile figures (the last line of the table, labeled oldest-fuel-first coverage) refer to coverage of the parameters heat per assembly and k_∞; they do not refer to burnup and initial enrichment.

5.2.2 Recommended Boiling Water Reactor Design Basis Fuel

For BWR fuel, there is presently no parameterization of k_∞, so percentiles have to be established for burnup and enrichment with some reasoning by analogy with PWR fuel. The combined tabulation of burnup and initial enrichment is given in Table 5-3. To use the analogy with PWR fuel, the combined tabulation of burnup and initial enrichment for PWR SNF, as given in Table 5-4 is needed.

For each table, the first row is the burnup bin limit in GWd/MTU, the second row is the percent of BWR fuel having lower burnup, the last column is initial enrichment, the next to last column is the percent of BWR fuel having higher value of initial enrichment, and the rest of the table values are the percent of fuel having both lower value of burnup and higher value of initial enrichment.

Table 5-3. Percentiles for Boiling Water Reactor Combined Burnup and Enrichment (Used for Criticality Design Basis)

Burnup Bin Limit in gigawatt days (GWd)/MTU																	
30	29	28	27	26	25	24	23	22	21	20	19	18	17	16	15		
Percent of Boiling Water Reactor Fuel With ≤ Burnup																	
40.95	38.11	35.11	31.86	29.03	26.89	24.13	21.23	18.79	16.92	15.27	14.14	12.25	11.15	10.20	8.33		
Percent of Fuel With Both Lower Value of Burnup and Higher Value of Initial Enrichment																%≥enrc	enrch
38.78	35.95	32.94	29.69	26.86	24.72	21.96	19.06	16.62	14.76	13.11	11.97	10.08	8.98	8.03	6.16	97.83	1.50
38.51	35.67	32.67	29.42	26.59	24.45	21.69	18.79	16.35	14.48	12.83	11.70	9.81	8.71	7.76	5.89	97.56	1.55
38.51	35.67	32.67	29.42	26.59	24.45	21.69	18.79	16.35	14.48	12.83	11.70	9.81	8.71	7.76	5.89	97.56	1.60
38.29	35.46	32.46	29.21	26.38	24.24	21.48	18.58	16.14	14.27	12.62	11.49	9.60	8.50	7.65	5.87	97.35	1.65
38.29	35.46	32.46	29.21	26.38	24.24	21.48	18.58	16.14	14.27	12.62	11.49	9.60	8.50	7.65	5.87	97.35	1.70
38.20	35.37	32.37	29.11	26.28	24.14	21.38	18.48	16.04	14.18	12.53	11.39	9.50	8.40	7.55	5.78	97.25	1.75
36.75	33.92	30.92	27.67	24.83	22.70	19.94	17.04	14.59	12.73	11.11	9.97	8.13	7.17	6.46	4.90	95.80	1.80
36.62	33.78	30.78	27.53	24.70	22.56	19.80	16.90	14.46	12.60	10.98	9.84	8.01	7.05	6.39	4.86	95.67	1.85
36.61	33.78	30.78	27.52	24.69	22.56	19.79	16.90	14.46	12.59	10.98	9.84	8.00	7.05	6.39	4.86	95.66	1.90
36.53	33.70	30.69	27.44	24.61	22.47	19.71	16.81	14.37	12.51	10.89	9.76	7.92	6.97	6.31	4.77	95.58	1.95
36.51	33.68	30.67	27.42	24.59	22.45	19.69	16.79	14.35	12.49	10.87	9.74	7.90	6.94	6.29	4.75	95.56	2.00
35.88	33.05	30.05	26.79	24.03	21.89	19.13	16.23	13.99	12.13	10.51	9.57	7.73	6.94	6.29	4.75	94.93	2.05
35.21	32.38	29.38	26.13	23.36	21.22	18.46	15.56	13.32	11.50	9.91	8.98	7.23	6.49	5.93	4.41	94.24	2.10
30.65	27.82	24.82	21.57	18.80	16.66	14.05	11.33	9.43	8.04	6.78	5.93	4.73	4.22	3.78	2.98	89.68	2.15
26.94	24.11	21.10	17.99	15.36	13.50	11.21	8.54	6.72	5.58	4.81	4.39	3.74	3.43	3.00	2.55	85.97	2.20
25.37	22.56	19.62	16.51	13.98	12.18	9.89	7.82	6.03	4.94	4.43	4.10	3.45	3.15	2.81	2.36	84.40	2.25
24.52	21.73	18.79	15.68	13.15	11.35	9.07	7.07	5.28	4.31	3.87	3.58	2.97	2.72	2.47	2.16	83.50	2.30
24.01	21.21	18.28	15.17	12.64	10.84	8.57	6.62	4.87	3.96	3.67	3.42	2.87	2.70	2.45	2.16	82.99	2.35
23.85	21.05	18.12	15.02	12.49	10.77	8.50	6.54	4.80	3.89	3.59	3.35	2.82	2.65	2.43	2.15	82.83	2.40
23.75	20.95	18.02	14.92	12.39	10.67	8.39	6.44	4.70	3.78	3.49	3.25	2.72	2.55	2.33	2.05	82.73	2.45
22.50	19.70	16.82	13.77	11.31	9.68	7.70	5.92	4.21	3.67	3.37	3.16	2.72	2.55	2.33	2.04	81.48	2.50
18.72	15.95	13.09	10.19	8.05	6.68	5.12	4.34	3.33	2.85	2.63	2.44	2.26	2.12	1.93	1.69	77.59	2.55
18.65	15.88	13.02	10.11	7.98	6.59	5.05	4.34	3.33	2.85	2.63	2.44	2.26	2.12	1.93	1.69	77.42	2.60
16.30	13.79	11.22	8.78	6.86	5.77	4.56	3.88	3.09	2.77	2.56	2.37	2.25	2.11	1.93	1.69	74.71	2.65
13.76	11.38	9.13	7.31	5.79	4.99	3.96	3.33	2.77	2.52	2.38	2.20	2.13	2.03	1.84	1.61	71.86	2.70
11.75	9.47	7.46	5.92	4.85	4.26	3.52	3.15	2.62	2.47	2.33	2.19	2.12	2.02	1.83	1.60	69.42	2.75
10.69	8.83	6.99	5.45	4.48	3.91	3.28	3.03	2.60	2.47	2.33	2.19	2.12	2.02	1.83	1.60	66.76	2.80
7.74	6.42	5.29	4.38	3.88	3.45	2.94	2.73	2.35	2.26	2.12	1.98	1.91	1.80	1.62	1.42	62.67	2.85
5.93	4.95	4.13	3.56	3.08	2.82	2.62	2.47	2.30	2.23	2.11	1.97	1.90	1.80	1.62	1.41	60.08	2.90
5.62	4.75	3.93	3.37	2.89	2.72	2.60	2.45	2.28	2.23	2.11	1.97	1.90	1.80	1.62	1.41	58.95	2.95
4.31	4.07	3.58	3.18	2.78	2.61	2.53	2.41	2.27	2.23	2.11	1.97	1.90	1.80	1.62	1.41	53.99	3.00
3.63	3.49	3.17	2.85	2.56	2.39	2.32	2.20	2.06	2.02	1.91	1.76	1.70	1.59	1.41	1.21	49.28	3.05
3.21	3.08	2.83	2.65	2.42	2.25	2.19	2.07	1.93	1.90	1.85	1.71	1.64	1.59	1.41	1.21	46.83	3.10
3.04	2.90	2.69	2.50	2.28	2.12	2.06	1.94	1.80	1.77	1.72	1.58	1.52	1.47	1.28	1.08	44.50	3.15
2.77	2.63	2.47	2.28	2.06	1.90	1.84	1.72	1.67	1.64	1.59	1.45	1.44	1.39	1.21	1.01	41.45	3.20
2.42	2.30	2.16	1.98	1.75	1.59	1.54	1.42	1.36	1.33	1.29	1.22	1.21	1.16	0.98	0.78	36.07	3.25
1.77	1.74	1.60	1.45	1.42	1.28	1.23	1.11	1.05	1.02	0.98	0.91	0.90	0.85	0.82	0.62	30.55	3.30
1.69	1.66	1.52	1.44	1.41	1.27	1.22	1.10	1.05	1.01	0.97	0.91	0.89	0.85	0.82	0.62	23.54	3.35
1.68	1.65	1.51	1.44	1.41	1.27	1.22	1.10	1.05	1.01	0.97	0.91	0.89	0.85	0.82	0.62	20.32	3.40
1.29	1.27	1.14	1.07	1.06	0.99	0.94	0.86	0.85	0.82	0.80	0.74	0.72	0.68	0.65	0.45	17.29	[illegible]

Table 5-4. Pressurized Water Reactor Combined Burnup and Enrichment (Used for Comparison Estimate of Boiling Water Reactor Criticality Design Basis)

Burnup Bin Limit in GWd/MTU									
22	21	20	19	18	17	16	15		
Percent of PWR Fuel With ≤ Burnup									
6.74	6.17	5.79	5.35	4.72	4.05	3.23	2.55		
Percent of Fuel With Both Lower Value of Burnup and Higher Value of Initial Enrichment								% > enrch	enrch
6.58	6.01	5.63	5.19	4.57	3.90	3.08	2.41	99.82	1.50
6.51	5.94	5.56	5.12	4.50	3.83	3.02	2.34	99.75	1.55
6.51	5.94	5.56	5.12	4.50	3.83	3.02	2.34	99.75	1.60
6.17	5.61	5.23	4.80	4.18	3.51	2.69	2.03	99.41	1.65
6.10	5.54	5.16	4.72	4.11	3.44	2.62	1.96	99.34	1.70
6.02	5.46	5.08	4.65	4.03	3.36	2.55	1.88	99.26	1.75
6.02	5.46	5.08	4.65	4.03	3.36	2.55	1.88	99.26	1.80
6.02	5.46	5.08	4.65	4.03	3.36	2.55	1.88	99.26	1.85
5.54	4.98	4.62	4.19	3.60	2.93	2.12	1.49	98.73	1.90
5.04	4.48	4.12	3.71	3.13	2.46	1.79	1.22	98.19	1.95
4.91	4.35	4.00	3.61	3.05	2.46	1.79	1.22	98.06	2.00
4.70	4.15	3.83	3.43	2.88	2.29	1.65	1.10	97.75	2.05
3.91	3.39	3.11	2.78	2.38	1.86	1.44	0.98	96.92	2.10
3.04	2.57	2.30	1.99	1.61	1.17	0.96	0.91	95.90	2.15
3.02	2.56	2.29	1.98	1.60	1.17	0.96	0.91	95.83	2.20
2.76	2.39	2.11	1.80	1.53	1.16	0.95	0.91	95.51	2.25
2.47	2.10	1.83	1.53	1.34	1.12	0.95	0.90	94.99	2.30
2.41	2.10	1.83	1.53	1.34	1.12	0.95	0.90	94.82	2.35
2.37	2.10	1.82	1.53	1.33	1.12	0.95	0.90	94.48	2.40
2.15	1.87	1.60	1.37	1.17	0.96	0.79	0.74	93.93	2.45
2.08	1.80	1.56	1.34	1.15	0.94	0.77	0.74	93.72	2.50
2.00	1.75	1.55	1.34	1.15	0.94	0.77	0.74	93.51	2.55
1.87	1.69	1.52	1.34	1.15	0.94	0.77	0.74	92.70	2.60
1.77	1.60	1.46	1.28	1.11	0.92	0.77	0.74	91.19	2.65
1.76	1.59	1.45	1.28	1.11	0.92	0.77	0.74	91.01	2.70
1.72	1.56	1.42	1.27	1.11	0.92	0.77	0.74	90.06	2.75
1.65	1.50	1.37	1.24	1.08	0.89	0.77	0.74	89.31	2.80
1.64	1.49	1.37	1.23	1.07	0.88	0.77	0.74	88.45	2.85
1.63	1.48	1.36	1.23	1.07	0.88	0.77	0.74	87.86	2.90

Table 5-4. Pressurized Water Reactor Combined Burnup and Enrichment (Used for Comparison Estimate of Boiling Water Reactor Criticality Design Basis) (Continued)

Burnup Bin Limit in GWd/MTU									
22	21	20	19	18	17	16	15		
Percent of PWR Fuel With ≤ Burnup									
6.74	6.17	5.79	5.35	4.72	4.05	3.23	2.55		
Percent of Fuel With Both Lower Value of Burnup and Higher Value of Initial Enrichment								% > enrch	enrch
1.58	1.44	1.34	1.22	1.07	0.88	0.76	0.74	86.66	2.95
1.48	1.35	1.26	1.14	0.98	0.79	0.68	0.65	85.78	3.00
1.42	1.29	1.20	1.08	0.98	0.79	0.67	0.65	83.55	3.05
1.26	1.14	1.05	0.94	0.84	0.65	0.53	0.51	81.31	3.10
1.12	1.01	0.99	0.94	0.83	0.65	0.53	0.50	79.73	3.15
1.02	0.90	0.89	0.88	0.78	0.64	0.53	0.50	78.26	3.20
1.00	0.89	0.88	0.87	0.78	0.64	0.52	0.50	75.25	3.25
0.98	0.87	0.86	0.85	0.76	0.63	0.51	0.49	73.06	3.30
0.97	0.86	0.86	0.85	0.76	0.63	0.51	0.49	72.23	3.35
0.97	0.86	0.86	0.85	0.76	0.63	0.51	0.49	70.28	3.40
0.91	0.80	0.80	0.79	0.71	0.61	0.50	0.47	66.81	3.45

The thermal requirement for BWR fuel is less stressing than for PWR fuel. The BWR fuel that would be thermally equivalent to the PWR design basis is as follows:

1. The total heat for a WP of 21 PWR assemblies at 850 watts each is 17.85 kilowatts; to get the same amount of heat from a 44 BWR uncanistered fuel waste package or a 40 BWR conceptual multi-purpose canister waste package (the present nominal BWR designs) would require over 400 watts per assembly, which is higher than any actual BWR assembly, indicating that 100 percent of the fuel is covered by such a requirement.

2. The largest BWR burnup in the Environmental Information Agency database (49 GWd/MTU) is used. For this reason, it is unnecessary to select a mass per assembly for BWR SNF. This corresponds to 100 percent of the fuel and generates only 388 watts per assembly at 10 years after discharge. Thus, BWR SNF can be more easily accommodated by a design. The BWR fuel simply cannot be as stressing as the PWR thermal DBF.

The criticality requirement for BWR fuel may be more stressing than for PWR fuel because of the greater prevalence of low burnup assemblies. This possibility cannot be accurately assessed at this time because there is not yet a criticality parameterization such as the PWR formula for k_∞ given in Section 5.1. Until such a formula is developed, the BWR criticality DBF will be determined with respect to the PWR criticality DBF. By analogy with the PWR criticality, an age of 10 years and an

initial enrichment of 3 percent is used. For these parameters the smallest burnup that can be found in the Characteristics Database is 21,000 MWd/MTU, which is slightly larger than the 20 GWd/MTU used for PWR, even though the average burnup for PWR is much higher than for BWR. The fraction of BWR fuel covered by these parameters is then estimated according to the following methodology.

For a criticality design point specified by age, burnup, and initial enrichment, all the fuel having higher enrichment and lower burnup will have a higher k_∞ and higher k_{eff}. However, because parameters of burnup and enrichment have opposing affects on k_{eff}, there can be a higher k_{eff} with higher enrichment and higher burnup or with lower enrichment and lower burnup, provided the k_{eff} increasing parameter dominates the k_{eff} decreasing parameter. To estimate the actual percentage of BWR fuel not covered by such a design basis, the following analysis is used by analogy with PWR fuel.

For the PWR criticality design basis facility DBF specified by 20,000 MWd/MTU and 3 percent initial enrichment, 1.69 percent of the fuel will have a higher k_∞ (from Table 5-2, 3rd column, last row). From Table 5-4, 3rd column, 10th from last row, it can be seen that 1.26 percent of PWR fuel will have both a higher enrichment and lower burnup. Thus, the ratio of the amount of fuel having a higher k_∞ to the fuel having a higher enrichment and lower burnup is 1.69/1.26 = 1.34. It is assumed that the ratio of BWR SNF having a higher k_∞ (than the DBF) to the fuel having a higher enrichment and lower burnup (than the DBF) will be the same as for PWR SNF. Applying this ratio to the percentage of BWR fuel having higher enrichment and lower burnup than the design basis (20,000 MWd/MTU, 3 percent initial enrichment, which is 2.11 percent of the BWR fuel from Table 5-3), the result is 2.11x1.34 = 2.82 percent as the estimate of the fuel that will have higher k_∞ than the design basis, which means that this DBF will be more stressing, with respect to criticality, than 97.2 percent of the fuel. Since the thermal/shielding design basis covers 100 percent of the BWR fuel, the combined coverage is simply the same as the criticality coverage. These parameters are summarized in Table 5-5.

Table 5-5. Recommended Boiling Water Reactor Design Basis Fuel (Approximately 97.2 Percent Combined Coverage)

Parameter	Thermal/Shielding	Criticality
Burnup	49,000 MWd/MTU	20,000 MWd/MTU
Initial enrichment	3.74%	3.0%
Age	10 years	10 years
Oldest-fuel-first Coverage (97.2%)	100%	97.2%

5.3 CONCLUSIONS

The recommended DBF has been given in Tables 5-2 and 5-5. This DBF will cover 96.1 percent of the PWR SNF, and 97.2 percent of the BWR SNF expected under the oldest-fuel-first acceptance strategy. WP design will use these parameter values in the analyses, which will demonstrate compliance with applicable requirements. SNF not covered (more stressing values for thermal and criticality parameters) may require additional analysis with more stressing parameters to demonstrate compliance, which may, in turn, require derating (or blending of fuel, or supplemental criticality control feature) to enable utilization of the WP designed according to the DBF recommended in this document.

5.4 UNRESOLVED ISSUES/FUTURE WORK

The main unresolved issue is the parameterization of k_∞ for BWR fuel. A lesser issue is extending the parameterization for both PWR and BWR fuels to ages beyond 20 years. It is expected that this will be accomplished in 1996. The results are not expected to be significantly different from the assumptions used in this analysis, and are not expected to change the DBF. However, any differences at all will require a revision of the *Waste Package Design Basis Fuel Analysis* (CRWMS M&O, 1995c) to show any changes in the percent coverage provided by the DBF. A revision will also be necessary if the project strategy determines an acceptance policy other than the oldest-fuel-first assumption used here.

6. WASTE PACKAGE DESIGN CONCEPTS

A goal of the Engineered Barrier Segment/Waste Package Development Department for the Yucca Mountain Site Characterization Project (YMP) is to develop Waste Package (WP) designs and support the development of the Engineered Barrier System requirements that meet the applicable regulatory requirements for safe disposal of spent nuclear fuel (SNF) and vitrified high-level waste (HLW) in a geologic repository. The definitions for the WP and Engineered Barrier System are taken directly from the Code of Federal Regulations (CFR), 10 CFR 60.2. The WP is defined to include "the waste form and any containers, shielding, packing, and other absorbent materials immediately surrounding an individual waste container. "The Engineered Barrier System" means the WP and the underground facility, where the underground facility is defined as "the underground structure, including openings and backfill materials, but excluding shafts, boreholes, and their seals." The Engineered Barrier Segment includes the emplaced WPs (the waste forms and any containers, shielding, packing, and other absorbent materials immediately surrounding individual waste containers) and any other engineered devices or material used to provide waste isolation.

The results presented in this section are in support of the *Waste Package Implementation Plan* (YMP 1995) and are based on the *Engineered Barrier Design Requirements Document* (YMP 1994b). The evaluations and results support the goals of the WP designs and processes as defined in these plan and requirements documents. Development of the *Controlled Design Assumptions* document (CRWMS M&O 1995n) has provided the several conceptual design teams with a common assumptions list. The *Controlled Design Assumptions* document is an accumulation of engineering assumptions and "Key Assumptions" derived from the requirements set forth by the U.S. Department of Energy (DOE). In addition, specific design goals have been established to guide the development of WP designs and to minimize potential risk to the public. The basic design requirements and design goals provided in Sections 2 and 3 are highlighted below.

Engineered Barrier Design Requirements Document Regulations

- Substantially complete containment for 300 to 1000 years
- Release requirement(s). Release rate of radionuclides from Engineered Barrier System/WP of 1 part in 100,000 per year of inventory at 1,000 years after emplacement; this requirement applies following the containment period of 300 to 1000 years
- Criticality (k_{eff} less than 0.95)
- Retrievability (. . . up to 50 years after initial emplacement. . .)
- 10 CFR 20 for worker dose.

Design Goals

- Centerline fuel rod temperature less than 350°C was adopted. This limit is maintained to permit performance credit for the fuel rod cladding. (to be verified [TBV])
- Topopah Spring welded unit 2 (TSw2) rock temperature limit of 200°C (1 m into the rock for borehole) (TBV)

- Thermal loading of the repository (to be determined [TBD])

- Reliability (TBD)
  - Design
  - Fabrication

- Engineered Barrier Segment/WP surface radionuclide dose (TBD).

For the conceptual design phase, engineering evaluations were conducted on selected WP and associated Engineered Barrier designs to establish their ability to satisfy design, materials, and performance requirements based on selected reference sets of the near-field environment and waste characteristics. Consideration is also being given to the WP and engineered barrier manufacturing processes anticipated for fabrication; repository procedures for WP closure, including postweld heat treatment and postweld inspection; and post-emplacement, remote, in-service-inspection of WPs selected for the performance confirmation program (see Section 9).

WP concepts were evaluated and documented for SNF and vitrified glass HLW. Early in the design cycle, a primary and alternate were selected from the conceptual design concepts for the canistered fuel (CF) disposal container, an uncanistered SNF (referred to as UCF) WP, and a primary concept for the defense HLW package. The current repository system is based upon the implementation of a CF container as the prime SNF waste handling concept throughout the waste management system as presented in Volume I of the Mined Geologic Disposal System (MGDS) Advanced Conceptual Design (ACD) report. However, to provide operational flexability and an ability to handle bare SNF at the repository, the UCF WP concept has been developed as a design option. The current WP concepts to be used with the selected drift emplacement MGDS concept are

1. CF with Disposal Container (WP)
2. Defense High-Level Waste (DHLW) WP
3. UCF WP

The CF container is a concept which includes dual-purpose or multipurpose types of canisters. In order to assess the performace of the CF container in a repository environment the Multi-Purpose Canister (MPC) conceptual design (CRWMS M&O 1994k) was selected to be the basis for evaluation of the CF container concepts. Thus, the use of the term MPC and CF for the purpose of this section are interchangable. These three WP design concepts support the DOE-fostered "Focused Design" effort. A brief description of the designs for these three concepts is provided in Section 1.2, with detailed design sketches provided in Appendix B, and the supporting analysis presented in Sections 6.3, 6.4, and 6.5. Section 6.1 provides a detailed description of the WP element of the Engineered Barrier Segment. Section 6.2 provides the MGDS/Engineered Barrier Segment analysis basis upon which the Engineered Barrier Segment/WP analysis is built. The analysis basis presented in Section 6.2 applies equally to each WP design.

6.1 WASTE PACKAGE CONCEPTUAL DESIGN DESCRIPTIONS

Over the repository lifetime, the WP containment barriers will perform various functions that will change with time. During the 100-year or longer operational period, the barriers will function as the vessel for handling, emplacing, and retrieving (if necessary) the contents of the WP. This will be followed by the 1,000-year containment period, and then the period of controlled release extending beyond 10,000 years. During the first 1,000 years following repository closure, the containment barriers will be relied upon to provide substantially complete containment of the radionuclides, and to impede release of radionuclides by aqueous transport from those WPs that have breached during the subsequent controlled release period. However, breached containment barriers are expected to continue to inhibit transport of liquid water into, and radionuclides out of, the WPs. The use of a multibarrier design will result in a lower breach rate distributed over a longer period of time. This will ensure that the release rate limits are met.

6.1.1 Waste Package General Design Features

The functions of the multibarrier shell are to provide a containment barrier, act as a structural member subject to external loadings, heat conduction, and provide sufficient shielding to limit the radiation induced corrosion (radiolysis) of the containment barriers. The functions of the UCF basket are structural support, heat conduction, and criticality control.

Although adhering closely to the codes, the disposal containers (including basket) will not be designed and built to any American Society of Mechanical Engineers pressure vessel codes, as the disposal containers do not function as pressure vessels. Consequentially, the disposal containers will not be N-Stamped. Disposal container materials will conform to American Society for Testing and Materials standards and may be specified to be American Society of Mechanical Engineers Code Case materials, but there will be no blanket requirement to use American Society of Mechanical Engineers Code Case materials in the disposal containers.

The shell of the metallic multibarrier disposal containers consists of a corrosion-resistant inner barrier plus a corrosion-allowance outer barrier. In all cases, the inner barrier is 20 mm thick Alloy 825. Outer barrier design depends on repository thermal load design conditions. The outer barrier for both the CF and UCF disposal container is 100 mm thick A 516 carbon steel for the high thermal load, dry repository; and, as a function of WP size and mass, is 60 to 65 mm thick C71500 cupronickel for the low thermal load, humid repository. The DHLW disposal container outer barrier is 50 mm thick C71500 for either thermal load design condition. C70600 (nominally 90/10 CuNi) will be examined as a somewhat lower cost alternative to C71500 (nominally 70/30 CuNi) for outer barrier material. Selection of weld filler materials appropriate to the materials of construction for the waste package will be a subject of development programs for waste package closure welds as well as disposal container fabrication (including cladding of the inner barrier to outer barrier).

Fabrication of the cylindrical multibarrier shell would be by one of the following: (1) ring(s) formed by roll-and-welded plate, forging, or casting to form the cylindrical outer barrier, with the inner barrier subsequently added as clad by welding, (2) two separately fabricated cylinders designed to fit loosely within one another, or (3) concentrically clad using centrifugal casting (not currently an option, as the Alloy 825 inner barrier material does not have a cast equivalent). Forgings and

castings have the advantage of producing seamless rings, whereas plate material would have to rolled and welded to form rings, from which to fabricate the outer barrier cylinder. The bottom lid(s) could either be clad, or two separate lids individually welded in place during fabrication. The top lids would not be made of clad material, and would be individually remotely welded in place after the appropriate disposal container is loaded with either a CF canister, individual UCF assemblies, or HLW canisters.

Three WP design configurations are required to dispose of the CF canisters, UCF, and HLW canisters. Besides the three configurations, there are consideration of size, fuel type, UCF basket configuration, and repository thermal load as it affects selection of WP outer barrier material.

The present requirement to consider multiple WP sizes results from two factors. In the case of CF, the two sizes are the largest that can be accommodated based on reactor facility existing crane load limitations. However, repository thermal loading studies continue to consider a range of WP thermal outputs, corresponding to various WP sizes (spent nuclear fuel capacities). Thus, to support the repository thermal loading studies, two different sizes of UCF disposal containers were investigated during ACD, having spent nuclear fuel capacities generally equivalent to the CF conceptual designs. The DHLW disposal container size (diameter), containing four glass canisters, corresponds generally to the large CF and UCF disposal container sizes.

All of the designs incorporate the multibarrier disposal container configuration. The same handling configuration is used for all waste containers, whether a CF, UCF, or DHLW disposal container. An extension of the waste container outer barrier, similarly on each end as a "skirt," results in the waste container ends being recessed. There are three holes equally spaced around each skirt in this design. As required, each set of holes may be engaged in various ways. For example, a fixture with radially actuated pins that move similar to the bolts in a safe door; or instead clevises may be affixed to the holes. Handling a waste container in the horizontal position may require a lifting fixture with shoes to engage each skirt extension as a "shelf." This configuration provides for handling the WPs both empty and after filling and closure. Alternative handling configurations will be investigated.

Handling features for the closure lids have yet to be detailed, but are envisioned to be a single, low-profile trunnion of equal size attached to each lid. The CF conceptual design lifting feature (CRWMS M&O 1994k) consists of six threaded blind holes in the outer lid; this feature would be used to lift the complete CF (even after the optional addition of filler material). The CF conceptual design did not indicate the design features for handling the CF shield plug.

6.1.2 WASTE PACKAGE DETAILED DESIGN FEATURES

Details of each WP design are shown on a number of figures, located in Appendix B. The figure series are shown in Table 6.1-1. The set of figures for each design begins with an isometric representation taken from the computer-aided design solid modeling, followed by a series of figures detailing the WP components and assemblies. Dimensional tolerancing has yet to be performed.

Table 6.1-1. Design Layouts

Figure Series Description	Figure Numbers
Large Waste Package Tube Design (21-PWR)	B.1-1 through B.1-12
Large Waste Package Tube Design (44-BWR)	B.1-13 through B.1-23
Small Waste Package Tube Design (12-PWR)	B.1-24 through B.1-34
Small Waste Package Tube Design (24-BWR)	B.1-35 through B.1-45
Large Low TL/Humid Waste Package Tube Design (21-PWR)	B.1-46 through B.1-57
Large Low TL/Humid Waste Package Tube Design (44-BWR)	B.1-58 through B.1-68
Small Low TL/Humid Waste Package Tube Design (12-PWR)	B.1-69 through B.1-79
Small Low TL/Humid Waste Package Tube Design (24-BWR)	B.1-80 through B.1-90
Defense High-Level Waste Disposal Container	B.2-1 through B.2-7
Large Canistered Fuel Disposal Container (21-PWR/40-BWR)	B.3-1 through B.3-4
Small Canistered Fuel Disposal Container (12-PWR/24-BWR)	B.3-5 through B.3-8
Large Low TL/Humid MPC Disposal Container (21-PWR/40-BWR)	B.3-9 through B.3-12
Small Low TL/Humid MPC Disposal Container (12-PWR/24-BWR)	B.3-13 through B.3-16

6.1.2.1 CANISTERED FUEL DISPOSAL CONTAINERS - WASTE PACKAGES

The CF conceptual designs (CRWMS M&O 1994k) resulted in just the two CF sizes, large and small (see Figures 6.1-1 and 6.1-2); the large CF size containing either 21 PWR or 40 BWR spent nuclear fuel assemblies, and the small CF containing either 12 PWR or 24 BWR spent nuclear fuel assemblies. The sealed CF canister will be placed into the disposal container, following which each of the two closure lids will be welded into place.

6.1.2.2 DEFENSE HIGH-LEVEL WASTE DISPOSAL CONTAINER-WASTE PACKAGE

This design addresses HLW from domestic fuel reprocessing plants, both commercial- and defense-related. Since the majority of this form of HLW is from the DOE defense-related programs, the MGDS organization has historically referred to this category of waste as DHLW. DHLW is defined as the highly radioactive material resulting from the reprocessing of SNF. This includes liquid waste produced directly in reprocessing any solid material derived from such liquid waste that contains fission products in sufficient concentrations, and other highly radioactive material that has been determined by the NRC, consistent with the law, as requiring permanent isolation. As used in this section, DHLW is defined to include commercial and defense HLW from reprocessing operations and not the SNF itself. Note that the 10 CFR 60 definition of HLW includes SNF. Figure 6.1-3 presents the major components of the DHLW WP.

The Savannah River Site HLW glass pour canister geometry was chosen as the basis for sizing the DHLW disposal container. The number of pour canisters per DHLW disposal container was chosen as four, to result in disposal container diameter approximately the same as for the large CF disposal container and UCF disposal container. The sealed pour canisters will be placed into the disposal container, following which each of the two closure lids will be welded into place.

6.1.2.3 UNCANISTERED FUEL WASTE PACKAGES

The tube type basket design has been selected as the preferred ACD design; the associated figures appear in Figures 6.1-4, 6.1-5, 6.1-6, 6.1-7.

The tube basket conceptual design is based on bundling together a number of identical full-length square tubes so as to form the necessary basket shape. PWR tubes would be formed by bending 5 mm thick 316B6A alloy sheet into a square tubular shape, approximately 4.6 m long. (It has been demonstrated that the approved 304B6A [Grade A] stainless-boron alloy may be bent around a radius 1.5 to 2 times material thickness, or about 8 to 9 mm radius.) A longitudinal weld (auto-Tungsten inert gas [tungsten inert gas] or electron beam) would be used to close the tube longitudinal seam.

The BWR tubes would be formed by bending 3.5 mm thick 316B6A alloy sheet into a square tubular shape. Compared to the alternative design approach of using 5 mm tube thickness and 316B4A alloy with reduced boron loading, the reduced thickness design results in substantial reduction of basket mass and cost, as the 316B6A alloy unit cost is only modestly higher than the 316B4A alloy.

Carpenter Technology Corporation of Reading, Pennsylvania is a producer of Type 304 stainless steel-boron alloys, both Grade A and Grade B. The Grade A material results in uniformly distributed fine borides, as it is manufactured by powder metallurgy techniques. The finely dispersed borides improve mechanical properties, workability, and weldability of the material as compared to the Grade B wrought material. Carpenter's testing has demonstrated that both grades are readily weldable using conventional stainless steel consumables. Both grades do experience reduction in impact resistance of the weld heat affected zone (with and without filler metal); however, this is to be expected, as the unalloyed stainless steel welds also experience reduction in impact resistance. Based on the foregoing experience, it is expected that weldability of stainless steel-boron alloy Types 316B6, 316B5, 316B4, etc. Grade A (e.g., 316B6A) would similar to that experienced with the 304 stainless steel-boron alloys.

Use of several interlocking aluminum plates is also introduced into the tube type conceptual design to provide an enhancement of heat transfer paths to the WP shell (thermal shunts). The plates also serve to separate the tubes into clustered assemblies.

Various angle-iron guides and other structural forms (made of the same material as the inner barrier, Alloy 825) are installed within the interior of the disposal container shell to provide basket structural support and to guide installation of the basket. Structural analyses of this basket support hardware will be performed in the future.

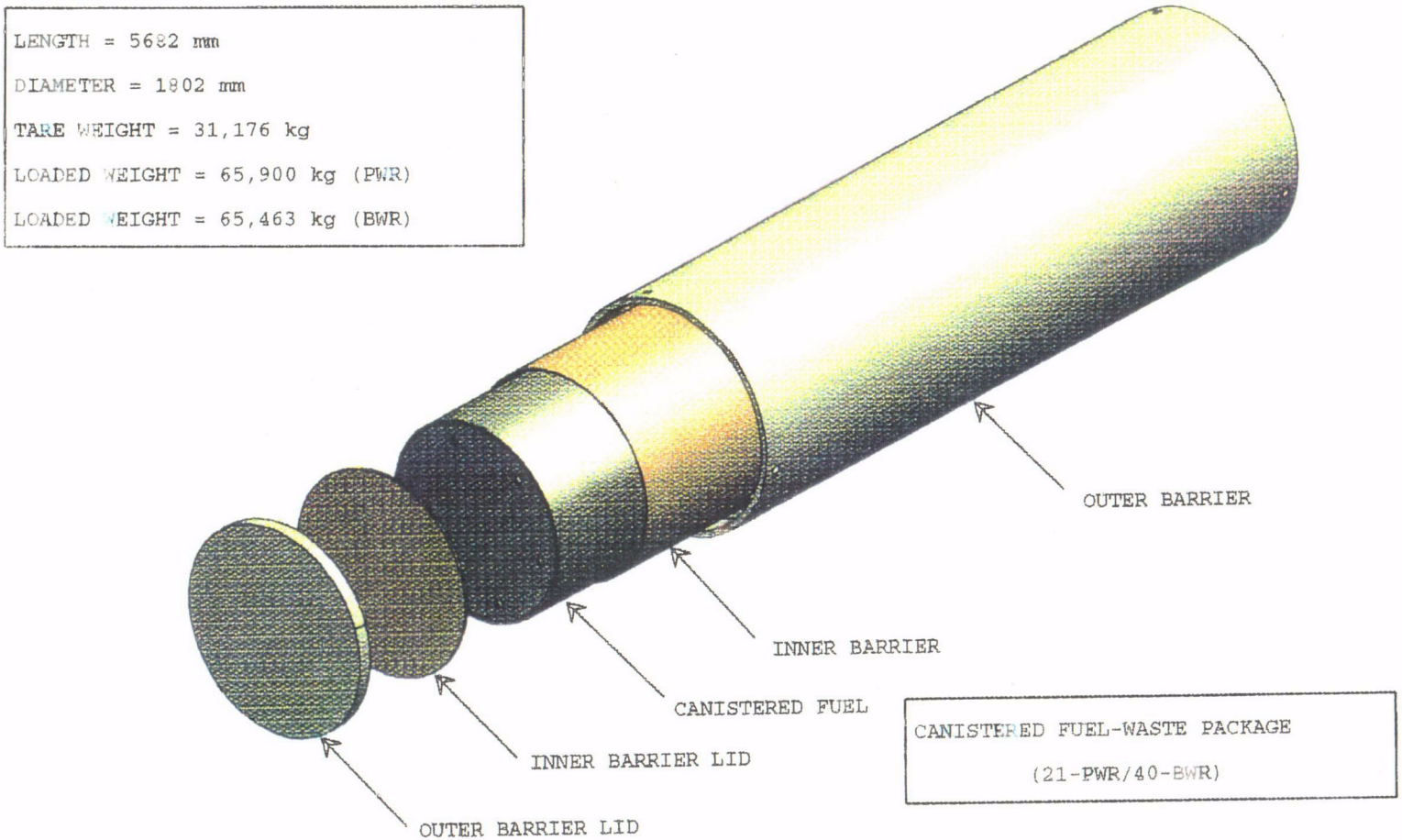


Figure 6.1-1. Large Canistered Fuel Waste Package Major Components

C01

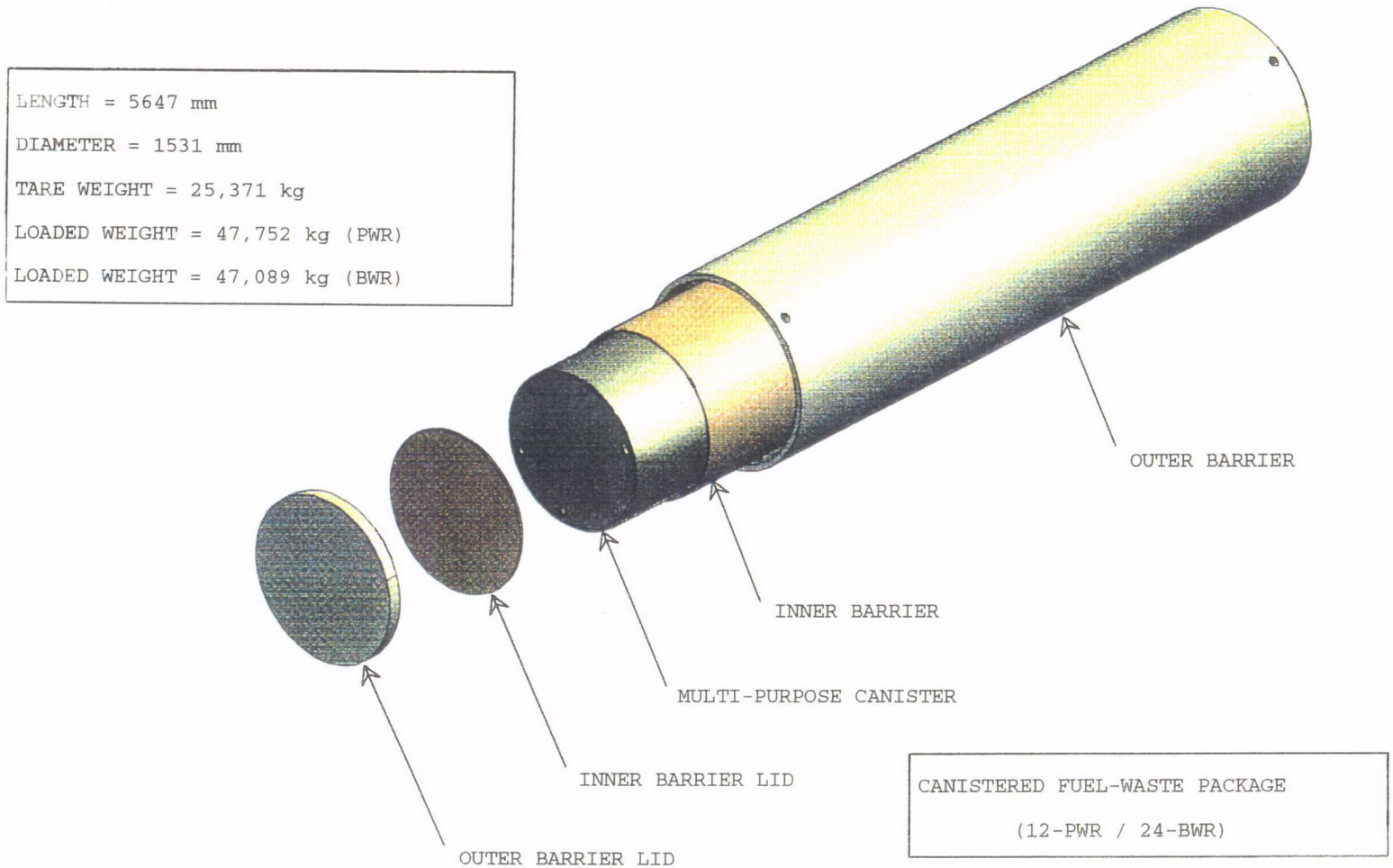


Figure 6.1-2. Small Canistered Fuel Waste Package Major Components

C02

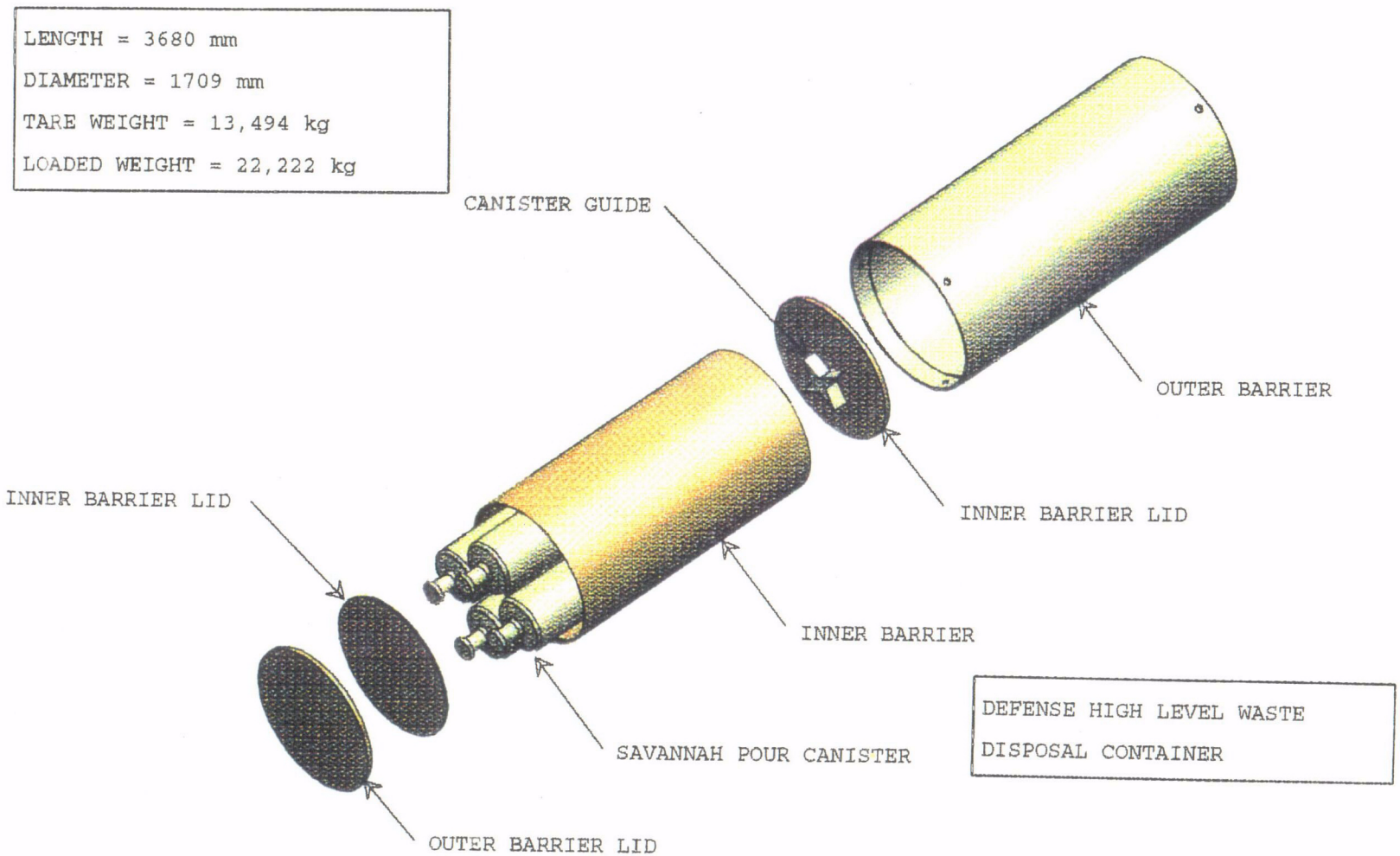


Figure 6.1-3. DHLW Waste Package Major Components

C03

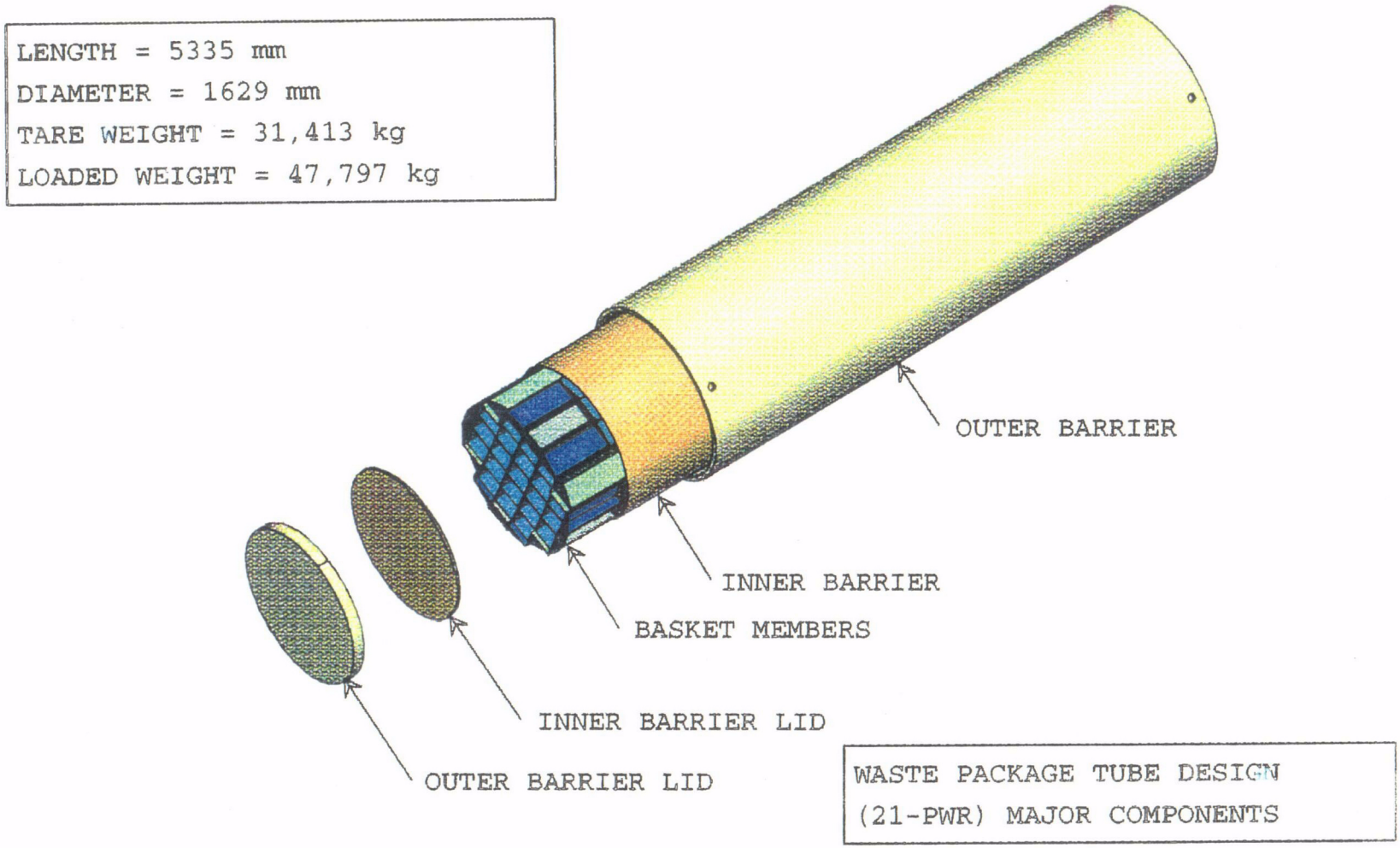


Figure 6.1-4. 21 PWR Uncanistered Fuel Waste Package Major Components

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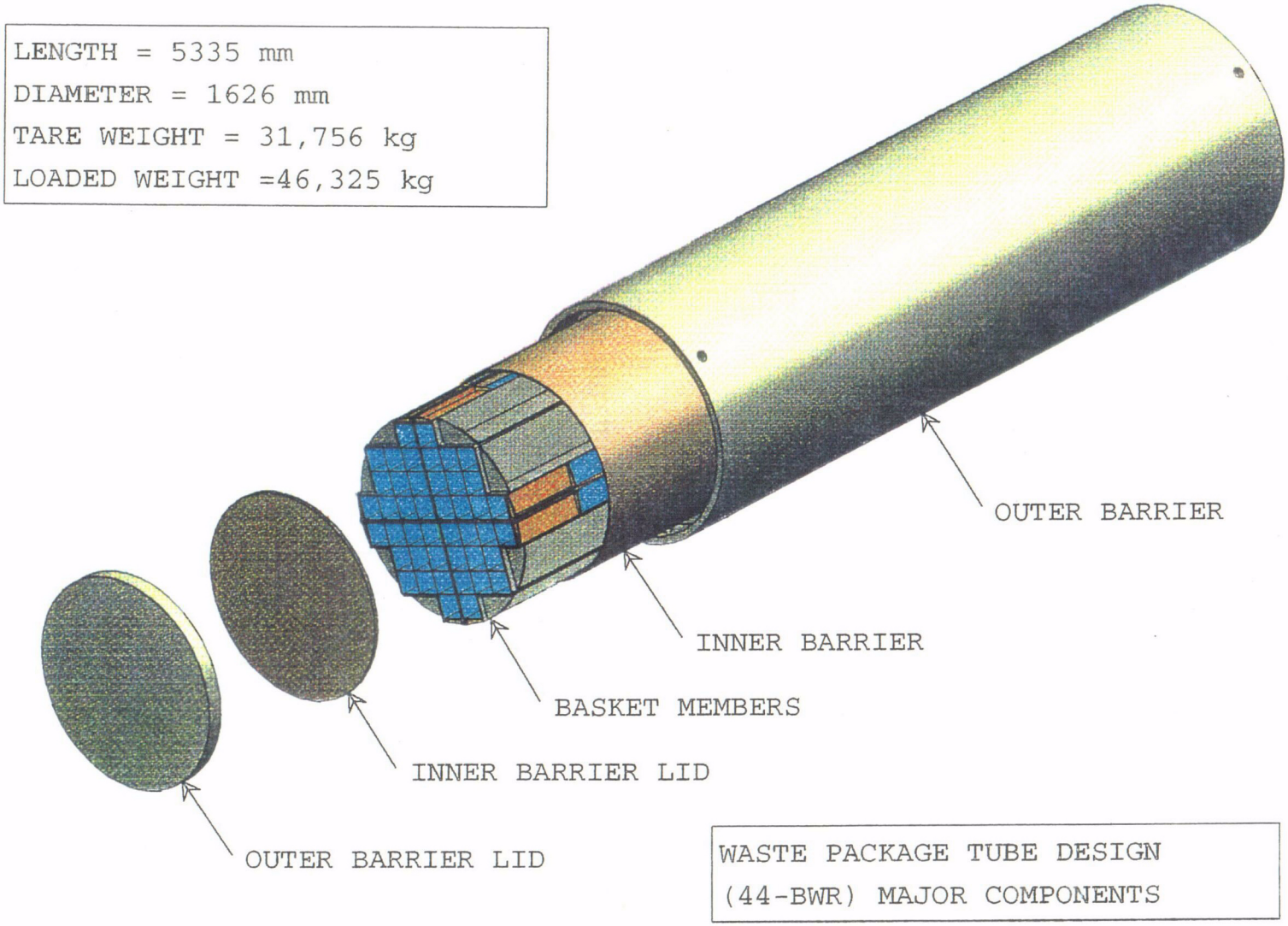

Figure 6.1-5. 44 BWR Uncanistered Fuel Waste Package Major Components

C05

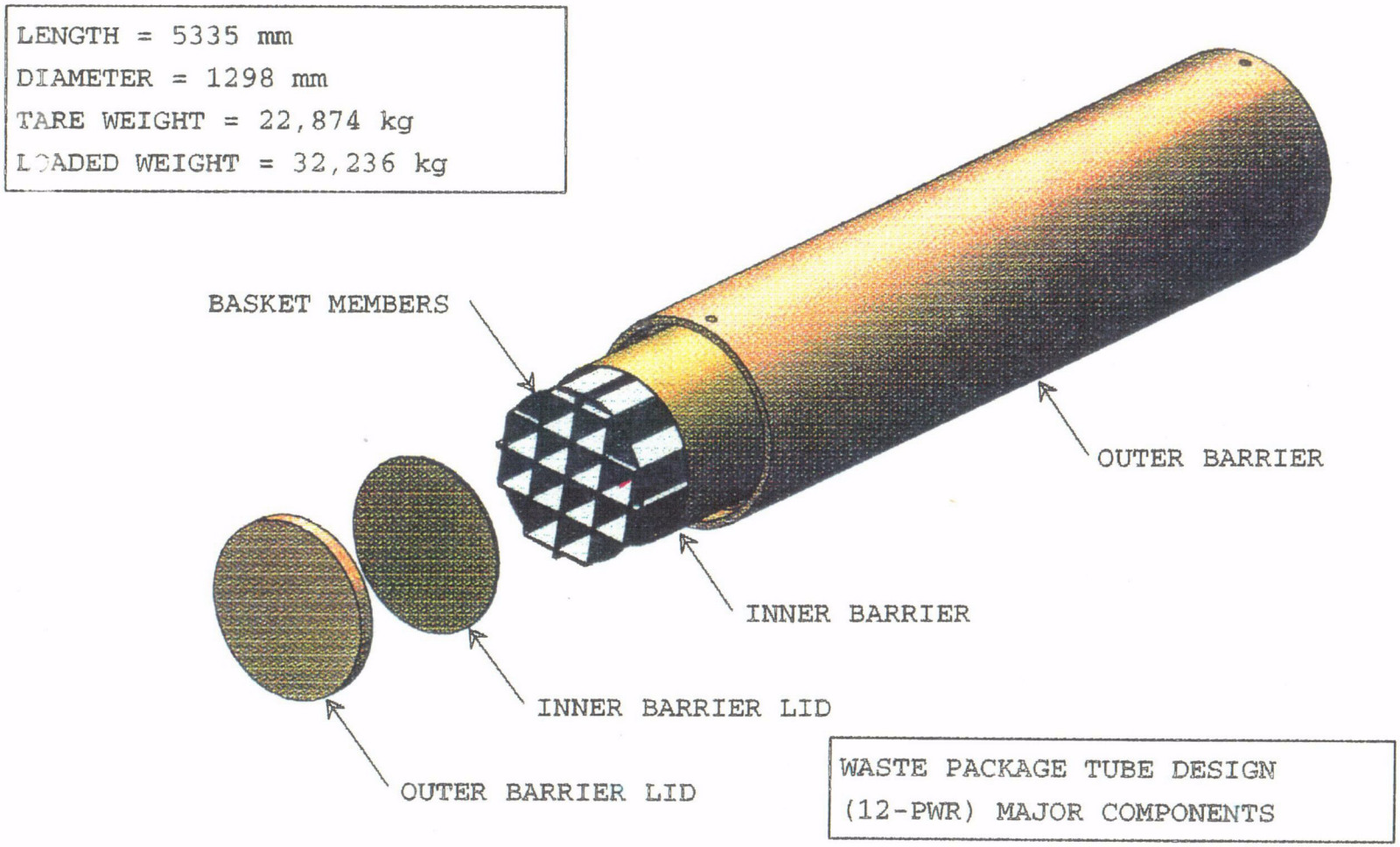

Figure 6.1-6. 12 PWR Uncanistered Fuel Waste Package Major Components

C06

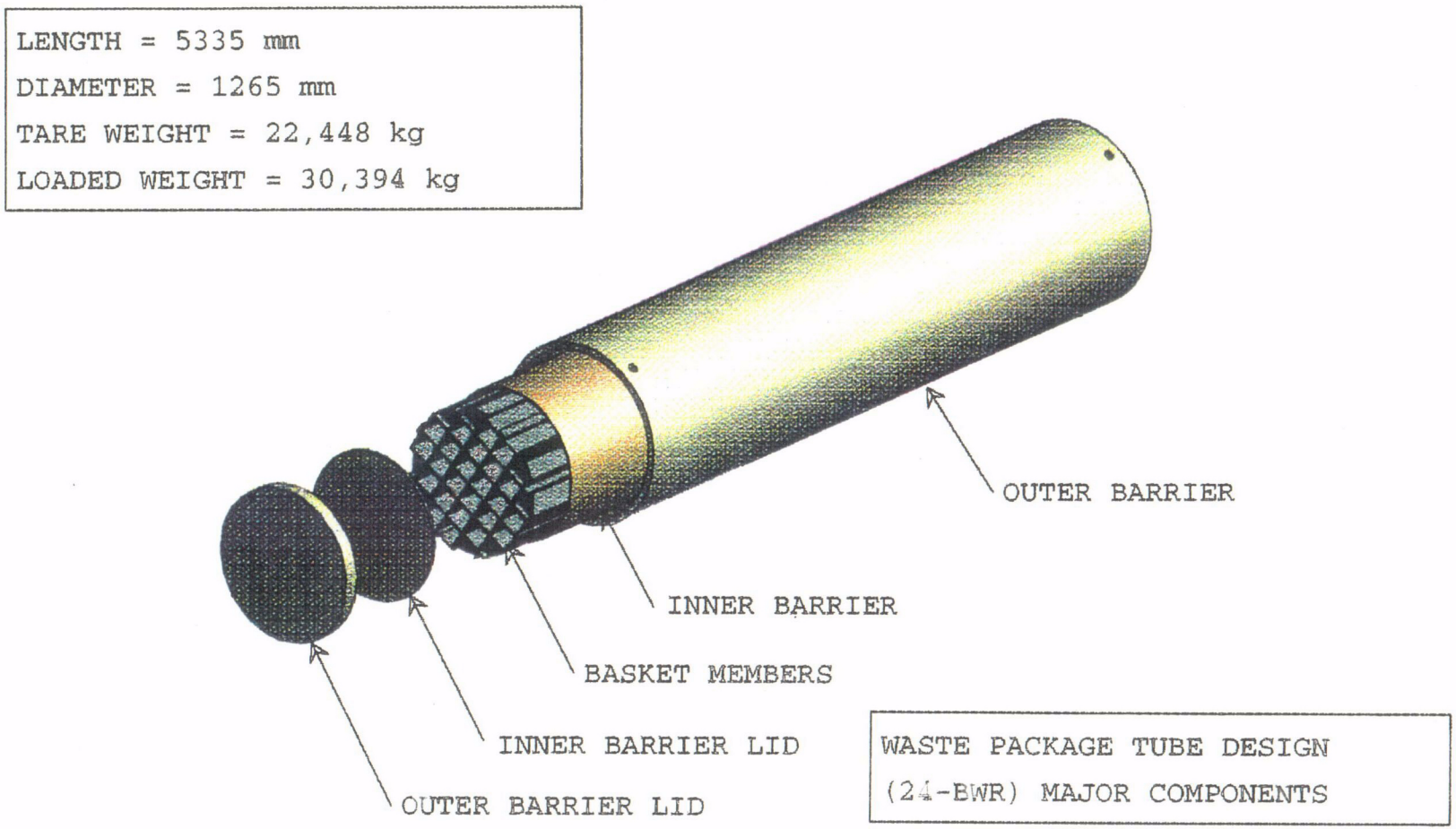

Figure 6.1-7. 24 BWR Uncanistered Fuel Waste Package Major Components

C07

6.1.3 Design Analysis Data

Several conceptual spent nuclear fuel WP designs were developed during ACD to accommodate both large and small WP sizes and to investigate alternative means of fabrication. These designs include CF, UCF, and DHLW disposal containers. Tables of WP dimensions and masses have been developed, which include most of the conceptual designs existing at the conclusion of ACD.

Table 6.1-2 presents waste container overall dimensions and masses for the case of high thermal loading/dry repository design conditions (carbon steel outer barrier). Table 6.1-3 presents the same information for the case of low thermal loading/humid repository design conditions (cupronickel outer barrier). Table 6.1-4 provides information broken down by component type and also provides information concerning filler weights.

Table 6.1-2. ACD WP Overall Dimensions, Masses, and Fabrication Costs - High Thermal Load/Dry Designs

Dimensions, Masses[1] Unit Costs, 1995$ Waste Package Configuration: Inner Barrier Clad to Outer Barrier[3]	21 PWR Canistered Fuel Waste Package (MPC CDR) (BU Credit)	40 BWR Canistered Fuel Waste Package (MPC CDR) (BU Credit)	12 PWR Canistered Fuel Waste Package (MPC CDR) (Flux Trap)	24 BWR Canistered Fuel Waste Package (MPC CDR) (BU Credit)	21 PWR UCF[2] Waste Package Tube Type (BU Credit)	44 BWR UCF[2] Waste Package Tube Type (BU Credit)	12 PWR UCF[2] Waste Package Tube Type (BU Credit)	24 BWR UCF[2] Waste Package Tube Type (BU Credit)	DHLW Waste Package (4 Canisters Savannah River)
Outside Diameter, D, mm	1,802	1,802	1,531	1,531	1,629	1,626	1,298	1,265	1,,709
Outside Length[1], L, mm	5,682	5,682	5,647	5,647	5,335	5,335	5,335	5,335	3680
Disposal Container Tare Mass, kg	31,176	31,176	25,371	25,371	31,413	31,756	22,874	22,448	13,494
MPC CDR, UCF, or DHLW Mass, kg	34,724	34,287	22,381	21,718	16,384	14,568	9,362	7,946	8,728
Loaded Mass without Filler, kg	65,900	65,463	47,752	47,089	47,797	46,325	32,236	30,394	22,222
Mass of Filler Material[4], kg	20,327	20,327	14,533	14,533	15,047	13,858	8,141	7,630	-
Mass including Filler Material, kg	86,227	85,790	62,285	61,622	62,844	60,183	40,378	38,024	-

Notes: BU = Burnup, MPC CDR = MPC Conceptual Design Report (CRWMS M&O 1994k)

1. Outside length includes reduced-thickness extensions of the outer barriers as "skirts" for handling purposes, extending 225 mm each end.
2. Tube basket design (tube wall thicknesses: PWR - 5 mm, BWR - 3.5 mm).
3. Inner barrier 20 mm Alloy 825, outer barrier 100 mm A 516 carbon steel for CF and UCF Waste Packages and 50 mm C71500 cupronickel for DHLW Waste Packages.
4. The use of filler material is a design option; filler material is assumed to be graded iron shot, filled to 85 percent of free space volume.

Table 6.1-3. ACD WP Overall Dimensions, Masses, and Fabrication Costs - Low Thermal Load/Humid Designs

Dimensions, Masses[1] Unit Costs, 1995$ Waste Package Configuration: Inner Barrier Clad to Outer Barrier[3]	21 PWR Canistered Fuel Waste Package (MPC CDR) (BU Credit)	40 BWR Canistered Fuel Waste Package (MPC CDR) (BU Credit)	12 PWR Canistered Fuel Waste Package (MPC CDR) (Flux Trap)	24 BWR Canistered Fuel Waste Package (MPC CDR) (BU Credit)	21 PWR UCF[2] Waste Package Tube Type (BU Credit)	44 BWR UCF[2] Waste Package Tube Type (BU Credit)	12 PWR UCF[2] Waste Package Tube Type (BU Credit)	24 BWR UCF (5) Waste Package Tube Type (BU Credit)	DHLW Waste Package (4 Canisters Savannah River)
Outside Diameter, D, mm	1,732	1,732	1,451	1,451	1,549	1,546	1,198	1,165	1,709
Outside Length[1], L, mm	5,606	5,606	5,559	5,559	5,247	5,247	5,225	5,225	3,680
Outer Barrier Thickness, mm	65	65	60	60	60	60	50	50	50
Disposal Container Tare Mass, kg	24,074	24,074	18,193	18,193	24,095	24,454	15,231	15,028	13,494
MPC CDR, UCF, or DHLW Mass, kg	34,724	34,287	22,381	21,718	16,384	14,568	9,362	7,046	8,728
Loaded Mass without Filler, kg	58,798	58,361	40,574	39,911	40,479	39,023	24,593	22,974	22,222
Mass of Filler Material[4], kg	20,327	20,327	14,533	14,533	15,047	13,858	8,141	7,630	-
Mass including Filler Material, kg	79,125	78,688	55,108	54,445	55,525	52,881	32,735	30,604	-

Notes: BU = Burnup, MPC CDR = MPC Conceptual Design Report (CRWMS M&O 1994k)

1. Outside length includes reduced-thickness extensions of the outer barriers as "skirts" for handling purposes, extending 225 mm each end.
2. Tube basket design (tube wall thicknesses: PWR - 5 mm, BWR - 3.5 mm).
3. Inner barrier 20 mm Alloy 825, outer barrier 100 mm A 516 carbon steel for CF and UCF Waste Packages and 50 mm C71500 cupronickel for DHLW Waste Packages.
4. The use of filler material is a design option; filler material is assumed to be graded iron shot, filled to 85 percent of free space volume.

Table 6.1-4. CF, UCF, and DHLW Disposal Container Masses

Number of SNF or DHLW Assy's	WP Type	Descriptor	First Barrier Mass (kg)	Second Barrier Mass (kg)	Third Barrier Mass (kg)	UCF WP Basket Mass (kg)	WP Basket Guide or HLW Struct. Mass (kg)	Tare Mass WP/CF Overpack Empty Mass (kg)	CF, SNF, or HLW Mass (kg)	Total Mass without filler (kg)	Avail. Void Vol. for filler (Cu. m)	Filler Material Mass (kg) Optional	Total Mass with Filler (kg)
4	DHLW WP	Savannah	3313	10160	0	0	21	13494	8728	22222	0.0000	0	22222
12	PWR CF	MPC	3869	21502	0	0	0	25371	22381	47752	3.2568	14533	62285
24	BWR CF	MPC	3869	21502	0	0	0	25371	21718	47089	3.2568	14533	61622
21	PWR CF	MPC	48353	26341	0	0	0	31176	34724	65900	4.5551	20327	86227
40	BWR CF	MPC	4835	26341	0	0	0	31176	34287	65463	4.5551	20327	85790
12	PWR T/WP	Tube Basket	2923	16711	0	2230	1010	22874	9362	32236	1.8244	8141	40378
24	BWR T/WP	Tube Basket	2822	16196	0	2140	1287	22448	7946	30394	1.7098	7630	38024
21	PWR T/WP	Tube Basket	3970	22009	0	4123	1311	31413	16384	47797	3.3718	15047	62844
44	BWR T/WP	Tube Basket	3960	21959	0	4142	1696	31756	14568	46325	3.1055	13858	60183

6.2 REPOSITORY/EMPLACEMENT SCALE ANALYSIS BASIS

Section 6.1 provided a summary of the WP designs. However, any analysis of specific WPs requires that a repository/emplacement analysis basis be performed to provide the required boundary conditions, environment conditions, a repository scale set of parameters, and a reasonable set of assumptions. Repository analysis bases for WP thermal design, criticality, and radiation shielding are discussed in Sections 6.2.1, 6.2.2, and 6.2.3, respectively. All the structural design parameters from the repository WP handling operations were not available at the time of preparation of this analysis. Since few of the repository operations have been firmly established, structural performance parameters cannot be addressed beyond the requirements specified in Sections 2 and 3. Thus, only those WP structural requirements will be addressed in Sections 6.3, 6.4, and 6.5. Additional requirements will be developed as the repository design proceeds into subsequent design phases and additional structural performance analysis will be performed.

Please note that a few of the analyses referenced in Section 6.2 were performed at a time when the repository subsurface design was not static and was slightly different than the current design. These analyses are used to show trends to be considered to understand why the current design was selected to move into subsequent design phases. These analyses, however, will be updated as the design moves forward, but are not yet available. Also, additional analysis results have been provided to represent a complete basis for the selection of the MGDS ACD design as presented in Volume I. The MGDS ACD design was selected based upon extensive alternative design evaluation and the additional analysis results have been provided in this volume to fully address the technical issues involved in selecting a design.

6.2.1 Thermal Analysis Basis

As part of an Engineered Barrier System for the containment of radionuclides, the WP must be shown to comply with all regulations and requirements that govern the conditions of the emplaced SNF and the near-field rock at the repository horizon. Temperatures in the WP and near-field host rock are key to radionuclide containment, as they directly affect oxidation rates of the metal barriers and the ability of the rock to impede particle movement. Postclosure release rates are regulated by 10 CFR 60, 10 CFR 960, and 40 CFR 191.

Maximum allowable temperatures are based on material performance criteria and are specified as design goals for the Engineered Barrier Segment/WP design. For SNF, the Commercial Spent Fuel Management Program (PNL 1983a) at Pacific Northwest Laboratory recommended a 380°C temperature limit on SNF cladding for five-year-cooled fuel to prevent creep rupture failure. A more conservative value of 350°C has been selected (CRWMS M&O 1993i) to account for uncertainties in source characteristics and heat transfer calculations. To limit the predicted thermal and thermo-mechanical response of the host rock and surrounding strata, maximum temperatures of 200°C for TSw2 (at the emplacement drift wall) and 115°C in the TSw3 (vitrophyre tuff) layer have been specified (see Section 3.22). Specific characteristics and requirements for the Engineered Barrier System are listed in the *Engineered Barrier Design Requirements Document* (YMP 1994b).

The method for WP thermal evaluations involves a three-model approach to determine the time-dependent WP thermal behavior. As depicted in Figure 6.2-1, a three-dimensional transient finite

element model of the WP emplacement (referred to as "Emplacement Model") provides the WP surface temperature history for use as a boundary condition in a detailed WP model. Resulting SNF basket wall temperature predictions from the detailed WP model provide the boundary conditions for an estimation of peak SNF cladding temperatures. Cladding temperatures are predicted in the WP model using a effective conductivity defined to represent an homogeneous SNF assembly. The effective conductivity was determined using a detailed third model of an intact SNF assembly (referred to as "SNF Model" or "Assembly Model". Cladding temperatures are also conservatively estimated using an empirical correlation. The effective thermal conductivity model results will be used for all WP analysis and will therefore be discussed in this section as part of the WP analysis basis (see Section 6.2.1.2).

The thermal environment of the WP in the repository will change with time and is affected by the heat produced in the WP. Therefore, the WP thermal evaluation must be a transient analysis that takes into account the time-varying heat load of the WP. This can be contrasted to an SNF storage cask analysis where the thermal environment is assumed constant and is provided by regulations.

The ANSYS V5.0A (SAS 1992) code has been approved for quality assurance work per the Management and Operating Contractor Quality Administrative Procedures for both thermal and structural analysis. ANSYS V5.0A is a commercially available finite-element thermal and mechanical analysis code. The analysis method used with the ANSYS code was compared to the methods used by the national laboratories, the results of which are presented in this section.

The Engineered Barrier Segment/WP thermal evaluation can be divided into three parts. The first is an analysis of the far-field repository thermal behavior, the second is an analysis of the near-field WP thermal behavior with boundary conditions from the repository analysis, and the third is an analysis of peak SNF cladding temperatures based on SNF basket temperatures and assembly heat load. Three finite element models were generated representing the repository, the WP, and an SNF assembly. The systems exhibit highly time-dependent behavior and must be modeled by a transient analysis, unlike an SNF storage or transportation analysis where steady state conditions can be assumed. To benchmark the WP emplacement thermal analysis, the emplacement scale thermal model using the ANSYS code was compared to models used by Sandia National Laboratories and Lawrence Livermore National Laboratory using the codes COYOTE and V-TOUGH, respectively.

The finite-element code COYOTE (SNL 1993), which is a thermal conduction solver, has been used by Sandia National Laboratories to develop a three-dimensional repository emplacement model similar to the ANSYS model used for WP design. This model has been used to predict near- and far-field repository thermal behavior for a variety of thermal loadings. Both the COYOTE model and the ANSYS model simulate the effects of boiling on temperature by assuming temperature-

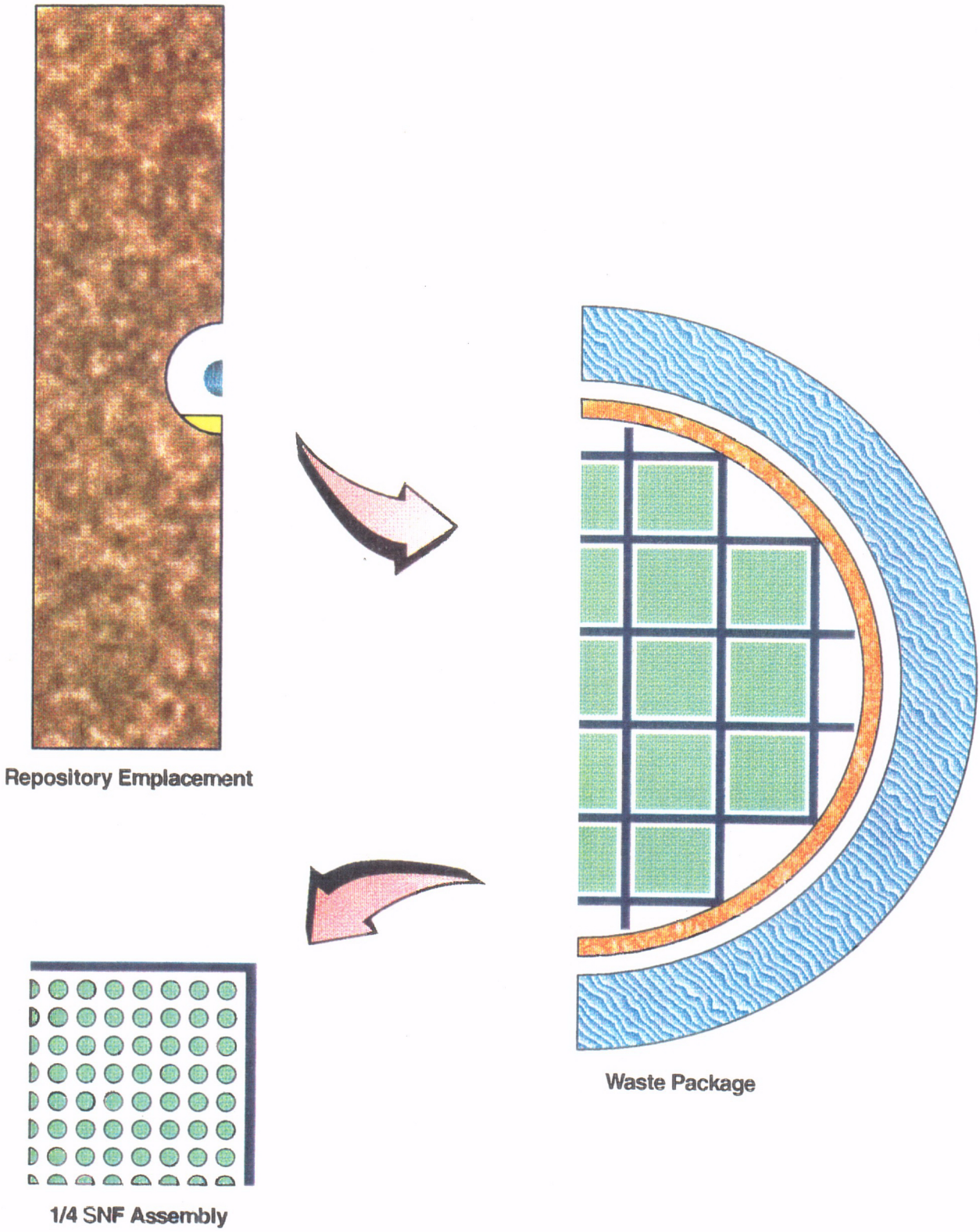

Figure 6.2-1. Three-Model Approach

C08

dependent heat capacitances provided in the *Reference Information Base* (YMP 1994a). The ANSYS model developed for WP design was run and compared to the COYOTE analysis with similar time-dependent SNF heat loads. The parameters of this case were as follows:

WP capacity	21 PWRs
Average SNF (CRWMS M&O 1995d)	22 years old, 42.2 Gwd/MTU burnup
Initial WP heat	10.2 kW
Drift diameter	4.3 m (14 ft)
Areal mass loading	111 MTU/acre
WP spacing	16 m
Drift spacing	20.5 m

Figure 6.2.2 displays the temperature comparison between the two models. At initial conditions, the ANSYS solution assumes a non-ambient temperature WP and the two models are expected to be different for the first year or two for this reason. However, WP surface temperatures in the COYOTE solution quickly rise from ambient and the two predictions differ by less than 5°C at the time of peak temperatures. Considering the different methods of calculating radiation heat transfer, the two solutions are nearly identical.

The benchmark test was also compared to evaluations performed at Lawrence Livermore National Laboratory. The V-TOUGH (LLNL 1994b) code, which numerically simulates the coupled transport of water, water vapor, air, and heat in fractured porous media, has been used by Lawrence Livermore to predict the hydrothermal behavior of the potential repository. While the V-TOUGH thermo-hydrological calculations include coupled water and heat transport, the complexity of the solution allows less detail in the WP near-field. The V-TOUGH code does, however, predict bulk water movement and emplacement humidity, which are critical to WP performance assessment. The following parameters were assumed for the analysis:

WP capacity	21 PWRs
Average SNF (CRWMS M&O 1994b)	First 2 years of youngest-fuel-first waste receipt
Initial WP heat	3.7 kW (Ramped to 11.2 kW at 1 year)
Drift diameter	7.6 m (25 ft)
Areal mass loading	140 MTU/acre
WP spacing	12 m
Drift spacing	21.7 m

Figure 6.2-3 displays the comparison results for the Lawrence Livermore National Laboratory benchmark. The inflection in the temperature behavior during the first year is due to the "ramp" increase in the average WP heat during that time, based on expected SNF receipt at the repository. After one year, the SNF type is fixed and the heat decay curve decreases smoothly on a logarithmic scale. Note also that Figure 6.2-3 is plotted on a logarithmic time scale.

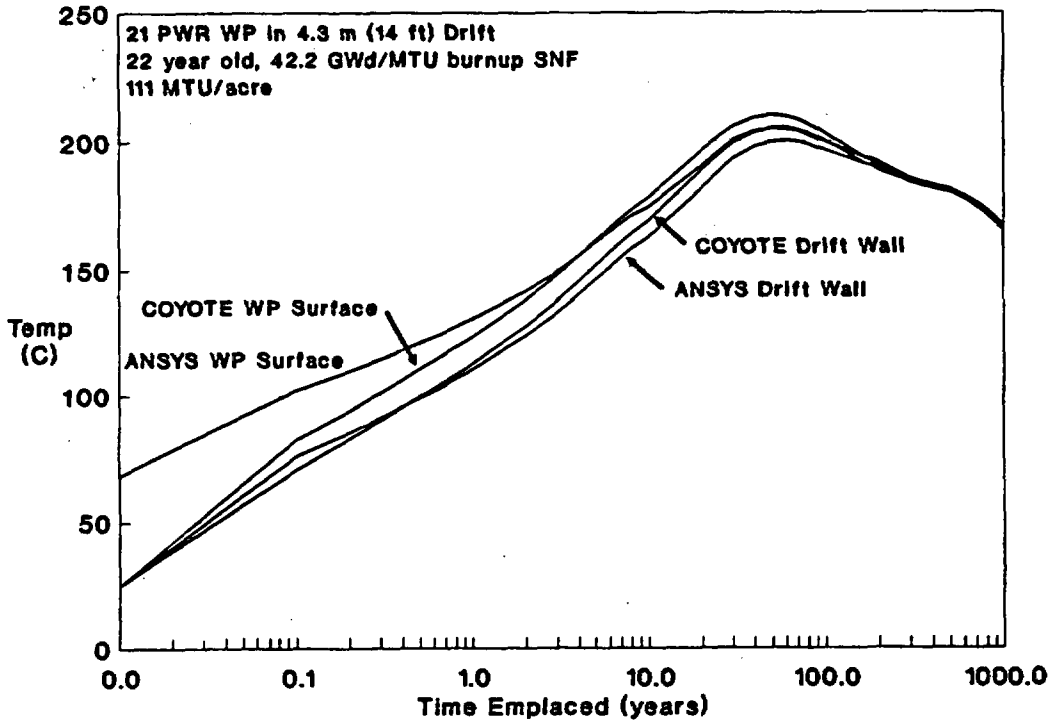

Figure 6.2-2. ANSYS/COYOTE Temperature Comparison

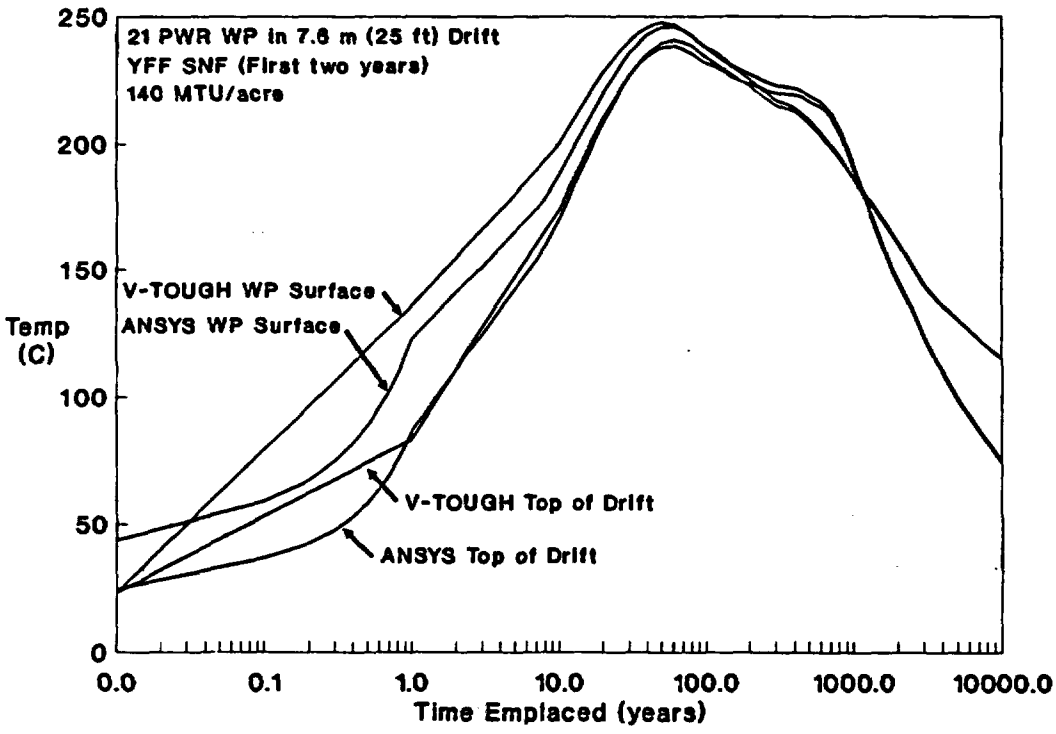

Figure 6.2-3. ANSYS/V-TOUGH Temperature Comparison

The V-TOUGH model conservatively over-predicts WP surface temperatures during the peak time compared to the ANSYS model, which has greater detail in the near-field. However, after 1,000 years, the ANSYS model over-predicts temperatures due to its assumption of an infinite repository compared to the V-TOUGH model, which assumes a disk-shaped repository to capture edge effects.

It is important to note that the analyses described here assumed a 22-year-old PWR assembly with a burnup of 42.2 Gwd/MTU (an assumed average). Therefore, the results describe only the average WP (i.e., approximately half of the WPs will be hotter). While all of the WPs (hot and cold) will collectively influence repository temperatures (average SNF characteristics), every WP must meet thermal goals (design basis SNF characteristics).

The benchmark calculations illustrate the strengths and weaknesses of each of the modeling tools available. It is obvious that, at this time, no one model can adequately describe all of the aspects involved in repository thermal loading. Each model was specifically developed to provide insight into a particular area or to answer specific questions about the thermal (or hydrological) behavior of the proposed repository at Yucca Mountain. However, it is valuable to compare results where model capabilities overlap.

6.2.1.1 Repository Thermal Loading

The first step in the thermal evaluation of the emplaced WP is to determine the effect of the WP decay heat on the repository host rock. The decay heat of individual WPs can vary greatly depending on the age, burnup, etc. of the loaded SNF. Figure 6.2-4 displays the heat produced after emplacement by several design basis SNF types and an assumed average of all SNF to be emplaced. The initial heat output from a design basis PWR assembly can be more than 75 percent greater than average, which can greatly affect the early temperatures of the repository. However, long-term thermal behavior of the repository rock is primarily a function of the areal mass loading or tons of SNF emplaced per acre. The repository response will be determined more by the integrated heat (the area under the curve) from the emplaced SNF and less by the initial heat of the individual WPs. The integrated heat per MTU over 10,000 years for the hottest design basis PWR assembly is only 14 percent greater than the average. Thus, an initial areal power density in kW/acre (W/m^2) will not adequately determine the thermal response because the areal power density will change differently with time, depending on the average SNF characteristics. Figure 6.2-5 displays the change in areal power density with time for two representative thermal loadings given in MTU/acre (kgU/m^2). For a given areal mass loading and assuming average SNF characteristics, the average host rock temperatures can be determined and then applied as the environment for a detailed thermal analysis of the near field.

To fully investigate the thermal loading effects in the near field (especially WP surface temperatures), a three-dimensional transient model is required. In previous two-dimensional models created by the M&O and national laboratories (CRWMS M&O 1994i), WP surface and drift wall temperatures represent the average down the length of the drift. While average drift wall temperatures can be predicted by a two-dimensional drift scale model for small WP spacings, WP surface temperatures will be under-predicted due to the axial smearing of the WP heat load. Therefore, a three-dimensional thermal model of an emplacement drift and WP has been constructed to capture WP-to-WP interactions, as well as drift-to-drift interactions. The ANSYS finite element

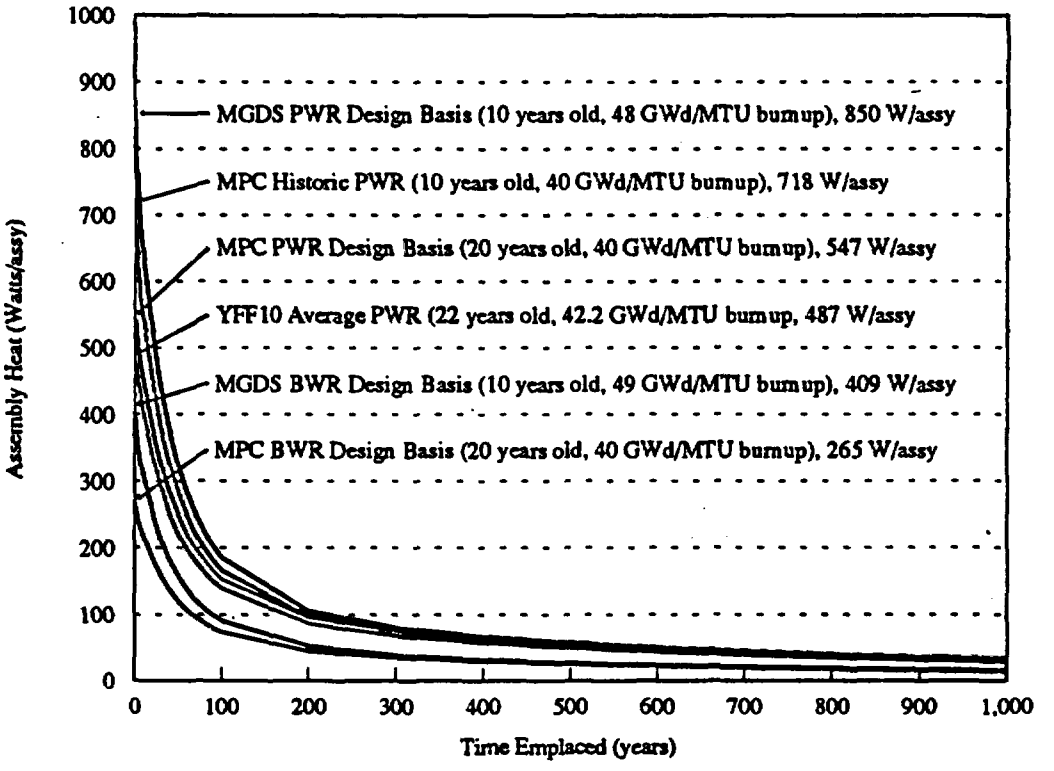

Figure 6.2-4. Decay of SNF Heat Over Time

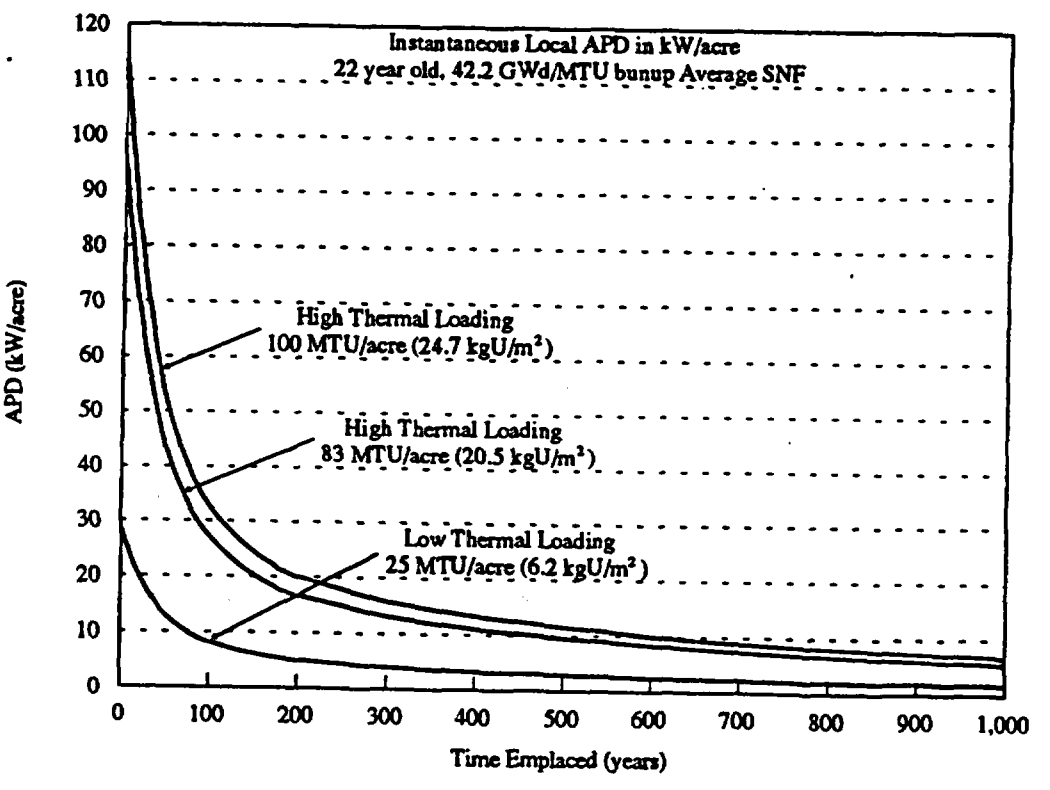

Figure 6.2-5. Time Decay of Areal Power Density

code was used to create the three-dimensional repository emplacement thermal model. The layered rock stratigraphy of the thermal model is shown in Figure 6.2-6. For the three-dimensional model, temperature-dependent rock material properties were taken from the *Reference Information Base* (YMP 1994a) and the WP is assumed to be a homogeneous cylindrical heat source with CF (assumed the conceptual MPC) outer dimensions a 0.1 m carbon steel overpack. The WP is assumed horizontally emplaced in a 5 m diameter drift with 0.7 m of concrete invert.

The three-dimensional model is a pillar section of the repository with planes of symmetry represented by two adiabatic boundaries halfway between the drifts and halfway between two WPs. The single-modeled WP is assumed to be in an infinite repository. The approximation of an infinite repository is reasonable up to 1,000 years or so after emplacement when repository thermal edge effects (not modeled here) will begin to influence the cool-down of the center of an actual repository. After about 1,000 years, depending on the spacings, the model used here will tend to over-predict near-field temperatures, so care must be taken in extrapolating to such long times. Benchmarks against V-TOUGH calculations at Lawrence Livermore National Laboratory, described in Section 6.2.1, quantify the effects of neglecting repository edge effects after 1,000 years.

Analyses performed at Lawrence Livermore National Laboratory (LLNL 1994a) have shown that there may be insufficient water circulation at the water table, due to low permeabilities, to maintain a constant temperature at that depth. Therefore, the three-dimensional emplacement model has a bottom boundary condition at a depth of 1,300 m instead of at the water table depth of about 611 m. This depth will ensure that the thermal behavior of the repository horizon will not be affected by this boundary condition assumption for over 1,000 years of model time.

Assuming that there is no forced ventilation of the drifts after WP emplacement, the only significant heat transfer paths between the WP and the drift wall would be thermal radiation and natural convection of the drift air. Initial calculations of potential free-convection cooling cells in the drift have indicated that natural convection heat transfer is much smaller than thermal radiation heat transfer for in-drift emplacement. Also, Sandia National Laboratories (SNL 1982) showed that radiation heat transfer in a closed *Site Characterization Plan*-type emplacement drift is an order of magnitude greater than convective heat transfer. To be conservative, the complex calculations of natural convection have been neglected and thermal radiation is assumed to be the only heat transfer path between the WP and the drift.

Previous analyses have investigated the effects of drift diameter on the near-field temperature (CRWMS M&O 1994i). During the operation phase of the repository, a somewhat smaller drift could result in about 10°C higher near-field temperatures. The effect of smaller drifts will increase as the diameter is decreased, bounded by the case of horizontal borehole emplacement, which is prohibitively hot for large WPs. The temperature drop from WP surface to drift wall will depend on the surface area of the drift wall that can receive heat from the WP. Smaller drift diameters have a smaller drift wall surface area and allow the WP to spread less of its heat down the emplacement drift length. At later times (hundreds of years after emplacement), the temperature drop from WP surface to drift wall is small and drift diameter is less important, unless the drift is backfilled with a relatively insulating material. Calculations of the effect of drift backfilling are discussed in Section 6.2.1.1.3.

6.2.1.1.1 Large WPs, 21 PWR and 40 BWR

To determine the effect of the large WP on repository near-field temperatures, the three-dimensional emplacement model was evaluated for a range of thermal loadings. The dimensions of the modeled WP were assumed to be those of the 21 PWR CF (assumed the conceptual MPC) with disposal container, which is representative of a large WP. The analysis is detailed in a supporting emplacement thermal model design analysis (CRWMS M&O 1995d). A parametric set of thermal loading cases, summarized in Table 6.2-1, is required because the thermal loading for the repository has been specified as a range in the *Controlled Design Assumptions Document* (CRWMS M&O 1995n) and the CF container must be shown to meet requirements over that range. Representative "high" and "low" thermal loadings of 100 MTU/acre (24.7 kgU/m^2), 83 MTU/acre (20.5 kgU/m^2), and 25 MTU/acre (6.2 kgU/m^2) were selected to be consistent with Volume II, Section 8.3.3.5 of this report.

Table 6.2-1. 21 PWR Thermal Loading Scenarios

Areal Mass Loading MTU/acre	Initial Areal Power Density kW/acre	WP Spacing (m)	Drift Spacing (m)
100 (high #1)	113.7	16.2	22.5
83 (high #2)	94.4	19.5	22.5
25 (low #1)	28.4	32.3	45.0
25 (low #2)	28.4	21.6	67.5
25 (low #3)	28.4	16.2	90.0

The low thermal loading drift spacings described in Table 6.2-1 represent the use of even multiples of the high thermal loading drift spacing (every other drift, every third, etc.). Low thermal loading #1 achieves the longest reasonable WP spacing and results in the lowest near-field temperatures; low thermal loading #2 has a WP spacing similar to that for high thermal loading #2; and low thermal loading #3 has a short WP spacing and represents the "localized disturbance" concept for low loadings. The "localized disturbance" concept depends on high temperature gradients near the emplaced waste package (due to the closely spaced WPs and widely spaced drifts) to drive local water in the rock away from the emplacement drifts. The viability of this concept depends on drift-scale hydrothermal effects, which are not modeled here; however, the temperature response has been determined.

It is important to note that the analyses described here are for a 22-year-old PWR assembly with a burnup of 42.2 GWd/MTU (an assumed average). Therefore, the results describe only the average WP (i.e., about half of the WPs will be hotter). The detailed WP evaluations, which use the results of these analyses as boundary conditions, estimate internal WP temperatures for design basis SNF characteristics to ensure that thermal goals have been attained. For a pillar model that represents one average WP in the repository, the heat of that package must be the average of all of the emplaced

Rock Layers from
Surface to Water Table
(Single WP in Center)

Figure 6.2-6. Repository-Scale Finite-Element Model

C09

WPs. While all the WPs (hot and cold) will collectively influence repository temperatures (average SNF characteristics), every WP must meet thermal goals (design basis characteristics).

Representative temperature contours for the 21 PWR WP and around the drift for low thermal loading #3 are displayed in Figures 6.2-7 through 6.2-15. The figures show the WP (cut through the mid-length) at 0, ½, 1, 5, 10, 50, 100, 500, and 1,000 years after emplacement. The red colors indicate temperatures above the boiling point (100°C) to illustrate the penetration of above-boiling temperatures at this "low" thermal loading. The short WP spacing of low thermal loading #3 results in boiling temperatures 3 m into the drift wall in 20 years; however, temperatures drop to below boiling in less than 200 years.

Figure 6.2-16 displays a comparison of maximum WP surface temperatures for the five thermal loading scenarios on a logarithmic time scale. As expected, low thermal loading #3 actually has higher near-field temperatures than high thermal loading #2 for the first five years due to its shorter WP spacing. However, by 500 years, the three low thermal loading temperatures have nearly converged to a single temperature that is much lower than that for both of the high loadings. Peak WP surface temperatures occur in 40 to 50 years for the high loadings and in about eight years for the low loadings. The figure demonstrates a significant difference in short-term, near-field temperatures between short and long WP spacings. If the goal of a low thermal loading scenario is to maintain low near-field temperatures, a low areal mass loading and a long WP spacing are needed.

Figure 6.2-17 compares maximum drift wall temperatures for the five loadings, and Figure 6.2-18 compares the maximum temperature at 3 m into the drift wall. Maximum temperatures always occurred at axial locations corresponding to the mid-length of the WP. Just as for the WP surface temperatures, the peak temperatures occurred earlier for the low loadings than for the high thermal loadings. Figure 6.2-19 compares the drift-to-drift midplane temperatures and demonstrates that the WP spacing has little effect on temperatures half-way between drifts. At the drift-to-drift midplane, temperatures are primarily a function of thermal loading in MTU/acre. For all of the thermal loading scenarios, the drift wall temperature limit of 200°C was not exceeded; however, the TSw2/TSw3 interface reached 118°C (greater than the limit of 115°C) at 1,000 years for the highest loading. The TSw2/TSw3 interface reached 106°C at 1,000 years for high thermal loading #2.

The emplacement thermal evaluations described above provide the boundary conditions for more detailed evaluations of specific WP concepts. In the three-dimensional emplacement model, the WP is represented by a homogeneous heat source such that the resulting near-field temperatures can represent any similar-capacity WP. For the purpose of the WP thermal evaluations in Sections 6.3 and 6.5, the surface temperatures predicted by the 21 PWR WP emplacement model can be assumed to apply to 21 PWR, 40 BWR, or 44 BWR detailed thermal evaluations for both the CF and UCF WPs. Using PWR surface temperatures for the BWR analysis may be somewhat conservative as PWR package heat loads are generally greater than BWR heat loads. DHLW WP thermal evaluation in Section 6.4 will use the drift wall temperatures predicted by the emplacement model.

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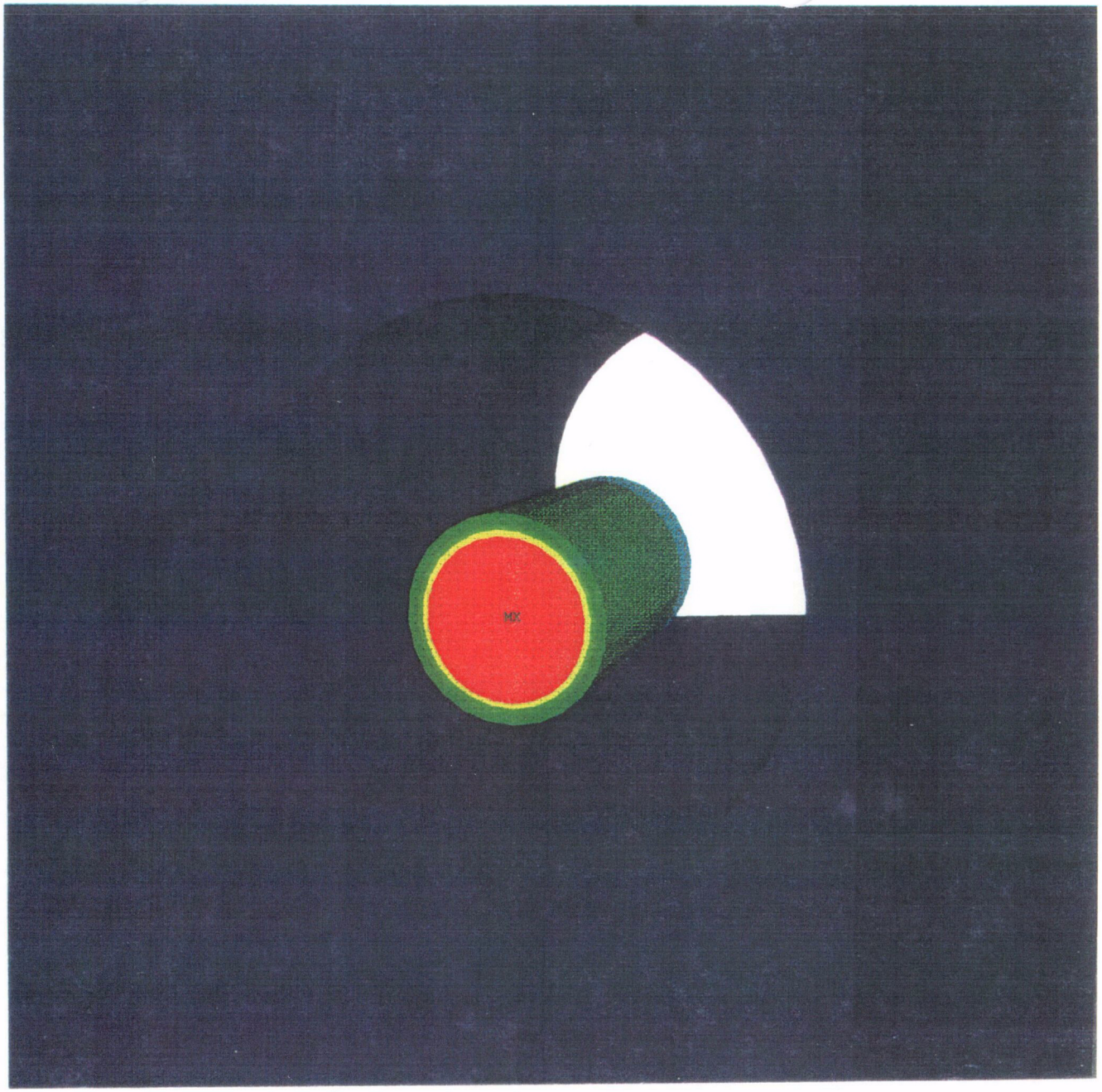


ANSYS 5.0 A
JAN 25 1995

Temperature
18
40
50
60
70
80
90
100
100+

Degrees C

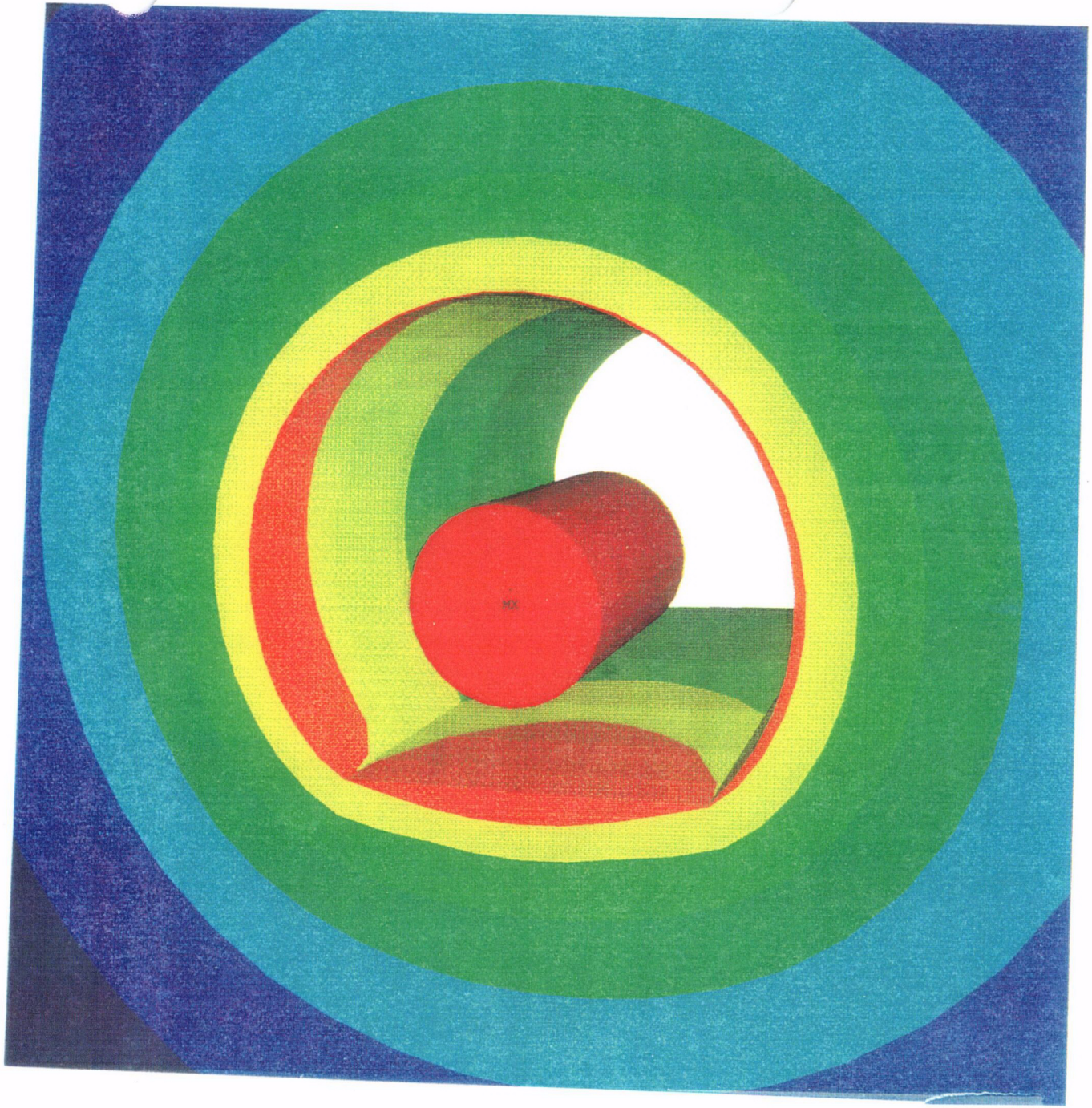
21 PWR WP
Mid-plane
Temperatures

25 MTU/acre
(16.2 m WP Spacing)
(90 m Drift Spacing)

22 year old SNF
42.2 GWd/MTU

C10

Figure 6.2-7. Low Thermal Load Initial Emplacement Temperatures



ANSYS 5.0 A
JAN 25 1995

Temperature
18
40
50
60
70
80
90
100
100+

Degrees C

21 PWR WP
Mid-plane
Temperatures

25 MTU/acre
(16.2 m WP Spacing)
(90 m Drift Spacing)

22 year old SNF
42.2 GWd/MTU

C11

Figure 6.2-8. Low Thermal Load Emplacement Temperatures at 6 Months

21 PWR WP
Mid-plane
Temperatures

25 MTU/acre
(16.2 m WP Spacing)
(90 m Drift Spacing)

22 year old SNF
42.2 GWd/MTU

C12

Figure 6.2-9. Low Thermal Load Emplacement Temperatures at 1 Year

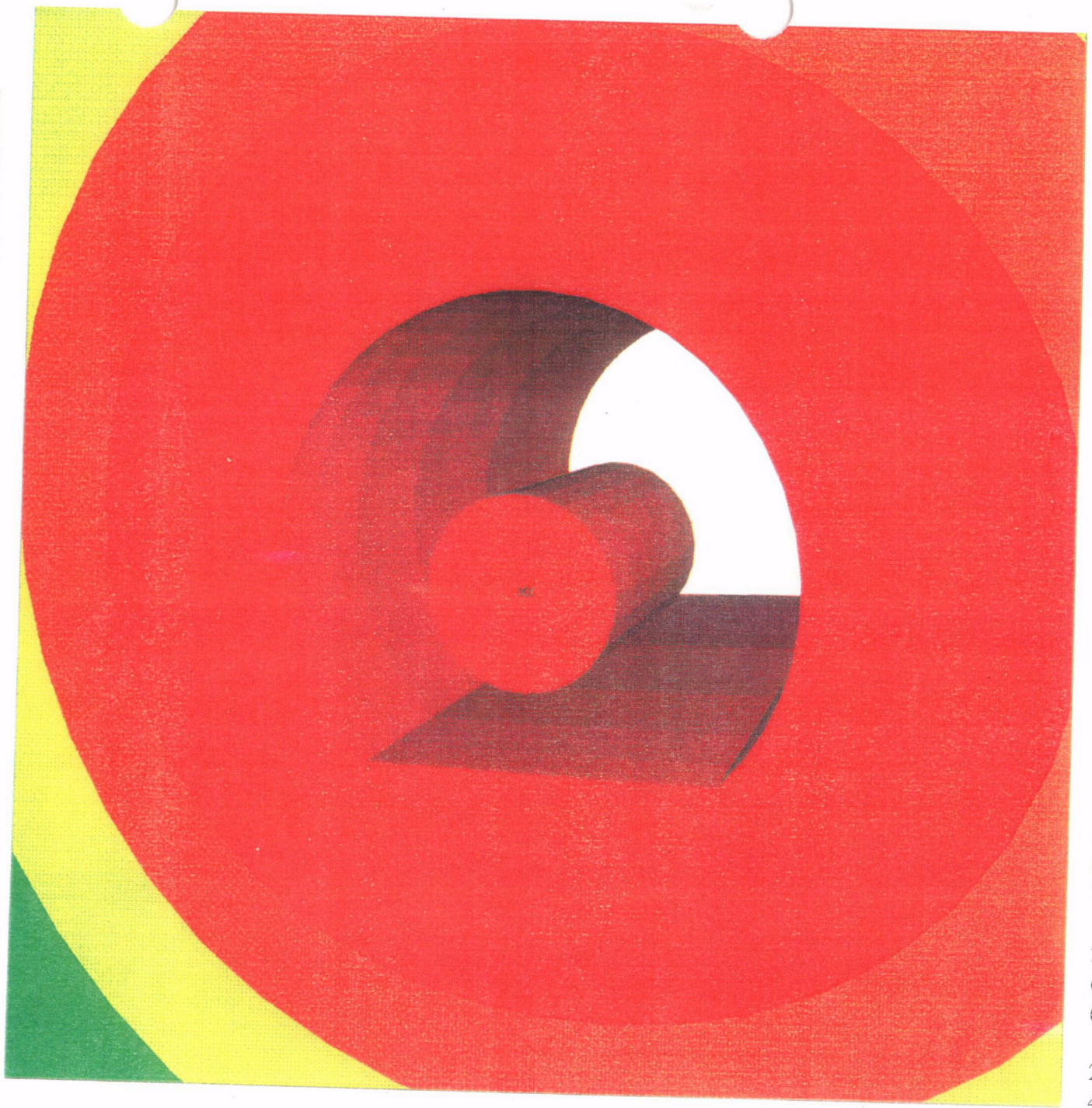
21 PWR WP
Mid-plane
Temperatures

25 MTU/acre
(16.2 m WP Spacing)
(90 m Drift Spacing)

22 year old SNF
42.2 GWd/MTU

C13

Figure 6.2-10. Low Thermal Load Emplacement Temperatures at 5 Years



21 PWR WP
Mid-plane
Temperatures

25 MTU/acre
(16.2 m WP Spacing)
(90 m Drift Spacing)

22 year old SNF
42.2 GWd/MTU

C14

Figure 6.2-11. Low Thermal Load Emplacement Temperatures at 10 Years